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SAFETY ANALYSIS

UTILIZATION OF INTERMETALLIC URANIUM ALUMINIDE (UAl_3, UAl_4, UAl_2) AND URANIUM OXIDE (U_3O_8) CERMET FUEL CORES IN THE FORD NUCLEAR REACTOR

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1. INTRODUCTION

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Intermetallic uranium aluminide (UAl_3, UAl_4, UAl_2) and uranium oxide (U_3O_8) cermet fuels differ from uranium - aluminum alloy fuels in that the meat is formed by high pressure compaction of uranium-bearing powder and aluminum matrix powder into a solid. The compacted meat is clad in the same manner as a fuel meat casting of uranium - aluminum alloy. Powder metallurgy produces a more uniform dispersion of uranium throughout the aluminum matrix and provides better control of the fuel plate uranium content than can be obtained with uranium - aluminum alloy.[3]

This analysis describes irradiation tests conducted on aluminide and oxide fuel plates between 1958 and 1976. Aluminide and oxide test plates were irradiated in the Materials Test Reactor (MTR), the Engineering Test Reactor (ETR), and the Advance Test Reactor (ATR) at the National Reactor Test Station (NRTS), Idaho Falls, Idaho, and the High Flux Isotope Reactor (HFIR) at the Oak Ridge National Laboratory, Oak Ridge, Tennessee. Aluminide test plates were irradiated in the German Karlsruhe FR2 reactor.

The primary parameter examined in irradiation tests was fuel core swelling ($\%\Delta V/V$) which could cause fuel plate failure. Post-irradiation tests were conducted by heating of test plates until blisters appeared indicating clad failure.

2. OPERATING EXPERIENCE

The MTR and the ETR have successfully used both alloy and aluminide fuels. Aluminide fuel has been used in the ATR since 1967. To date, the ATR has consumed approximately 19,000 plates in 300,000 MWD of operation. Aluminide fuel has been used in the University of Missouri Research Reactor (MURR) since 1970 and the Massachusetts Institute of Technology Reactor (MITR) since resumption of operations in 1975.

Since 1965, over 76,000 U_3O_8 fuel plates have been operated to depletion at the HFIR with no major difficulties. On two occasions during that period, fuel elements developed suspected fission product leaks. In one case, the apparent leak was so insignificant that the element was operated to depletion. In the second case, the element was removed after 1500 MWD.[3] Destructive examinations of irradiated test specimens and actual HFIR fuel plates have produced no evidence of blisters, cladding separation, matrix cracking, or any defects indicative of incipient failure.

Aluminide and oxide fuels were developed and used because it was found that those materials permitted better process control and produced fuel elements superior in quality to those produced from alloys.[3] For example, in 1967 at the ETR, a partial fuel element melting occurred due to a blocked flow channel. Although the fission product inventory in the fuel was substantial, the remarkable ability of aluminide fuel material to retain fission products permitted easy cleanup of the reactor and return to power within two days.[5]

Table 1 provides operating parameters for the primary users of aluminide and oxide plate fuels and compares them with similar parameters for the Ford Nuclear Reactor (FNR). Each of the reactors using plate-type fuel utilizes plates of approximately the same meat thickness, nominally 0.02 in, with clad thicknesses that vary from approximately 0.01 to 0.02 in. With these dimensions being relatively constant, it can be seen from Table 1 that the significant operating parameters of thermal power density, peak fission density, and heat flux are more severe by orders of magnitude for the reactors listed than the same parameters for the Ford Nuclear Reactor.

Tests referred to in succeeding sections were performed on fuel plates with meat and clad thicknesses nominally the same as those described above.

3. PHYSICAL CHARACTERISTICS

Aluminide and oxide cores are not as strong as alloy cores. However, FNR elements are not operated near any high stress limit. A slight strength reduction is not a serious analytical consideration. The basic strength and integrity of the plate cladding remains unchanged.

Aluminide and oxide cores are more malleable than alloy cores. In the manufacturing process, thickening of the fuel core at the ends during the plate-rolling operation is eliminated.[3] The result is more uniform clad thickness which adds to overall strength and assures a uniform fission product barrier.

The melting temperature of alloy cores is about 1560° F. The melting temperature of aluminide and oxide cores is generally considered to be the melting point of the 1100 Al matrix, about 1200° F. Except in cases of complete flow blockage at high power or violent transients, none of these temperatures are expected to be approached. In any case, it is the disruption of the fuel clad which could result in the release of fission products, which would occur at about the same temperature for any type of fuel. Moreover, the type of event that could produce such elevated temperatures is characterized by essentially complete insulation of the fuel by film boiling wherein the vapor generated completely covers the heat transfer surface.[12] The rate of temperature rise is of the order of thousands of degrees per second. Melting of an alloy core would occur only a few milliseconds later than melting of an aluminide or oxide core.[3]

4. REACTOR PHYSICS

No reactor physics changes will occur as a result of the conversion from alloy to aluminide or oxide fuel meat. The reactor core flux distributions and rod worths will be unaffected since all physical dimensions and characteristics will be unchanged.

5. FUEL SWELLING

Aluminide and oxide fuels exhibit lower swelling rates than alloy fuels. In

TABLE 1

TRAINING, RESEARCH, AND TEST REACTOR OPERATING PARAMETERS

Parameter	Materials Testing Reactor (MTR)[1]	Engineering Test Reactor (ETR)[1]	Advance Test Reactor (ATR)[1]	High Flux Isotope Reactor (HFIR)[3]	High Flux Brookhaven Reactor (HFBR)[4]	Ford Nuclear Reactor (FNR)	
Year placed in service	1952	1956	1967	1965	1965	1958	
Thermal power (MW)	40	175	40	100	40	2	
Thermal power density (MW/l)	0.75	1.2	2.8	1.5	0.5	.025	
Fuel element meat volume (cc)	365	550	798	3475	870	354	
U-235 per element (gm)	200	400	975	2600	315	140	
U-235 burnup (%)	-	25	25	30.6	34	35	
Peak fission density (fiss/cc)	-	1.8×10^{21}	1.8×10^{21}	1.9×10^{21}	1.24×10^{21}	5.44×10^{20}	Rev 4/78
Fuel element surface area (ft^2)	15	23	34	147	36	15	
Heat flux (BTU/ft^2-hr)	3.5×10^5	5×10^5	4×10^5	2.5×10^6	3.8×10^5	3.68×10^4	Rev 4/78
Fuel surface temperature ($^{\circ}$ F)	239	329	356	300	304	159	
Coolant flow rate (gpm)	24,000	44,000	16,000	17,000	16,600	980	
Fuel element materials:							
Cladding	1100 Al	1100 Al	6061 Al	6061 Al	6061 Al	1100 Al	
Core (wt% Uranium)	18% U-Al Alloy and UAl_x	22% U-Al Alloy and UAl_x	41% UAl_x	41% U_3O_8	30% U_3O_8	14% U-Al Alloy	

the former, swelling rate decreases with increasing uranium concentration due to the fact that the viod fraction of the fuel plates tends to increase with increasing uranium content. For fuel plates with high void fractions, early core life swelling is absorbed by the voids and the onset of swelling is delayed until later in core life.[6]

5.1 National Reactor Test Station (NRTS) Fuel Swelling Tests

Figure 1 presents swelling information from 1958 NRTS tests performed on aluminide fuels.[7] The percent volume change in the fuel meat is related to the peak burnup in fissions/cc of fuel meat for both alloy and aluminide fuels. As shown, initial void content in aluminide fuel is an important parameter in fuel swelling considerations. Expected fuel core swelling with no manufacturing voids is 6.38% $\Delta V/V$ per 10^{21} fissions/cc.[18] Expected swelling curves for UAl_x fuel cores with 4% and 7% voids are shown.

NRTS initially set a 7% $\Delta V/V$ swelling limit for the flat plate ETR geometry.[8] The limiting swelling criteria was selected because some warping, attributed to swelling, was periodically observed in flat sample platelets irradiated to the 7% $\Delta V/V$ swelling level. In the more stable curved geometry of FNR fuel, a 7% $\Delta V/V$ would not result in any failure mode. In fact, the upper limit on fuel plate swelling can be based upon: 1) Integrity of the fuel plate cladding and 2) Thermal-hydraulic considerations such as flow blockage. In a practical sense, the upper limit can be the maximum experimentally or operationally verified swelling wherein clad rupture does not occur. A high percentage volume change in the fuel plate results in a very low percentage reduction in fuel channel area. For example, in the FNR geometry, a 20% change in the fuel core thickness (.004 inches) results in a ?.4% change in the .117 inch flow channel thickness and the flow area. Such a change is not a significant operational consideration.

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As can be seen on Figure 1, the maximum FNR fission density results in a 3.5% $\Delta V/V$ in alloy fuels and no swelling in aluminide fuels because of the void fraction.

The results of NRTS irradiation tests conducted in 1967 on aluminide fuel plates in the MTR are shown in Table 2 and Figure 2.[1] For core fission densities up to 1.4×10^{21} fissions/cc, swelling reached a maximum of approximately 7.5% $\Delta V/V$.

Additional results of NRTS irradiation tests conducted in 1967 on aluminide fuel plates in the ETR are shown in Table 3 and Figures 3A and 3B.[1] Swelling in aluminide and oxide cores was consistently less than swelling observed in alloy cores as can be seen in Figure 3A. Up to fission densities of 8.25×10^{20} fissions/cc, the maximum alloy swelling observed was 5.25% $\Delta V/V$ while the maximum aluminide swelling was 4.5% $\Delta V/V$.

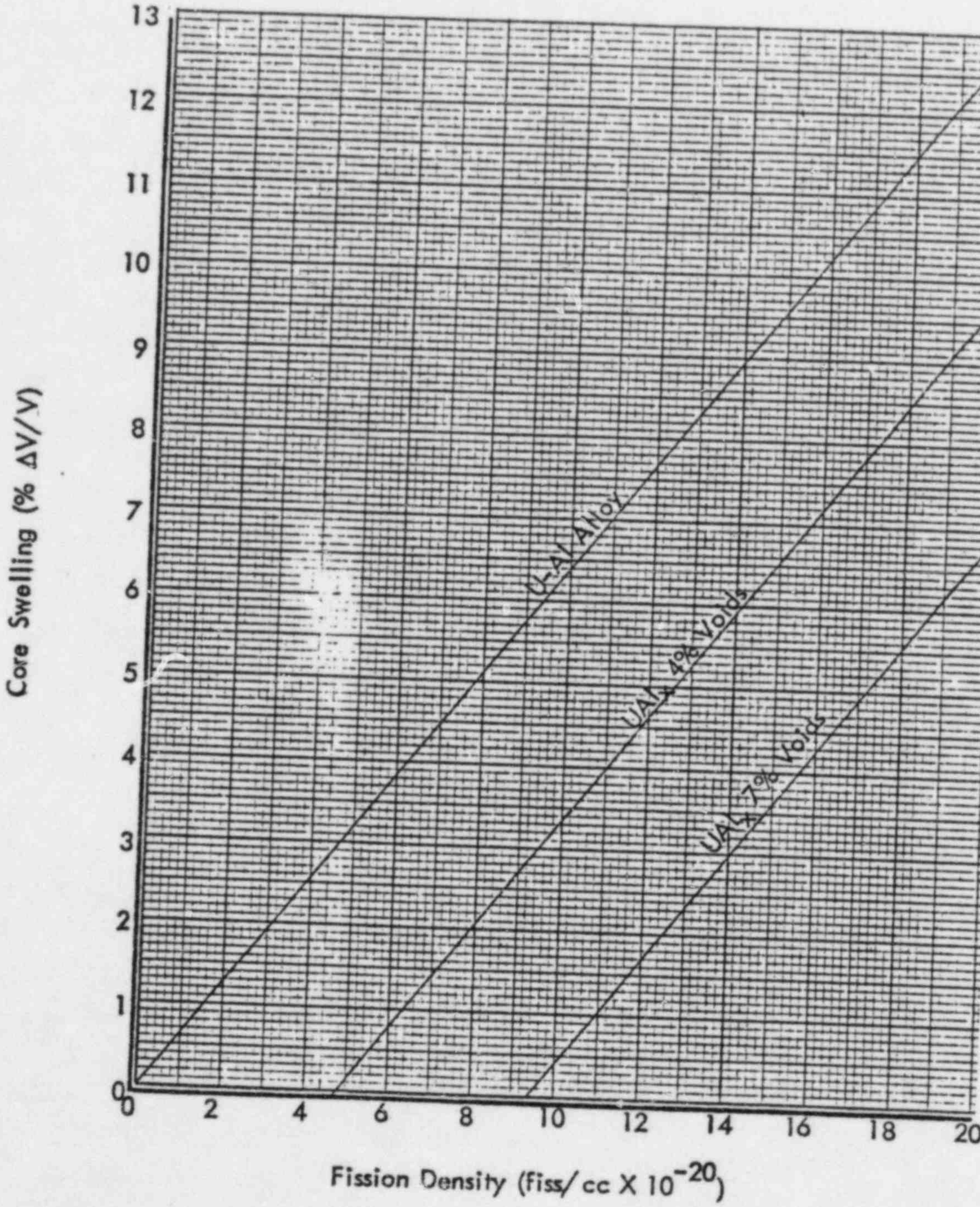


FIGURE 1

IRRADIATION SWELLING OF URANIUM - ALUMINUM FUELS[7]

