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PERFORMANCE EVALUATION OF CORE II AND III
ADVANCED TEST REACTOR FUEL ELEMENTS

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ABSTRACT

Post-irradiation examinations were performed on five fuel assemblies that had been in service in the Advanced Test Reactor (ATR) during startup Cores II and III. Both UAl_x and U_3O_8 were used as the fissile dispersants in these aluminum clad, plate type, dispersion fuel elements. The examination showed that the elements had performed well. The ATR fuel design basis was for a 17 day, 4250 MWd cycle. Three of the fuel elements examined had been in service for over twice the design exposure, or 1.5×10^{21} fission/cc of plate core. A few of the plates were found to have cladding ruptures attributed to thin cladding at core ends.

SUMMARY

Five fuel assemblies used during the early stages of operation of the Advanced Test Reactor (ATR) were given post-operation examinations. This reactor uses aluminum clad, plate type, fuel elements. Both U_3O_8 and UAl_x powders have been used as the fissile dispersants.

It was the purpose of this examination to evaluate the performance of the commercially produced fuel elements by making comparisons to laboratory produced samples previously tested. The elements were metallographically examined, dimensioned, analyzed for ^{235}U depletion, and blister anneal tested. The post-irradiation blister anneal testing was conducted on samples removed from the plates of these elements. These tests indicated that in general the commercially produced fuel plates had a blister resistance at least as good as had been experienced on the laboratory produced samples. Plates fueled with UAl_x powder continued to out-perform those fueled with U_3O_8 . Extensive gamma scanning and chemical burnup determinations were performed on the plates in order to determine the locations and magnitudes of the peak fuel burnup areas. As had been predicted, the peak burnups were found to be 1.5×10^{21} fiss/cc of plate core.

The original design basis was for a fuel life of 17 days of full power operation (4250 MWd). This would have resulted in a peak fuel burnup of 7×10^{20} fiss/cc of fuel plate core. However, because of the favorable results obtained from developmental efforts, it was possible from the second full power reactor cycle onward to extend the fuel element life to over twice this exposure, (1.5×10^{21} fiss/cc). The post-test examinations demonstrated that this extension of the fuel element life was warranted. In these elements, a few of the plates were found to have small cladding ruptures. These were located at the ends of the fuel plate cores and were attributed to thin cladding.

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PERFORMANCE EVALUATION OF CORE II AND III

ADVANCED TEST REACTOR FUEL ELEMENTS

I. INTRODUCTION

The Advanced Test Reactor (ATR) is one of the world's largest test reactors with an operating thermal power level of 250,000 kilowatts. Situated in close proximity to the Materials Testing Reactor (MTR) and Engineering Test Reactor (ETR) at the National Reactor Testing Station in Idaho, the ATR will be used in the development of advanced reactor cores as well as advanced fuel for power and space programs.

The ATR is designed to provide an extremely high neutron flux (1×10^{15} thermal) environment for a multiplicity of high-pressure loops. The cylindrical symmetry of the experiment loops, the diameter of these loops (averaging about 3 inches) and the large number of samples which can be irradiated at one time make this an extremely useful and flexible irradiation facility. The reactor is a light-water moderated and cooled system which employs the flux concentration principle (flux traps) to achieve very high neutron flux levels for a given reactor core power density. The core geometry used is similar to that of a four-leaf clover with four flux traps in the leaves (or lobes), one where the leaves join, and four in the spaces between the leaves. In five of the nine experimental spaces, flux levels can be adjusted to different initial values. Each test space has individual spectral adjustments. Such time-and-money-saving flexibility could have been accomplished otherwise only by nine separate reactors.

During the startup operations of the ATR, four loadings of fuel elements were used. These were designated Cores 0, I, II, and III. Core 0 was used for zero power physics studies to establish the relationship between the Advanced Test Reactor Critical Facility (ATRC) and the ATR⁽¹⁾. No fuel elements from this core were destructively examined. Core I was used for intermediate power operation where fuel element performance was evaluated under a number of contrived thermal-hydraulic conditions⁽²⁾.

The last phase of the ATR startup program was designed to demonstrate the high-power operational capabilities of the reactor⁽³⁾. Two cores were used for this. During the balanced power demonstration tests, Core II was used. Three elements from this core, two of which had been operated in peak power positions, were destructively examined. The fuel elements in this core were operated for 35 full power days. This was twice the design life since it was originally planned that the fuel elements would have a life of 17 days⁽⁴⁾.

Core III was used for the final phase of the startup program. This core was used during the operation of the reactor for 27 days at 250 MW maintaining an unbalanced power division. In the unbalanced power mode of operation used during this demonstration, 40 MW of power were generated in two lobes of the reactor core, 60 MW were generated in two other lobes, and 50 MW were generated in the center lobe. Two fuel elements were destructively examined following this test run. After the successful completion of the startup operations, the reactor was placed into routine operation. At this time, the fuel element loadings were designated by cycle numbers starting with cycle number 1.