TABLE 2

DATA ON SPECIMENS CONTAINING POWDERED UAl_3 IRRADIATED IN L-51 POSITION OF MTR[1]

Sample Composition Number	Number Samples Tested	Cladding Alloy	Core Matrix Alloy	UAl_3 in Core (wt%)	Heat Transfer Surface (ft^2)	U-235 per cc of Core (g)	Core Fission Densi (\$10^{20}$ fiss/c
113	15	X8001	X8001	46.7	27.6	1.08	20
114	10	6061	6061	46.7	27.2	1.09	19
116	9	1100	1100	29.6	15.9	0.597	12
118	6	XAP001	1100	47.0	27.6	1.20	19

Total = 40

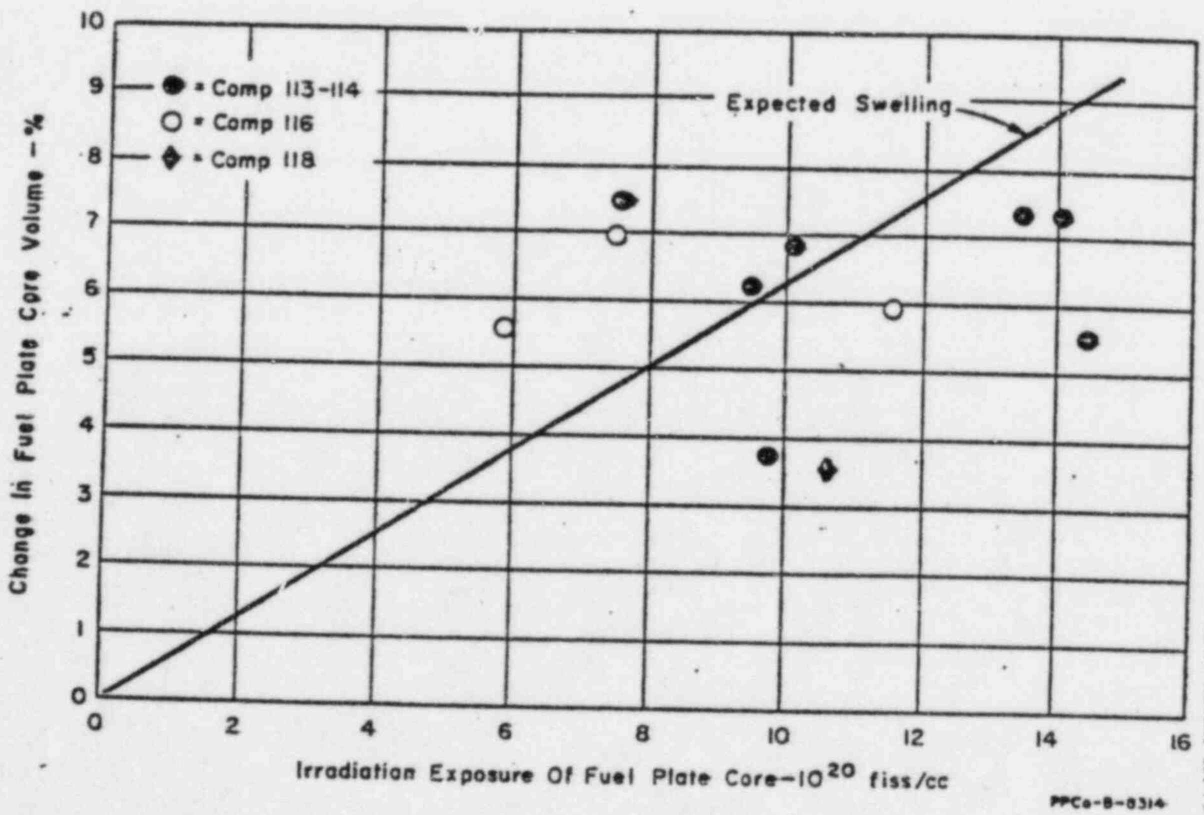


FIGURE 2

SWELLING PRODUCED IN POWDERED UAl_3 FUEL PLATE CORES BY IRRADIATION IN L-51 POSITION OF THE MTR[1]

TABLE 3

DATA ON SPECIMENS IRRADIATED IN J-8 POSITION OF ETR[1]

Sample Composition Number	Number Samples Tested	Cladding Alloy	Core Matrix Alloy	Fissile Compound	Fissile Compound in Core (wt%)	U235 per cc of Core (g)	Oxide Coating Thickness (mils)	Calculated Plate Surface Temp (°C)	Core Fission Density (10^{20} fiss/cc)
ATR 1	4	X8001	X8001	U_3O_8	34.9	0.93	2	170	6.8 to 7.7
ATR 2	4	X8001	X8001	UO_2	32.0	0.90	2	140	3.4 to 6.0
ATR 3	4	X8001	X8001	UAl_3	35.4	0.78	2	170	5.3 to 8.7
ATR 4	4	6061	X8001	UAl_3	48.2	1.16	1	180	10.4 to 11.2
ATR 5	4	6061	X8001	U_3O_8	44.2	1.29	1	170	6.5 to 11.4
ATR 6	4	6061	X8001	UO_2	41.2	1.24	1	180	12.3 to 12.5
ATR 7	4	6061	X8001	U_3O_8	34.9	0.95	2	180	7.3 to 7.8
ATR 8	4	6061	X8001	UO_2	32.9	0.89	2	150	5.7 to 6.1
ATR 9	4	6061	X8001	UAl_3	35.4	0.79	2	180	5.3 to 8.7
ETR 10	4	1100	22 wt% U-Al			0.68	-	100	7.1 to 8.3
ETR 11	4	1100	1100	UAl_3	30.9	0.65	-	100	8.2

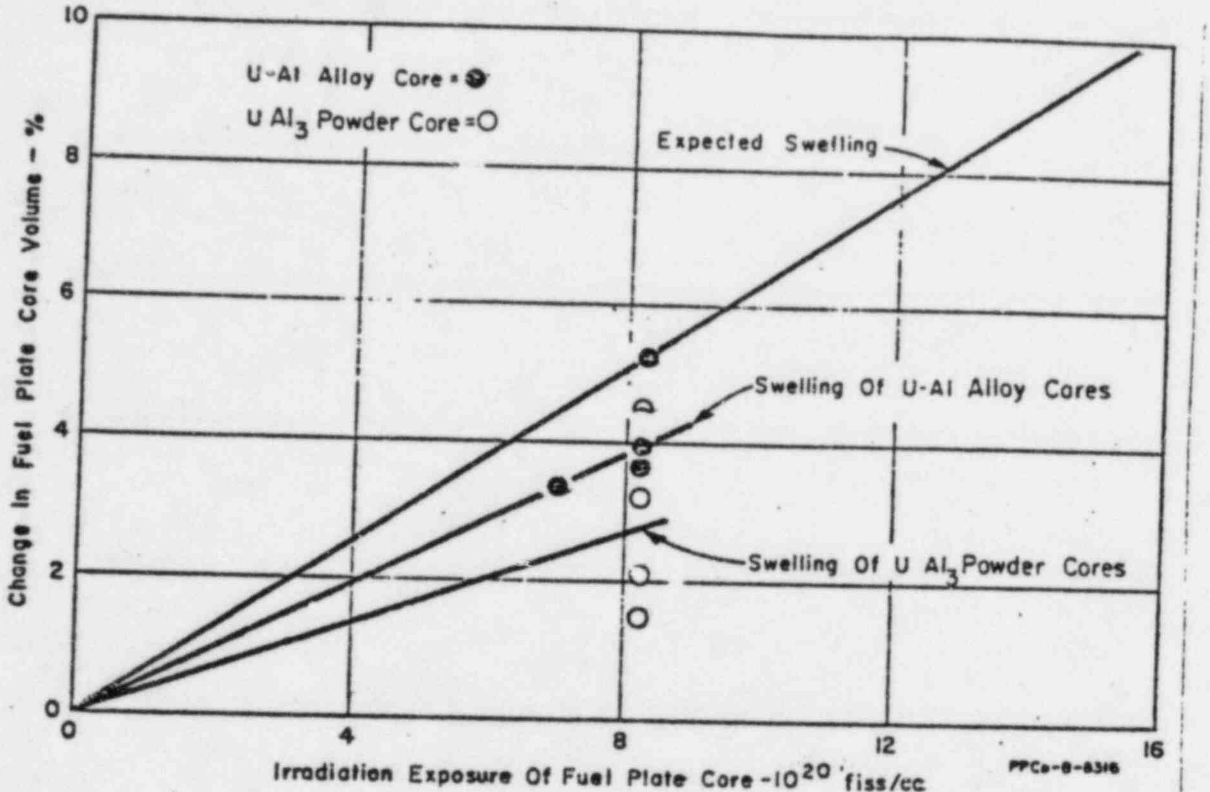


FIGURE 3A

SWELLING PRODUCED IN POWDERED UAl_3 ALLOY FUEL PLATE CORES BY IRRADIATION OF SAMPLES WITHOUT A THERMAL BARRIER COATING IN THE J-8 POSITION OF THE ETR[1]

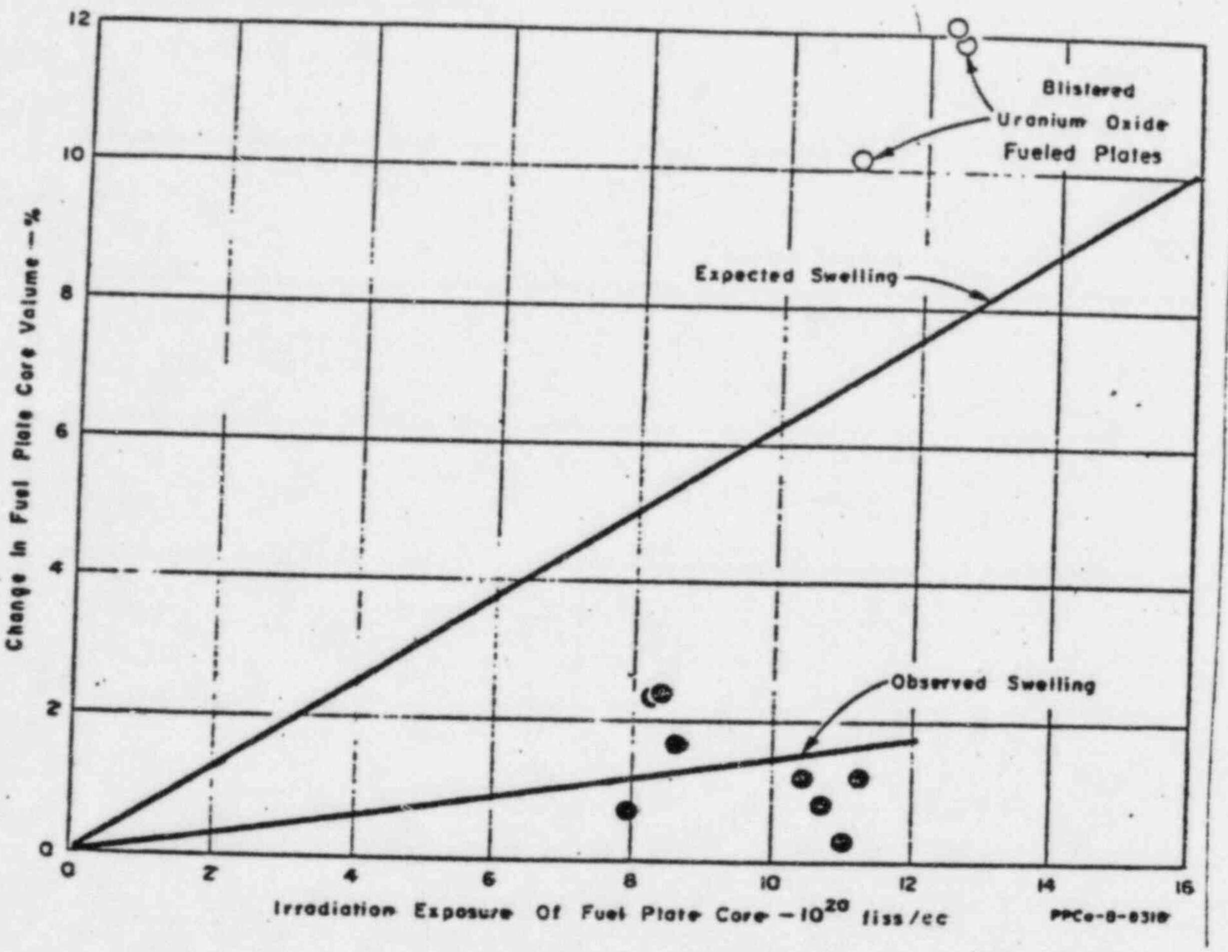


FIGURE 3B

SWELLING PRODUCED IN POWDERED UAl_3 FUEL PLATE CORES BY IRRADIATION OF ALUMINUM-CLAD PLATES COATED WITH A THERMAL BARRIER OF Al_2O_3 IN THE J-8 POSITION OF THE ETR CORE[1]

The maximum aluminide swelling from Figure 3B was 2.5% $\Delta V/V$ for fission densities up to 11×10^{20} fissions/cc.

Another set of 1967 NRTS irradiation test results for aluminide fuel plates in the ETR is shown in Table 4 and Figure 4.[1] Maximum observed swelling was approximately 5% $\Delta V/V$ up to fission densities of 1.6×10^{21} fissions/cc.

Continuing studies at NRTS of reactor fuels and the effects produced by fissioning have led to a better understanding of the swelling mechanism.[5] Swelling is produced primarily by solid fission products. Figure 5 shows calculated and experimental swelling data for several fuel systems. The strong tendency of fuel porosity to contain the initial swelling is well illustrated. The inherent lower density of fuels made by powder metallurgy offers benefits in terms of fuel stability.

Experiment INC-16-1 was conducted at NRTS in 1971 on a single UAl_x fuel element designated as XA003F. Individual plates from the element were examined.[16]

Nineteen plates from ATR elements similar to XA003F were measured for length and width changes caused by fission product swelling. The average length change was plus .22%; average width change was plus .13%.[16] These values establish that swelling volume changes occur almost entirely in the thickness of the fuel plate.

Measured fuel plate swelling, based upon thickness changes at three locations on three XA003F plates, is tabulated in Table 6. A plot of the swelling data, Figure 6, shows that, with the exception of sample 3-3, swelling does not exceed the predicted value of 6.38% $\Delta V/V$ per 10^{21} fissions/cc.

The INC-16-2 experiment in 1974 consisted of nine UAl_x ATR composition fuel plate specimens being irradiated in the ETR for 55,612 MWD at fluences of $1.75 - 2.26 \times 10^{14}$ n/cm^2/sec.[17] Specimen data are provided in Table 7.

Measured core swelling values were considerably less than expected by calculation. Figure 7 is a plot of fuel core swelling versus fission density for the experiment.

Visual inspection of the test plates showed surface discoloration but no indication of failure. Photomicrographs revealed the irradiated fuel to have a sound microstructure.

5.2 Oak Ridge National Laboratory (ORNL) Fuel Swelling Tests

Test irradiations were conducted at ORNL in HFIR to measure the performance and physical characteristics of aluminide and oxide fuel dispersions.[2] Fuel plate data are shown in Table 8. Swelling data are shown in Figure 8. The U_3O_8 identified as powder blend, PB-01, is the oxide presently used at HFIR.[9] UAl_x blend, PB-32, is a material similar to that used in the ATR.

TABLE 4

DATA ON SPECIMENS IRRADIATED IN G-12 PRESSURIZED WATER LOOP OF ETR[1]

Sample Composition Number	Number Samples Tested	Cladding Alloy	Core Matrix Alloy	Fissile Compound	Fissile Compound in Core (wt%)	U-235 per cc of Core (g)	Core Fission Density (10^{20} fiss/cc)
1	11	6061	X8001	UAl_3	48	1.17	7.1 to 19
2	6	6061	X8001	UAl_3	52	1.31	6.4 to 14

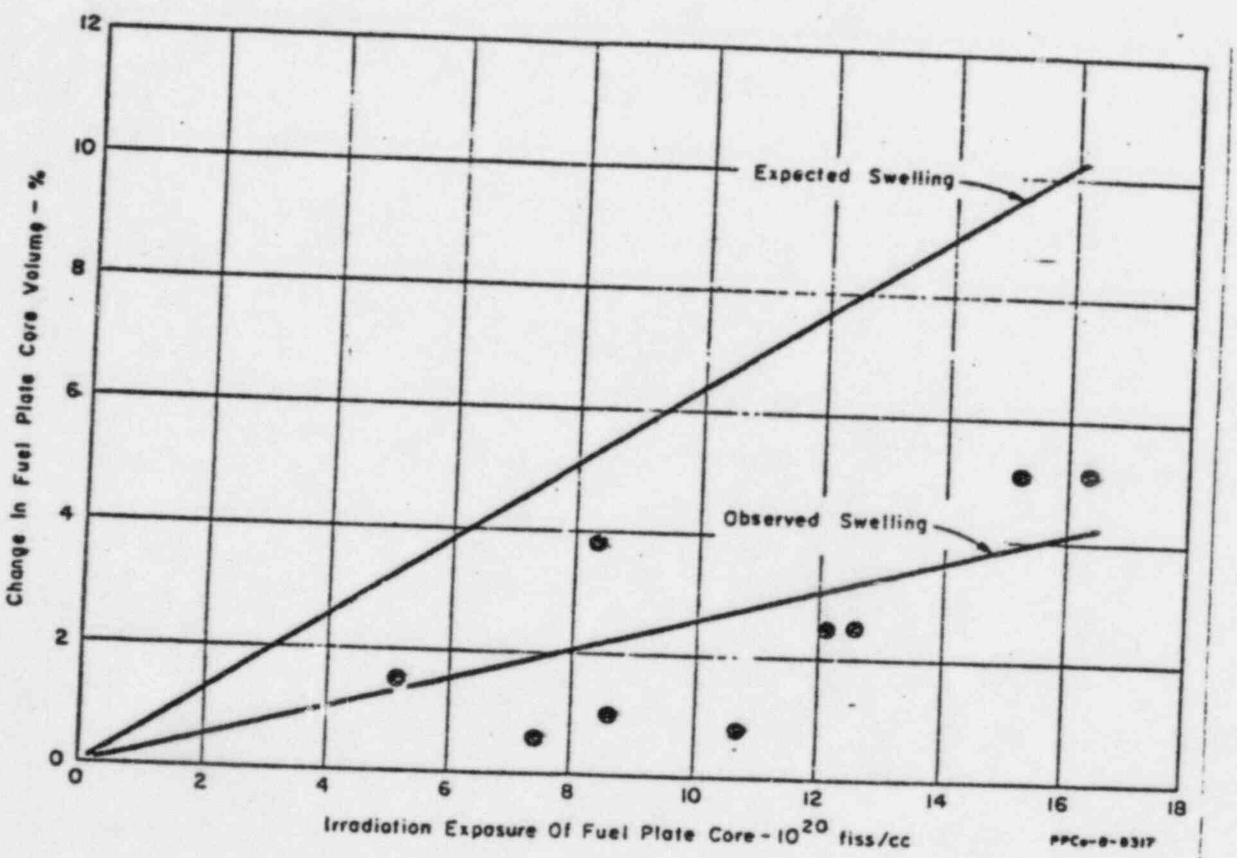


FIGURE 4

SWELLING PRODUCED IN POWDERED UAl_3 FUEL PLATE CORES BY IRRADIATION IN G-12 PRESSURIZED WATER LOOP OF ETR[1]

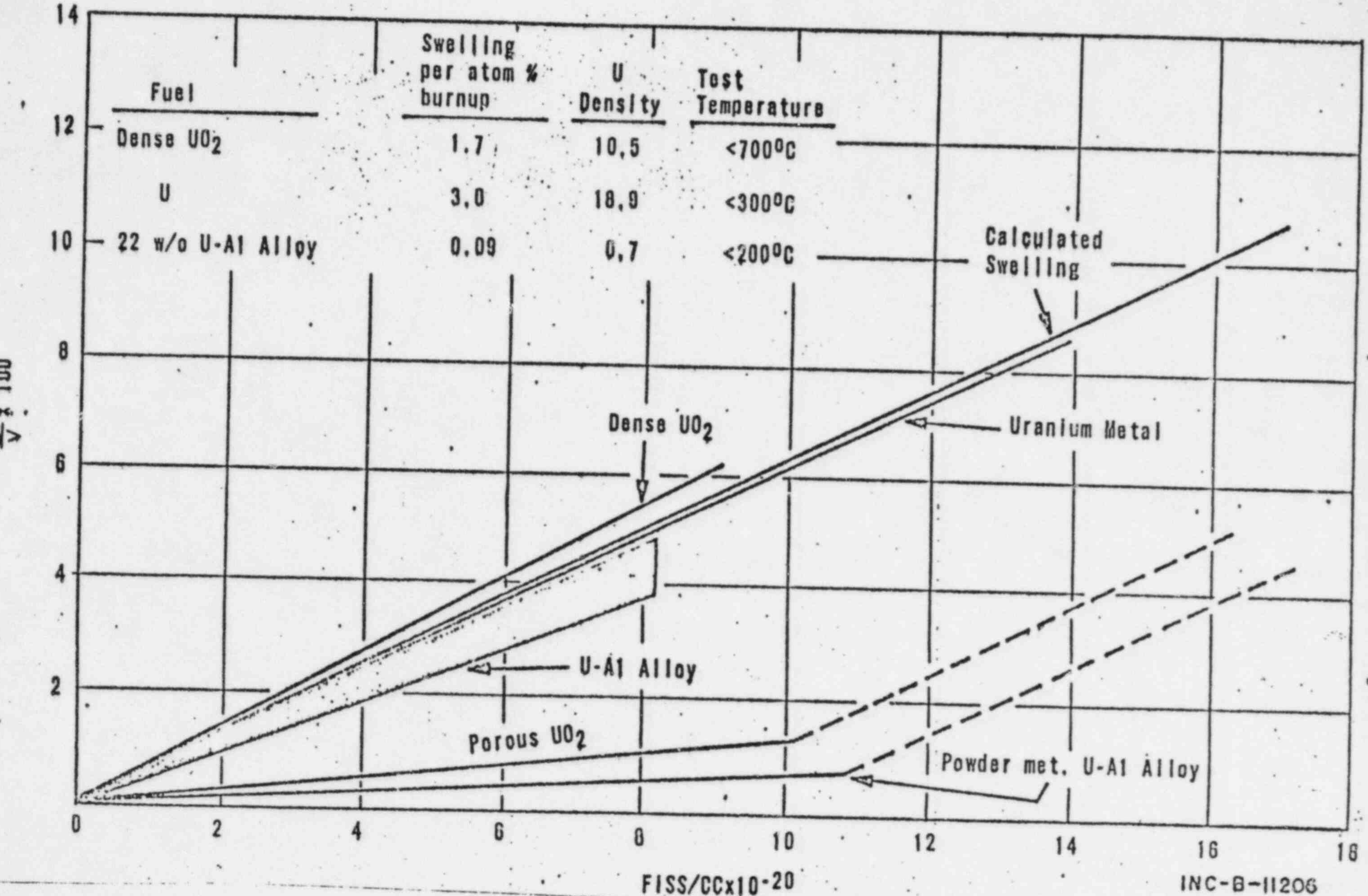


FIGURE 5

FUEL SWELLING INDUCED BY SOLID FISSION PRODUCTS[5]