This report is being written to describe the results of the post-irradiation examination of these fuel elements. The post-irradiation examinations were conducted for the following purposes:

- (1) To determine the distribution of fuel burnup in the plates of the elements.
- (2) To assess the stability of the fuel elements under actual operating conditions.
- (3) To compare the performance of full size commercially produced fuel elements with that projected from the testing of miniature fuel plates.
- (4) To determine if the safe operating limits of the current fuel element designs for the ATR have been approached or exceeded.
- (5) To determine the cause of any performance anomalies.

The operating parameters for the fuel elements examined are listed in Table I.

TABLE I

Operating Parameters of ATR Startup Program Fuel Elements
Which Were Destructively Examined

<u>Fuel Element Number</u>	<u>Used in Core</u>	<u>Day of Operation</u>	<u>Mode of Operation</u>	<u>Type Fuel Element</u>	<u>Irradiation Position</u>
A172C	II	35	250 MW Balanced	Std. U_3O_8	F11
XA3G	II	35	Balanced	7F $UA1_x$	F13
XA8G	II	35	Balanced	7F $UA1_x$	F14
A16D	III	27	250 MW 40-50-60 Unbalanced	Std. $UA1_x$	F11
XA20G	III	27	Unbalanced	7F $UA1_x$	F13

II. FUEL ELEMENT DESCRIPTION

A schematic diagram of the ATR core is shown in Figure 1. It consists of 40 fuel elements arranged in a serpentine annulus between and around the nine flux traps. Each fuel element consists of 19 parallel curved plates of different width, attached to side plates forming a 45 degree sector of a circular annulus in cross section. Each of the 19 fuel plates is 49.5 inches long with an active fuel length of 48 inches. The 17 inner fuel plates (2 through 18) are 0.050 inch thick with a 0.020 inch thick fuel core and 0.015 inch of cladding. The two outermost fuel plates (1 and 19) of each element are made with extra thick clad to withstand the greater hydraulic pressure differential resulting from the variability of the outer coolant channels. These channels are formed by the outer fuel plates and the other reactor core components. The outside plate with the greatest radius of curvature (plate 19) is 0.100 inch thick with 0.040 inch of clad on each side of the core. The inside plate with the smallest radius of curvature (plate 1) is 0.080 inch thick with 0.030 inch of clad on each side of the core.

Figure 1 - Schematic cross section at core mid-plane.

The side plates are vented by localized slots to provide inter-connection between all channels and thereby reduce lateral pressure differentials. A typical arrangement is shown in Figure 2. Pads are included on the exterior surface of the side plates to ensure adequate coolant channels between elements and to provide a minimum flow path for interconnecting flow through the vents and external channels.

The fuel element end fittings are designed for support and positioning of the elements on the flux trap baffles of the reactor core. The end fittings are attached to the fuel element side plates by welding and provide minimum restriction to coolant flow through the element. In addition to providing a lifting handle, the top-end fitting includes a curved baffle to direct coolant flow into the fuel channels with a minimum of turbulence and to prevent mal-distribution. A comb is positioned across the top and bottom mid-span edges of the fuel plates to provide additional support and spacing.

Figure 2 - Fuel element details.

During the Cores II and III startup cycles, three types of fuel elements were used (U_3O_8 standard, UAl_x standard, and 7F). In all three, the fuel element geometry is as illustrated in Figure 2, but the fuel material and its distribution differ. Each type was given a post-irradiation examination.

The differences between the three types of elements and the materials used in the construction of the elements are indicated in Table II. The standard U_3O_8 type element was the reference fuel system for the reactor⁽⁵⁾, and the first elements procured for the ATR were this type. However, during the design and construction of the ATR fuel elements, powdered uranium-aluminide compounds (UAl_x) were developed⁽⁶⁾. Irradiation testing of miniature fuel plates demonstrated that the plates containing UAl_x would tend to be more stable than those fueled with U_3O_8 ^(7,8). Consequently, powdered uranium-aluminide is now the standard fuel for the ATR.

Thermal-hydraulic studies made on the fuel elements while the reactor was under construction revealed that in the hot-spots of the reactor core the localized stresses on the fuel plates would exceed the yield strength of the plates. In order to overcome this, a new fuel loading prescription was developed. The zone loaded elements made to this prescription were designated 7F type elements. As indicated in Table II, three loadings of ^{235}U and boron are used in the fuel plate cores of the 7F type elements. In the standard fuel elements, the weight of fuel and boron in a given volume of core is the same in each of the 19 plates.

TABLE II

Fuel Element Materials and Loading

Type Element	Standard U_3O_8	Standard UAl_x	7F UAl_x
Total Wt. ^{235}U - g	975	975	1075
Fuel Loading, g ^{235}U /cc Core			
Plates 1, 2, 18, 19	1.22	1.22	0.976
Plates 3, 4, 16, 17	1.22	1.22	1.232
Plates 5 through 15	1.22	1.22	1.488
Fuel Material	U_3O_8	UAl_x	UAl_x
Core Matrix Material	X8001	X8001	X8001
Burnable Poison Material	B_4C	B_4C	B_4C

TABLE II (Contd.)