TABLE 6

SWELLING DATA ON UAl_x FUEL ELEMENT XA003F IRRADIATED IN ATR[16]

Fuel Element Plate	Coupon Number	Distance From Top of Plate (in)	Fission Density (10^{20} fiss/cc)	Core Swelling (% ΔV/V)
1	1	3	2.5	1.6
1	2	14	7.1	2.0
1	3	25	9.1	3.0
3	1	3	3.0	1.9
3	2	14	8.4	5.5
3	3	25	11.0	9.0
5	1	3	3.4	2.2
5	2	14	9.8	3.0

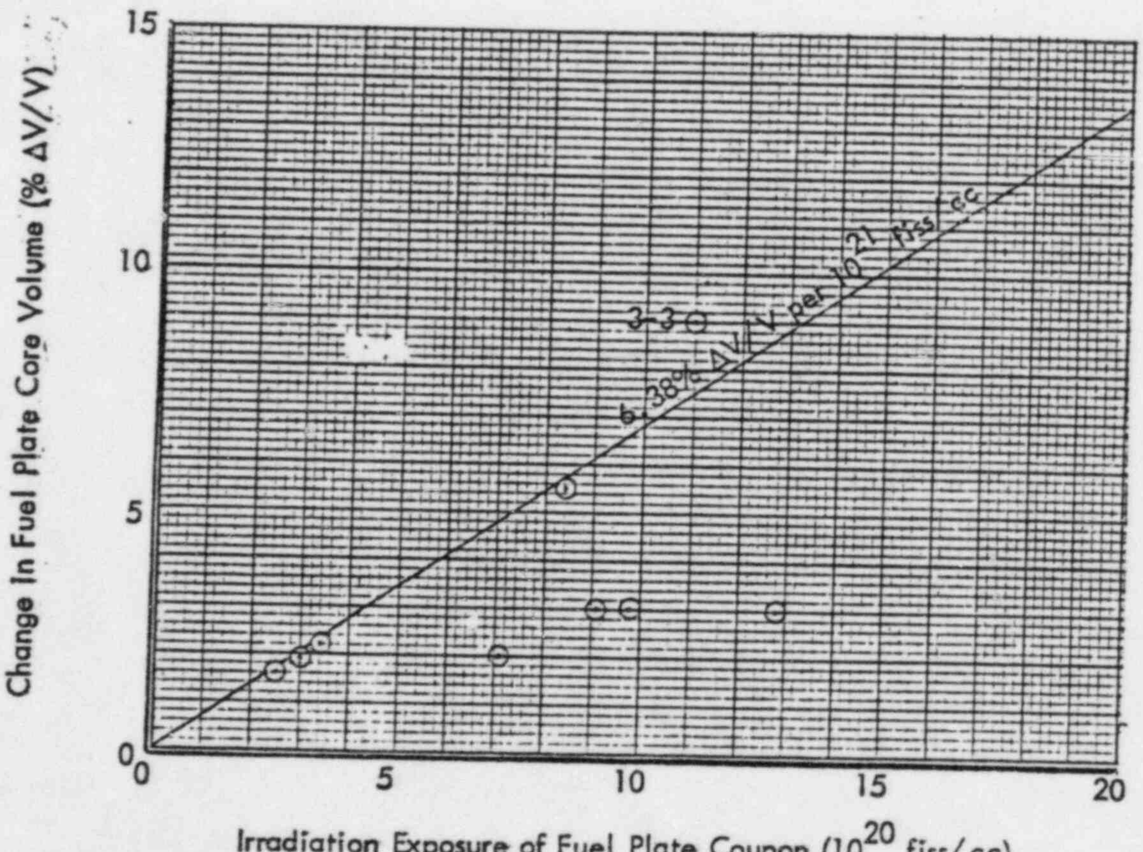


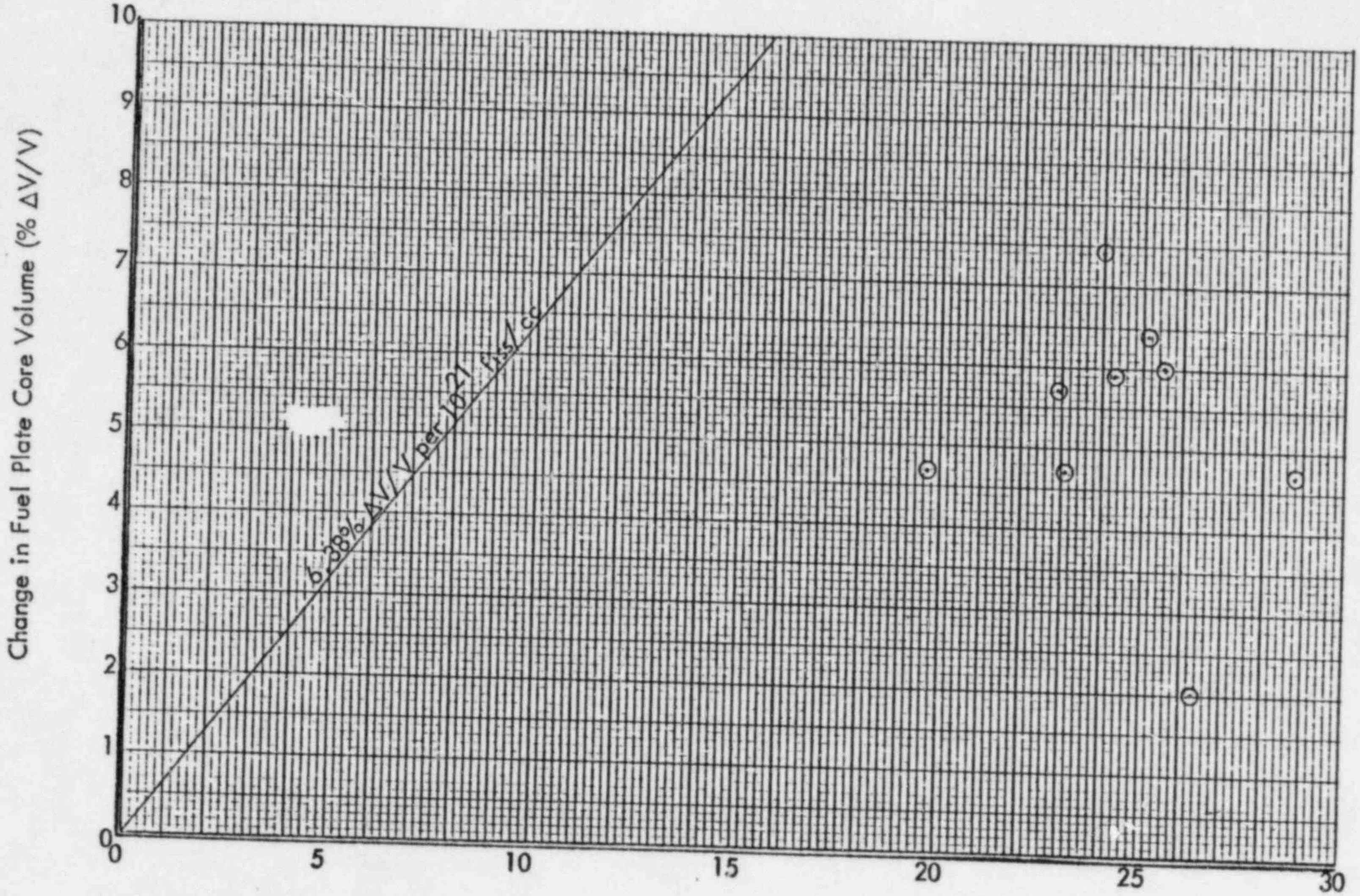
FIGURE 6

SWELLING PRODUCED IN UAl_x FUEL ELEMENT XA003F IRRADIATED IN ATR[16]

TABLE 7

DATA ON UAl_x ATR FUEL PLATE SPECIMENS IRRADIATED IN ETR[17]

Plate Number	U-235 Per cc of Core (gm)	Uranium Weight in Core (gm)	Enrichment (%)	Plate Surface Temperature (°C)	Percent Burnup (%)	Fission Density (10^{20} fiss/cc)	Core Swelling (% ΔV/V)	Blister Temperature (°C)
169-4	1.51	.7468	91.86	100 - 260	82.6	26.3	2.0	565
169-5	1.51	.7286	91.86	100 - 260	90.9	28.8	4.7	>565
169-11	1.21	.6715	91.86	90 - 340	90.5	23.1	4.7	538
169-12	1.22	.6747	91.86	90 - 340	95.2	24.3	5.9	565
169-19	.953	.5424	91.86	90 - 270	98.3	19.7	4.7	565
169-36	1.21	.6554	93.10	90 - 340	98.3	25.1	6.4	>565
168-37	1.21	.6772	93.10	90 - 340	99.9	25.5	6.0	538
169-38	1.23	.6967	93.10	90 - 340	97.0	25.0	7.4	538
169-39	1.22	.6651	93.10	90 - 340	93.3	23.9	5.7	565



Irradiation Exposure of Fuel Plate Core (10^{20} fiss/cc)

FIGURE 7

SWELLING PRODUCED IN UAl_x ATR FUEL PLATE SPECIMENS IRRADIATED IN ETR[17]

TABLE 8

PERTINENT CORE ATTRIBUTES OF FUEL PLATES IRRADIATED IN HFIR IN PM CAPSULE 1[2]

Plate	Dispersoid				Void Content of Dispersion (vol%)	Uranium Loading	
	Powder Blend	Type	Concentration			Total (g)	Fissile Density (g U-235/cm^3 of core)
			(wt%)	(vol%)			
11-3	PB-33	Arc-cast UAl_x	53.2	31.7	3.1	1.088	0.00
21-4	PB-33	Arc-cast UAl_x	54.0	31.9	4.5	1.089	0.00
31-1	PB-33	Arc-cast UAl_x	54.8	32.2	5.8	1.088	0.00
12-3	PB-01	High-fired U_3O_8	47.1	21.7	4.1	1.187	1.41
22-4	PB-04	Burned U_3O_8	49.8	23.8	8.2	1.187	1.43
32-4	PB-32	Arc-cast UAl_x	51.4	30.3	1.3	1.085	1.31
13-4	PB-01	High-fired U_3O_8	40.1	17.5	3.0	0.953	1.14
23-1	PB-04	Burned U_3O_8	42.0	19.1	6.4	0.954	1.15
33-4	PB-35	Arc-cast UAl_x	53.7	31.6	4.6	1.087	0.00
14-3	PB-32	Arc-cast UAl_x	52.9	30.6	4.2	1.086	1.32
24-3	PB-32	Arc-cast UAl_x	52.2	30.3	3.3	1.085	1.31
34-2	PB-36	Arc-cast UAl_x	53.2	30.9	4.3	1.089	1.32
15-4	PB-32	Arc-cast UAl_x	62.8	38.6	6.6	1.344	1.67
25-4	PB-32	Arc-cast UAl_x	64.1	37.3	8.4	1.345	1.69
35-4	PB-36	Arc-cast UAl_x	63.0	38.7	7.3	1.342	1.66
16-2	PB-11	High-fired U_3O_8	39.9	17.3	2.8	0.954	0.00
26-3	PB-34	High-fired U_3O_8	39.9	17.4	2.3	0.954	0.00

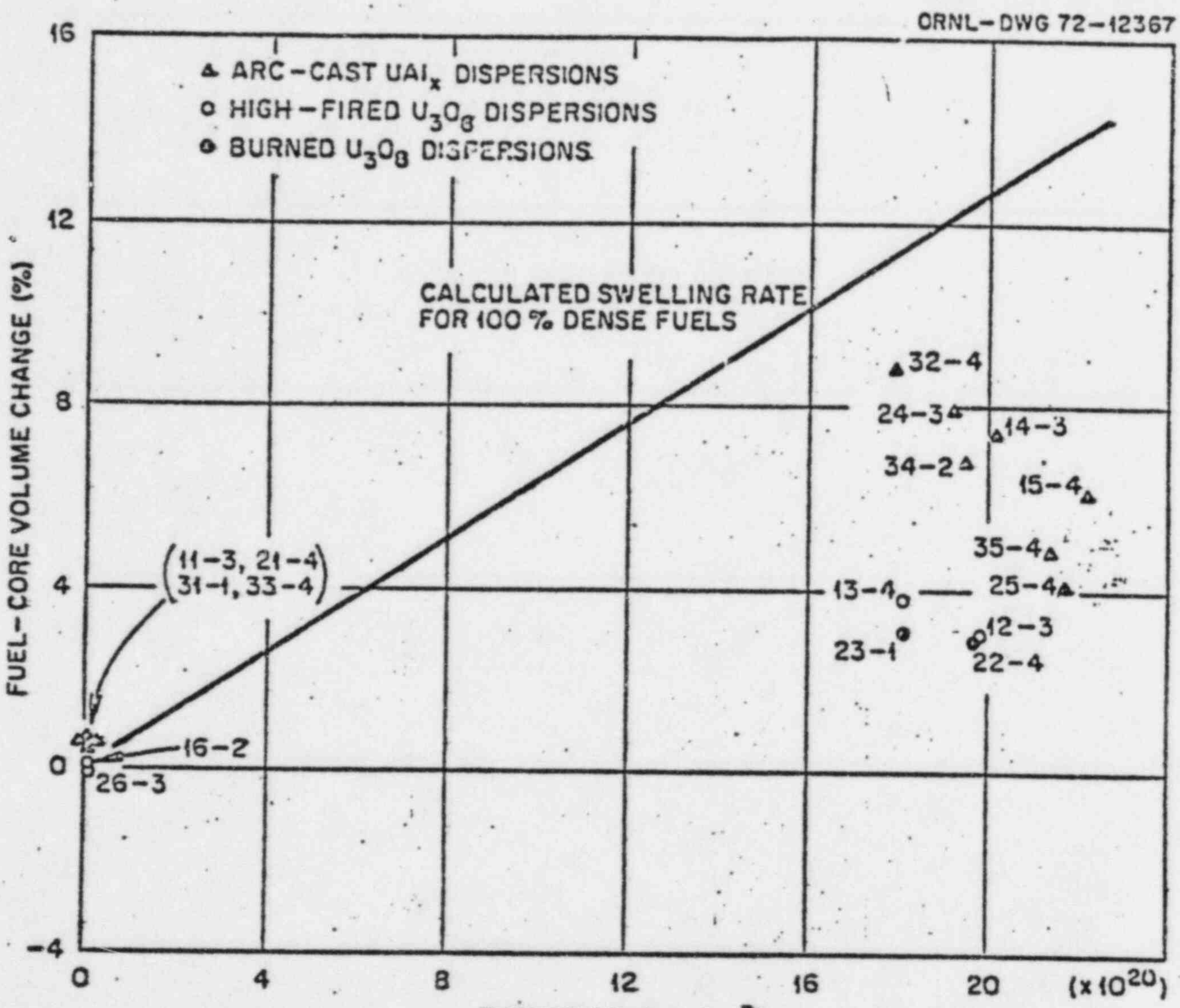


FIGURE 8

SUMMARY OF FUEL-CORE SWELLING DATA FOR MINIATURE FUEL PLATES IRRADIATED IN THE ORNL PM-1 EXPERIMENT. DATA POINTS ARE IDENTIFIED BY THE FUEL PLATE IDENTIFICATION NUMBER[2]

All 17 test plates were inspected visually in the HFIR pool approximately 24 hours after shutdown of the experiment. The plates were free of warpage and distortion and generally appeared to be in excellent condition.

The test plates were allowed to cool for approximately three weeks in the pool and then transferred to the High Radiation Level Examination Laboratory (HRLEL) hot cells for detailed examination and evaluation. Examination consisted primarily of visual inspection of the plate surfaces for indications of structural damage or defects, determination of the plate weights and densities both before and after chemical removal of corrosion-product films, analytical determination of the speciment burnups, and metallographic examination.

In general, the examinations showed all 17 test plates to be in excellent condition. The density and weight-loss data indicated that the plates had increased less than 0.002 inches in thickness, including the oxide film formed on the plate surfaces during irradiation, and that corrosion had reduced the thickness of the cladding of these plates by less than 0.0002 inches. Extensive visual and metallographic examination of sections from each plate revealed no indications of actual or incipient structural failure of any of the test plates. Consequently, it was concluded that all 17 plates performed quite satisfactorily under the imposed conditions of burnup levels of $1.8 - 2.2 \times 10^{21}$ fissions/ cm^3 at irradiation temperatures of 60 to 98° C. Slight but significant differences in the swelling of the various fuel dispersions were observed. As shown in Figure 8, plates containing aluminide dispersions consistently swelled more than plates containing the U_3O_8 dispersions when irradiated to comparable burnup levels.

5.3 German Fuel Swelling Tests

In 1976, UAl_3 and UAl_2 fuel plates as listed in Table 9 were irradiated in the Karlsruhe FR2 reactor.[15] Maximum burnups attained were 53 to 72% which correspond to fission densities of 1.8×10^{21} to 2.7×10^{21} fissions/ cc.

Fission product swelling occurred at a lower rate than the expected 6.38% $\Delta V/V$ per 10^{21} fissions/ cc. Measured values in Figure 9 show that, up to approximately 40% burnup which corresponds to a fission density of approximately 1.5×10^{21} fissions/ cc, the swelling rate is approximately 3% $\Delta V/V$ per 10^{21} fissions/ cc. Above fission densities of 1.5×10^{21} fissions/ cc, the rate of volume increase was considerably greater. The swelling rate in UAl_2 plates with identical uranium content was somewhat higher than that of UAl_3; the difference is not much greater than the largest scatter in the measured values.

TABLE 9

URANIUMALUMINIDE-CONTENT AND IRRADIATION CONDITIONS FOR THE FUEL PLATES

Test Rig No.	UAl_x Content of the Dispersion (wt%)	Mean Surface Temperature (°C)	Mean U-Burnup (%)	No. of plates Irradiated
1	50 UAl_3	70	16	4
2	50 UAl_3	70	47	4
3	50 UAl_3	70	34	4
4	50 UAl_3	70	26	1
	45.5 UAl_2	70	26	3
5	45.5 UAl_2	70	21	4
6	50 UAl_3	70	44	2
	45.5 UAl_2	70	44	2
7	50 UAl_3	70	6.5	3
	45.5 UAl_2	70	6.5	1
8	50 UAl_3	120/135/150	28	4/4/4
9	50 UAl_3	150/165/180	53	2/2/2
	45.5 UAl_2	150/165/180	53	2/2/2
10	50 UAl_3	150/165/180	72	2/2/2
	45.5 UAl_2	150/165/180	72	2/2/2
11	50 UAl_3	150/165/180	60	1/1/2
	45.5 UAl_2	150/165/180	60	1/1/2
	54.5 UAl_2	150/165	60	2/2

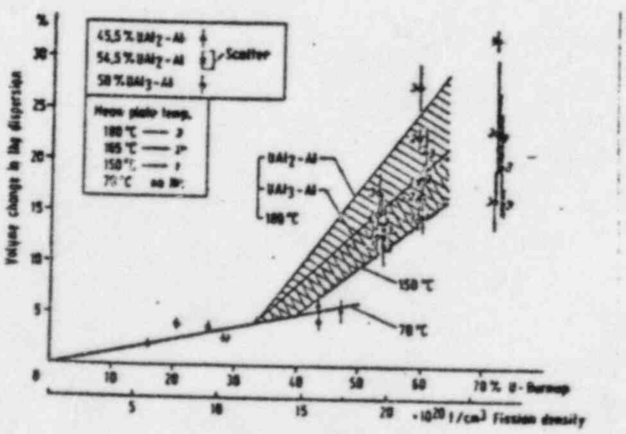


FIGURE 9

SWELLING OF THE FUEL PLATES (RELATIVE TO THE FUEL DISPERSION VOLUME) UNDER IRRADIATION AS A FUNCTION OF BURNUP AT VARIOUS IRRADIATION TEMPERATURES[15]

6. BLISTER FAILURE

6.1 NRTS Blister Failure Tests

Irradiated aluminide and oxide test specimens at NRTS were heated in air for 30-minute periods starting at a temperature of 260° C.[1] After each heating period, specimen plates were inspected for blistering and, if not blistered, heated to successively higher temperatures. The results are shown in Figure 10.

Figure 11 presents blister anneal data for aluminide fuel.[11] For a given peak fission density, the curve relates the minimum temperature at which fuel plate blistering may be expected.

Post-irradiation blister failure test results for experiment INC-16-1 are tabulated in Table 12 and plotted in Figure 12.[16] Annealing temperature was raised in 50° F increments during the test. In general, blister temperatures are within 50° F of the failure - no-failure curve established for UAl_x fuel plates. Matrix-cracking was the predominant blister mechanism, though some microcracks were observed in the fuel.

All fuel plate specimens in the INC-16-2 experiment were blister tested following irradiation. The results are tabulated in Table 7 and plotted in Figure 13. A previously established UAl_x blister curve is also shown in Figure 13. The INC-16-2 specimens blistered well above the blister temperature of 465° C which had previously been established to be constant beyond a fission density of 1.5×10^{21} fissions/cc.[16]

6.2 German Fuel Plate Temperature Failure Tests

The test plates listed in Table 9 were annealed for one hour from 200 - 500° C in steps of 100° C following irradiation. Volume changes were measured after each step.

Up to 300° C no volume changes were evident. At 400° C, the first volume changes occurred. These took place only in the sample plates which were irradiated at the lowest mean surface temperature of 70° C. Plates irradiated at higher temperatures showed pronounced swelling at 500° C. The sample plates irradiated at 70° C blistered at 500° C annealing temperature. In all other cases, no blisters were visible.

7. HEAT TRANSFER CHARACTERISTICS

Rev 4/78 In Appendix A, the peak fuel plate temperature during reactor operation is calculated to be 156° F or 69° C. Peak meat center temperature is approximately the same value because the temperature rises in the meat and clad are less than 1° F.

The thermal resistance, K, for the powder metallurgy fuel meat is approximately

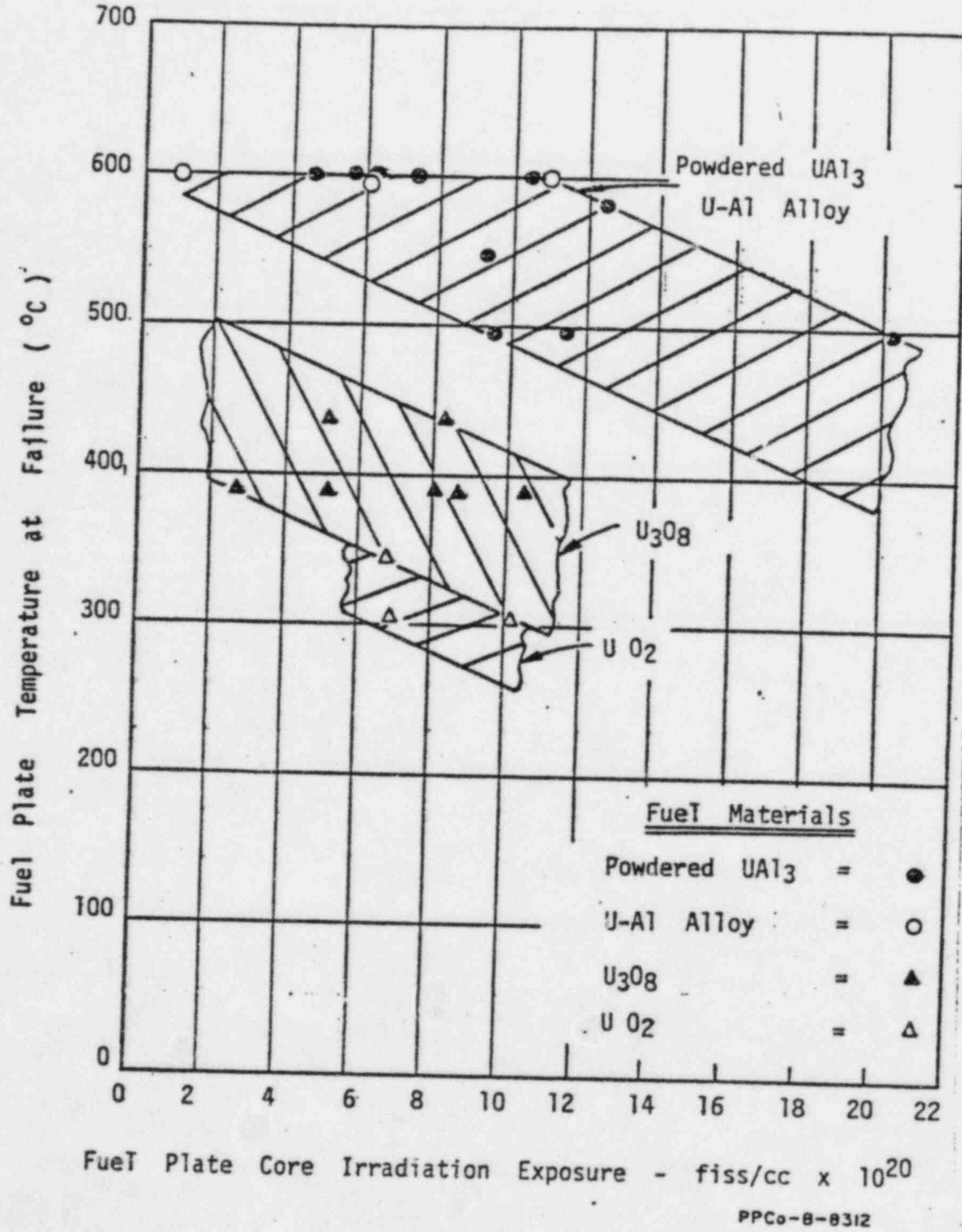


FIGURE 10

RESULTS OF POST-IRRADIATION ANNEAL OF SAMPLE FUEL PLATES IRRADIATED IN L-51 POSITION OF THE MTR CORE

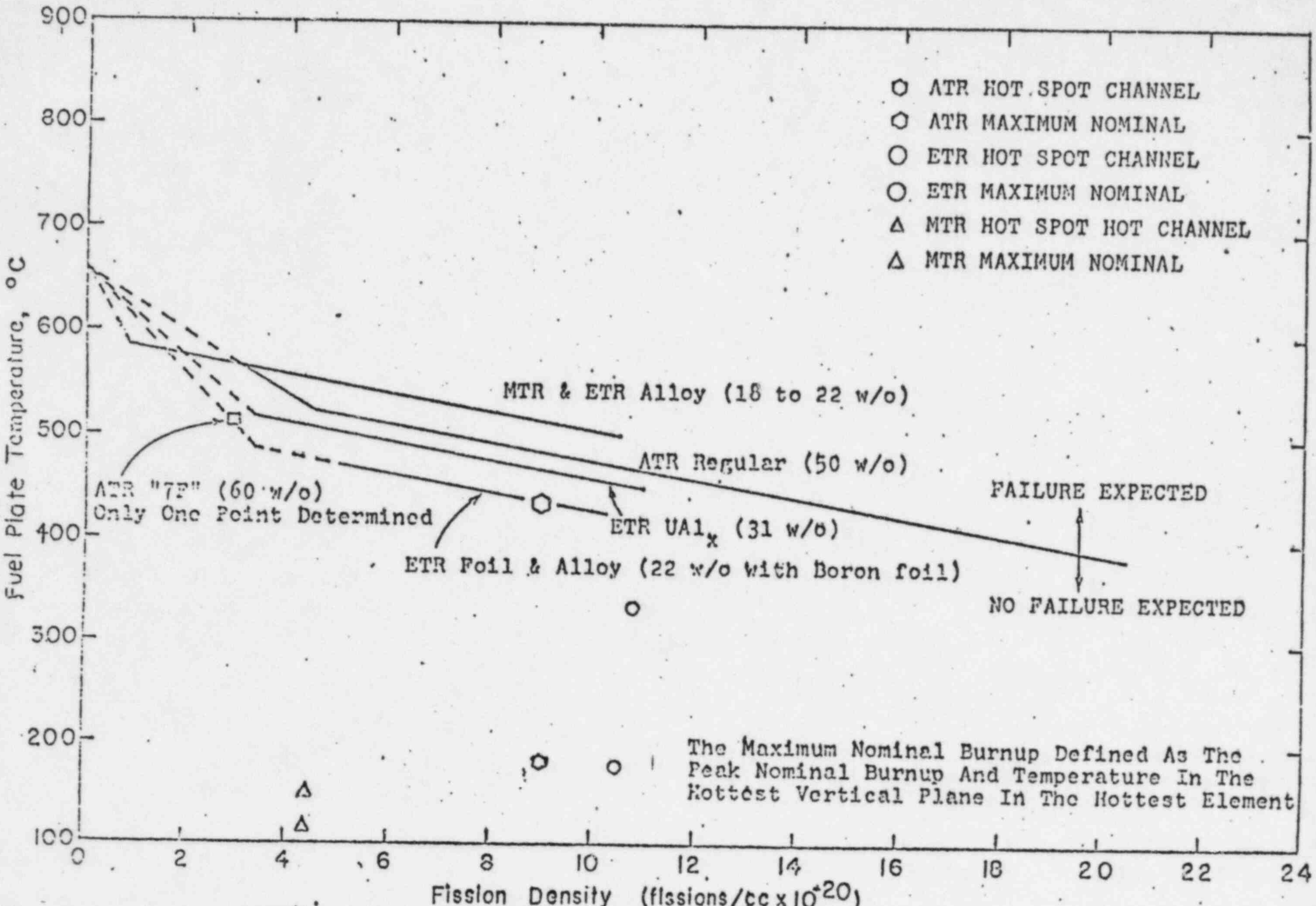


FIGURE 11

BLISTER ANNEAL DATA FOR ALUMINIDE (UAl_x) FUEL[11]

TABLE 12

BLISTER DATA FOR UAl_x FUEL ELEMENT XA003F

Plate	Coupon Number	Estimated Peak Exposure In Blistered Sample (10^{20} fiss/cc)	Blister Temperature (°F)
8	1	16	>1000
8	2	13	>1000
8	3	9	>1000
15	1	18	850
15	2	15	900
15	3	10	1000
17	1	17	950
17	2	14	1000
17	3	9	950
18	1	14	850
18	2	12	900
18	3	8	950
19	1	15	900
19	2	13	900
19	3	8.6	950

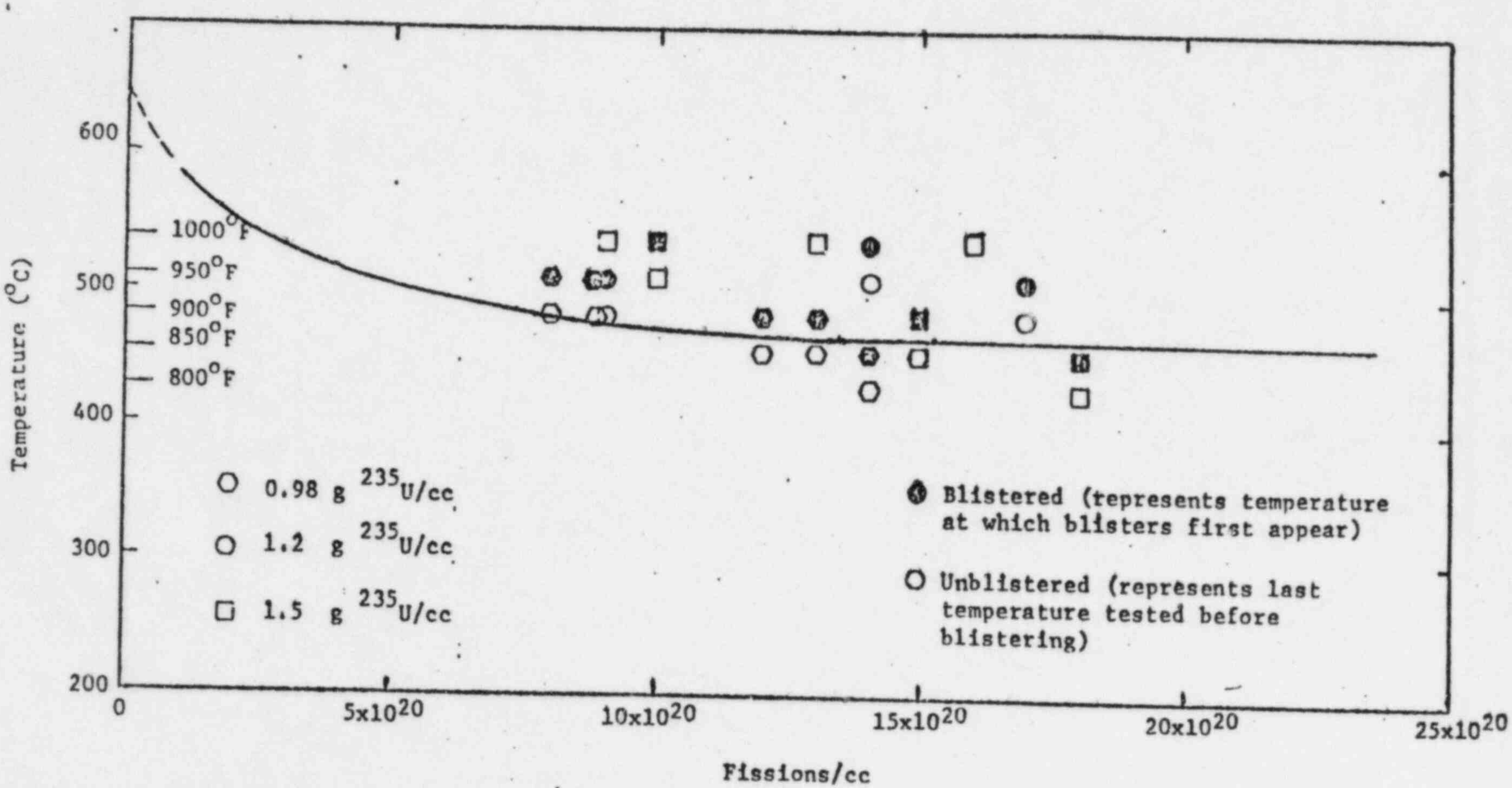


FIGURE 12

BLISTER TEMPERATURES FOR UAl_x FUEL ELEMENT XA003F PLATES SUPERIMPOSED ON FAILURE - NO-FAILURE CURVE ESTABLISHED FOR UAl_x SAMPLE FUEL PLATES

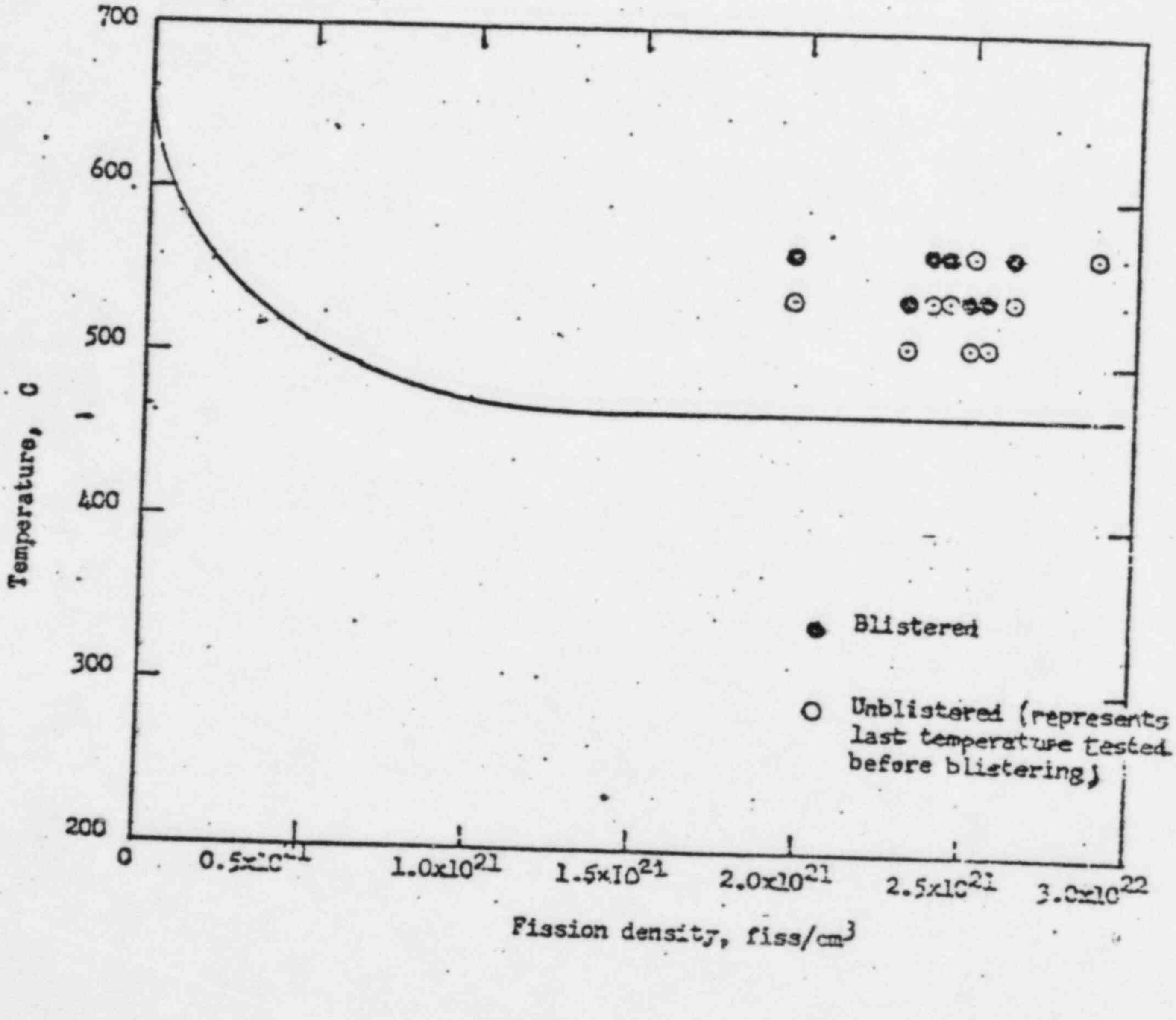


FIGURE 13

POST-IRRADIATION BLISTER FAILURE PLOT FOR UAl_x ATR FUEL PLATE SPECIMENS IRRADIATED IN ATR[17]

90% of alloy thermal resistance.[10] The conversion to aluminide or oxide meat will result in approximately a 10% rise in the meat differential temperature. However, since the total meat differential temperature is less than 1° F, as
Rev 4/78 compared to a 36° F clad – coolant interface differential temperature, the effect is inconsequential.

Oxide coatings of up to two mils build up on fuel clad during fuel use.[1, 15] A temperature rise of approximately 16° F across the oxide coating could result
Rev 4/78 in increased maximum fuel plate temperatures of approximately 172° F.[4]

Rev 4/78 The FNR peak fuel clad temperatures of 156° F and 172° F with oxide coating are well below failure temperatures for aluminide and oxide fuels for all fuel fission densities for which data was taken.

8. FISSION DENSITY

Fission density limits are based upon acceptable fuel swelling limits and failure due to fuel blistering.

Table 13 provides a summary of fuel plate swelling and blister failure data from Figures 2 – 13. The swelling data provide the maximum fuel plate core swelling observed up to the maximum fission density listed for the test data reviewed. Similarly, the blister data provide minimum blister failure temperatures observed up to the maximum fission densities listed.

Rev 4/78 For peak FNR clad operating temperatures of 156 – 172° F, no fuel blister failures were experimentally observed.

The maximum fuel plate swelling observed, 12.7 % $\Delta V/V$, reduces the FNR element coolant flow channel by 2.2%, from 0.117 inches to .114 inches,
Rev 4/78 a reduction that is operationally acceptable.

FNR fission density limits should be based upon the data in this analysis and the peak values routinely achieved by operating reactors. The following fission density limits in fissions/cc are requested for the FNR.

	Fuel Material	Table 13 Test Data	Operating Reactors	Proposed FNR Limit
Rev 4/78	Alloy (U-Al)	2.0×10^{21}	2.0×10^{21} (GETR)	1.8×10^{21}
Rev 4/78	Aluminide (UAl_4, UAl_3, UAl_2)	2.88×10^{21}	1.8×10^{21} (ATR)	1.8×10^{21}
Rev 4/78	Oxide (U_3O_8)	2.0×10^{21}	1.9×10^{21} (HFIR)	1.8×10^{21}

TABLE 13

SUMMARY OF FUEL PLATE SWELLING AND BLISTER FAILURE DATA

Fuel Plate Swelling Data

Fuel Material	Figure	Maximum Core Swelling Observed (% $\Delta V/V$)	Maximum Fission Density Attained (10^{20} fiss/cc)
U-Al Alloy	1	12.7	20.0
U-Al Alloy	3A	5.3	8.5
UAl_x	2	7.5	14.0
UAl_x	3A	4.5	8.0
UAl_x	3B	2.5	11.0
UAl_x	4	5.0	16.5
UAl_x	6	9.0	12.8
UAl_x	7	6.4	28.8
UAl_x	8	9.0	22.0
UAl_x	9	5	17.5 (70° C)
U_3O_8	8	4.0	20.0

Fuel Plate Blister Failure Temperature Data

Fuel Material	Figure	Minimum Blister Failure Temperature (°C/°F)	Maximum Fission Density Attained (10^{20} fiss/cc)
U-Al Alloy	10	600/1112	11
UAl_x	10	500/932	20
UAl_x	12	450/842	18
UAl_x	13	550/1022	29
U_3O_8	10	380/716	10.5
UO_2	10	300/572	10.5

APPENDIX A

PEAK FUEL CLAD TEMPERATURE CALCULATIONS FOR COMPARING MEAT AND CLAD DIFFERENTIAL TEMPERATURES WITH THE CLAD - COOLANT INTERFACE DIFFERENTIAL TEMPERATURE

A.1 FUEL PLATE PARAMETERS

A.1.1 Surface Area

A 25-element core consists of 21 standard, 18-plate elements and four control rod, 9-plate elements for a total of 414 fuel plates.

$$A_s = 414 \ 2(w \times l) \tag{A1}$$

where w = Plate width, .21 ft
l = Plate length, 2 ft

$$A_s = 345 \text{ ft}^2$$

A.1.2 Fuel Meat Volume

$$V_m = 414 \ t_m \times w \times l \tag{A2}$$

where t_m = Meat thickness, .00167 ft

$$V_m = .290 \text{ ft}^3 \tag{A3}$$

$= 501 \text{ in}^3$

$= 8212$ cc/core

Rev 10/77 $=$ 358 cc/18-plate standard element

Rev 10/77 $=$ 179 cc/9-plate special element

A.1.3 Core Volume

A 25-element core is configured in a 5 X 5 array of elements. Each element measures 3 in X 3 in X 24 in.

$$V_c = 3.125 \text{ ft}^3 \tag{A4}$$

$= 5400 \text{ in}^3$

$$= 88,490 \text{ cc}$$

$$= 88.49 \text{ l}$$

A.1.4 Core Flow Area

$$A_c = N_c \left[w_c \times t_c \right] \tag{A5}$$

where N_c = 414 coolant channels
w_c = .218 ft channel width
t_c = .0098 ft channel thickness

$$A_c = 414 \left[.218 \times .0098 \right]$$

$$= .884 \text{ ft}^2$$

A.2 REACTOR THERMAL POWER DENSITY

$$TPD = 2 \text{ MW}/V_c \tag{A6}$$

$$= .0226 \text{ MW/l}$$

A.3 REACTOR FISSION DENSITY

Each standard fuel element contains 140 gm U-235 and is 17% burned up at end of life. Standard fuel element burnup in gm is

$$BU = .17 \times 140 \tag{A7}$$

$$= 23.8 \text{ gm}$$

The average fission density in a standard element is

$$FD_{avg} = \frac{BU}{V_m} \times \frac{A_o}{N_{235}} \times \frac{\sigma_f}{\sigma_{f+c}} \tag{A8}$$

where V_m = 354 cc/standard element
A_o = 6.02 X 10^{23} atoms U-235/gm-mole
N_{235} = 235 gm U-235/gm-mole
σ_f/σ_{f+c} = 0.85

The term, σ_f/σ_{f+c}, accounts for the fact that not all U-235 atoms which absorb neutrons are fissioned.

$$FD_{avg} = \frac{23.8}{354} \times \frac{6.02 \times 10^{23}}{235} \times 0.85$$

$$= 1.46 \times 10^{20} \text{ fissions/cc}$$

Rev 4/78 Peak to average flux ratio in the core is approximately 1.86.

$$FD_{pk} = \underline{1.86} FD_{avg} \tag{A9}$$

$$= \underline{2.72} \times 10^{20} \text{ fissions/cc}$$

Control rod fuel elements are 35% depleted at end of life. Similar fission density calculations yield

Rev 4/78

$$FD_{pk} = \underline{5.44} \times 10^{20} \text{ fissions/cc}$$

A.4 FISSILE MATERIAL DENSITY

Fissile material density in fuel plates is the plate loading divided by the fuel volume.

$$FMD = \frac{W_{235}}{V_m} \tag{A10}$$

$$= \frac{140 \text{ gm}}{354 \text{ cc}}$$

$$= .395 \text{ gm/cc}$$

A.5 REACTOR HEAT FLUX

$$Q_{avg} = \frac{MW}{A_s} \tag{A11}$$

where $MW = 2$ MW
$= 2 \times 10^6$ watts
$= 6.82 \times 10^6$ BTU/hr

$$Q_{avg} = \frac{6.82 \times 10^6}{345}$$

$$= 1.98 \times 10^4 \text{ BTU/hr-ft}^2$$

Peak heat flux corresponds to peak neutron flux.

Rev 4/78

$$Q_{pk} = \underline{1.86}\, Q_{avg} \tag{A12}$$

$$= \underline{3.68 \times 10^4} \text{ BTU/hr-ft}^2$$

A.6 COOLANT VELOCITY

$$v = \frac{F}{A_c} \tag{A13}$$

where F = 980 gpm coolant flow rate
= 2.19 ft^3/sec

$$v = \frac{2.19}{.884}$$

$$= 2.48 \text{ ft/sec}$$

$$= 8919 \text{ ft/hr}$$

A.7 REYNOLD'S NUMBER

The average coolant temperature in the core is assumed to be 120° F. The average temperature near the clad - coolant interface is assumed to be 135° F, the mean between 120° F and an estimated 150° F clad temperature.[12] 135° F is the temperature used to evaluate fluid parameters.

$$Re = \frac{v\rho l}{\mu} \tag{A14}$$

where v = 8919 ft/hr coolant velocity
ρ = 61.5 lb/ft^3 coolant density
μ = 1.19 lb/ft-hr coolant viscosity
l = Characteristic channel dimension, $2t_c$ = 0.0196 ft

l was chosen as .0196 feet based upon flow analysis between parallel plates.[14] The characteristic channel dimension is the hydraulic diameter of the channel which is equal to four times the cross sectional area divided by the wetted perimeter.

$$Re = \frac{8919 \times 61.5 \times .0196}{1.19}$$

Rev 10/77

$$= \underline{9.05 \times 10^3}$$

For a smooth channel of assumed relative roughness, $\epsilon / D = 5 \times 10^{-5}$, the Reynold's Number describes flow in the transition region from laminar to turbulent flow on the Moody Friction Chart, Figure A1.

A.8 PRANDTL NUMBER

$$Pr = \frac{\mu c_p}{K} \tag{A15}$$

where c_p = .998 BTU/lb-°F coolant specific heat
K = .377 BTU/hr-ft-°F coolant thermal resistance

$$Pr = \frac{1.19 \times .998}{.377}$$

$$= 3.15$$

A.9 NUSSELT NUMBER

The Dittus - Bolter equation is used to evaluate Nusselt Number.

$$Nu = 0.023(Re)^{0.8}(Pr)^{0.4} \tag{A16}$$

Rev 10/77

$$= \underline{53}$$

A.10 HEAT TRANSFER COEFFICIENT

The heat transfer coefficient for the film at the clad - coolant interface is

$$h = \frac{KNu}{T} \tag{A17}$$

$$= \frac{.377 \times 53}{.0196}$$

Rev 10/77

$$= \underline{1022} \text{ BTU/hr-ft}^2\text{-}^\circ\text{F}$$

A.11 THERMAL RESISTANCE

A model of a coolant channel is shown in Figure A2. The value of thermal resistivity, K_c, for the clad is the value for 1100F aluminum. The value for K_m is an experimentally determined value for oxide cermets similar to the powder metallurgy aluminides and oxides that will be used in the new FNR fuel.[10]

A.12 MAXIMUM FUEL PLATE CLAD TEMPERATURE

From Figure A2, the differential temperature between the meat centerline, t_m, and bulk coolant, t_w, is evaluated by

$$Q_{pk} = \frac{t_m - t_w}{R_m + R_c + R_F} \tag{A18}$$

since $R_F >> R_c$ and R_m.

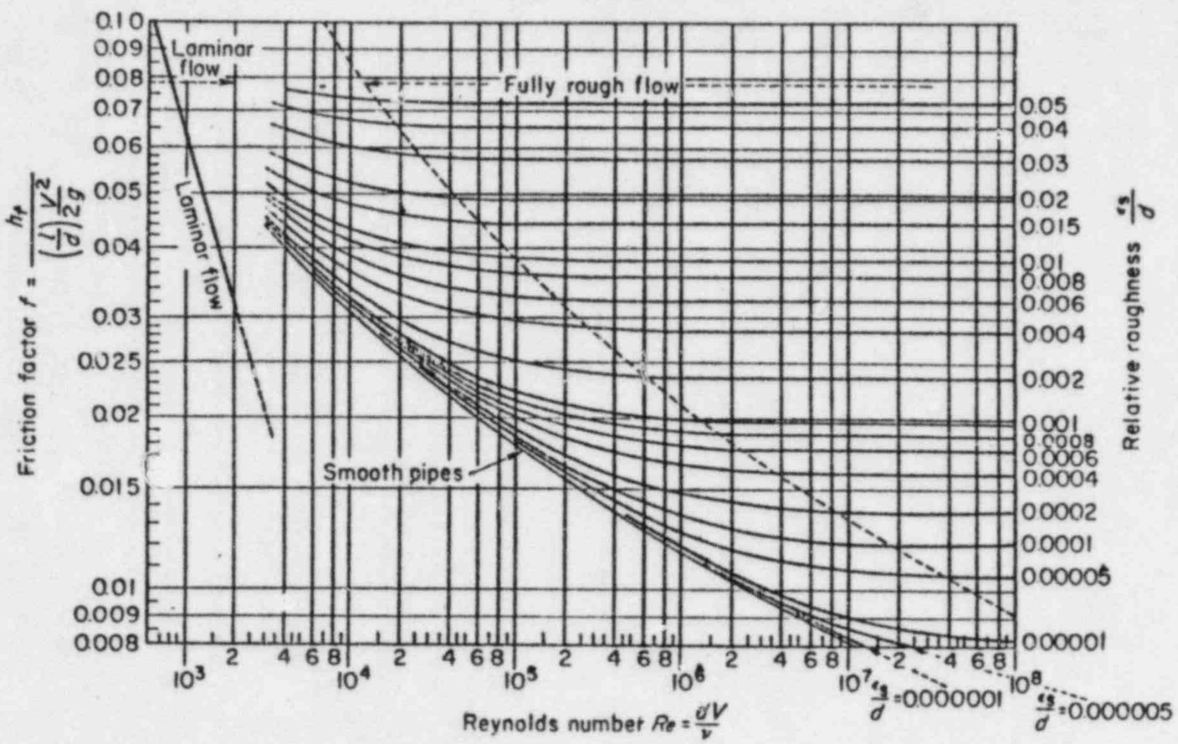
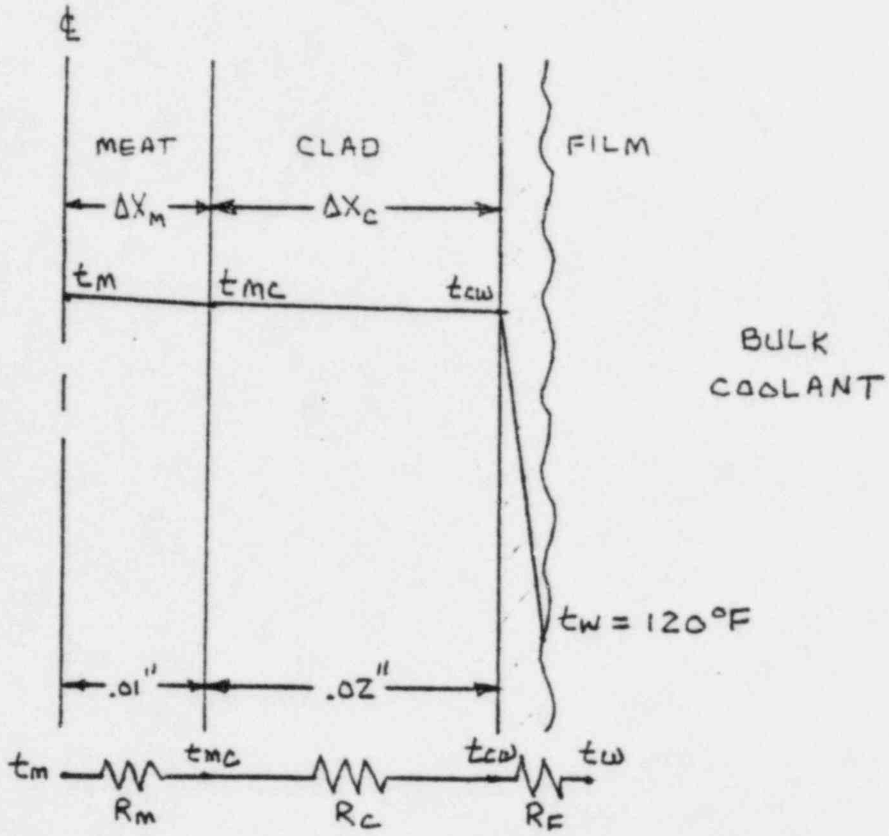


FIGURE 1A

FRICTION COEFFICIENT AS A FUNCTION OF REYNOLDS NUMBER FOR ROUND PIPES OF VARIOUS RELATIVE ROUGHNESS RATIOS ε/D



$$R_m = \frac{\Delta x_m}{K_m} = \frac{.00083}{103} = 8 \times 10^{-6} \text{ hr-ft}^2\text{-}^\circ \text{ F/BTU}$$

$$R_c = \frac{\Delta x_c}{K_c} = \frac{.00167}{115} = 1.4 \times 10^{-5} \text{ hr-ft}^2\text{-}^\circ \text{ F/BTU}$$

Rev 10/77

$$R_F = 1/h = 1/1022 = \underline{9.8 \times 10^{-4}} \quad \text{hr-ft}^2\text{-}^\circ \text{ F/BTU}$$

FIGURE A2

FNR FUEL THERMAL RESISTANCE MODEL

$$Q_{pk} = \frac{t_{cw} - t_w}{R_F} \qquad \text{(A19)}$$

and

$$t_{cw} = t_m$$

which means the temperature rises in the clad and meat are negligible compared to the rise across to clad - coolant interface. The fuel element plates are essentially at a uniform radial temperature, t_{cw}. From equation (A19),

$$t_{cw} = t_w + Q_{pk} R_F$$

Rev 4/78

$$= 120 + \underline{3.68 \times 10^4} \times \underline{9.8 \times 10^{-4}} \ {}^{o}F$$

$$= 120 + \underline{36}$$

$$= \underline{156} \ {}^{o}F$$

REFERENCES

1. G. W. Gibson, "The Development of Powdered Uranium Aluminide Compounds for Use as Nuclear Reactor Fuels", IN-1133, TID-4500, December, 1967.

2. M. M. Martin, A. E. Richt, and W. R. Martin, "Irradiation Behavior of Aluminum Base Fuel Dispersions", ORNL-4856, May, 1973.

3. "Reactor Safety Evaluation of ORNL Proposal to Modify Fuel in ORR", ORNL, February, 1977.

4. P. Tichler, "Review of Proposed Increase in Fuel Element Loading and Fuel Burnup", memorandum, Brookhaven National Laboratory, February, 1977.

5. W. C. Francis, "Fuel Elements for Thermal Test Reactors - Performance at NRTS", Paper for presentation at the AEC - Industry Meeting, Water Reactor Fuel Element Technology, U. S. AEC Headquarters, Germantown, Maryland, January 29-30, 1968.

6. C. Julian, "Evaluation of a 6.2 Kilogram U-235 Core Loading for the Missouri University Research Reactor", Docket 50-186, Technical Specification Request for Change, University of Missouri, July 28, 1970.

7. J. H. Crawford and M. C. Wittels, "Radiation Stability of Nonmetals and Ceramics", Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958, Vol. 5, United Nations, Geneva, 1958, pp. 300-310.

8. G. W. Gibson and O. K. Shupe, Annual Progress Report on Fuel Element Development for FY 1961, U. S. Atomic Energy Commission Report, IDO-16727, March, 1962.

9. W. J. Werner and J. R. Barkman, "Characterization and Production of U_3O_8 for the High Flux Isotope Reactor", ORNL-4052, April, 1967.

10. D. L. McElroy, R. S. Graves, and J. P. Moore, "Physical Properties of Two-Phase Materials Used in Fuel Cores", Metals and Ceramics Division Annual Progress Report, ORNL-4570, October, 1970.

11. Reactor Engineering Branch Annual Report, IN-1335, Idaho Nuclear Corporation, November, 1969.

12. A. J. Chapman, Heat Transfer, McMillan Publishing Co., New York, 1974.

13. L. F. Moody, "Friction Factors for Pipe Flow", Transactions ASME, Vol. 66, No. 8, 1944.

14. M. M. El-Wakil, Nuclear Heat Transport, Intext Educational Publishers, New York, 1971, p. 248.

15. W. Dienst, S. Nazare, and F. Thummler, "Irradiation Behavior of UAl_x Dispersion Fuels for Thermal High Flux Reactors", Journal of Nuclear Materials, 64, 1977

16. M. Graber and R. Hobbins, "Irradiation Testing of Sample Fuel Plates to Very High Burnups", INC-16-1, U. S. Atomic Energy Commission Report ANCR-1016, October, 1971.

17. R. Hobbins, "Irradiation Experiment", INC-16-2, Aerojet Nuclear Company, 1974.

18. M. Graber, "Radiation Effects on Dispersion Fuels", Annual Progress Report on Reactor Fuels and Materials Development, U. S. Atomic Energy Commission Report IDO-17154, February, 1966.