Boron Loading, mg Nat B/cc Core			
Plates 1, 2, 18, 19	5.2	5.2	14.2
Plates 3, 4, 16, 17	5.2	5.2	7.5
Plates 5 through 15	5.2	5.2	1.1
Plate Cladding Material	6061-0 Al	6061-0 Al	6061-0 Al
Side Plate Material	6061-T6 Al	6061-T6 Al	6061-T6 Al
End Fitting Material	356 Al	356 Al	356 Al

III. FUEL ELEMENT EXAMINATIONS

Upon reaching the hot cells, the fuel elements were visually inspected under normal and oblique lighting, optically inspected at macro and micro magnifications and microconstituent analyses were made with the electron microprobe. Measurements were made of total plate lengths before and after disassembly, of plate thicknesses before and after chemical stripping of oxide thickness and of plate volumes before and after stripping. Gamma scans were made to complement burnup analysis and samples were blister-tested.

It should be pointed out that not all of the examinations were made on every element or on all the plates of any one element. The plates to be used for a certain examination were selected on the basis of providing the best compromise between the needs of thermohydraulics, physics calculations, metallurgy, and budget considerations.

The most thorough examination was made on XA3G. The effort was reduced on succeeding elements. Certain of the plates from some of the elements had complete plate thickness and oxide thickness surveys while on others only maximums were determined. Certain of the plates had complete gamma scan surveys while others had only centerline scans taken. Some metallography samples were taken for the purpose of oxide evaluation, others for fuel stability and still others for post-anneal blister observations. Burnup samples were selected from locations which cover the range of gamma activities for correlation with the gamma scans. The various examinations which were performed on the plates from each of the fuel elements are indicated in Table III.

1. Visual Examination

All the elements were carefully inspected visually. After the end boxes were removed but before the plates were removed from the side plates, the fuel plates and channels were examined by axial viewing under back lighting conditions (Figure 3). After the fuel plates were removed from the side plates, each was inspected using regular and oblique lighting and with stereo magnification. The plates exhibited good stability. The visual examination showed a light color oxide coating on the surface of the plates which in general appeared uniform over the plate surfaces. Darker areas appeared at the edge of the core near the center of certain plates (Figure 4). This discoloration appears to be a temperature effect which occurred at burnup of 1.3×10^{21} fission/cc and above. Corrosion metallography, oxide thickness measurements, and plate thickness measurements did not detect spalling in these areas.

Anomalies did appear at the end of the core of certain plates at a frequency rate of about one per 100 fuel plates. As discussed in another section of this report, these anomalies were called "pimples." It was

TABLE III

Post-Irradiation Examination of ATR Cores II and III Fuel Elements

<u>Fuel Element</u>	<u>Plate Number</u>	<u>Measure Length</u>	<u>Visual Inspection</u>	<u>Oxide Thick. Survey</u>	<u>Oxide Thick. Max. Only</u>	<u>Detailed Gamma Scan</u>	<u>Gamma Scan Only</u>	<u>Thick. Meas. Survey</u>	<u>Thick. Meas. Max. Only</u>	<u>Oxide Met. Sample</u>	<u>Chem. Strip</u>	<u>Thick. Measurement</u>	<u>Determine Density</u>	<u>B.U. Samples</u>	<u>Met. Samples</u>	<u>Blister Samples</u>
XA3G	1	X	X	X		X		X			X	X	X	X		
	2	X	X				X		X	X					X	X
	3	X	X	X		X		X			X	X	X	X		
	4		X				X		X							
	5	X	X	X		X		X			X	X	X	X		
	6		X		X		X		X							
	7		X				X		X							X
	8		X		X		X		X							
	9		X		X		X		X	X						
	10	X	X	X		X		X			X	X	X	X		
	11		X		X		X		X							
	12		X		X		X		X							
	13		X		X		X		X							
	14		X		X		X		X	X					X	
	15	X	X	X		X		X			X	X	X	X	X	X
	16		X				X		X	X					X	X
	17	X	X	X		X		X			X	X	X	X		
	18	X	X		X		X		X							
	19	X	X	X		X		X			X	X	X	X		

TABLE III (Contd.)

<u>Fuel Element</u>	<u>Plate Number</u>	<u>Visual Inspection</u>	<u>Oxide Thickness Survey</u>	<u>Oxide Thickness Max. Only</u>	<u>Detailed Gamma Scan</u>	<u>Gamma Scan Only</u>	<u>Thick. Measurements Survey</u>	<u>Thick. Meas. Max. Only</u>	<u>Oxide Met. Sample</u>	<u>B.U. Samples</u>	<u>Met. Samples</u>	<u>Blister Samples</u>
XA8G	1	X	X		X		X					
	2	X		X		X		X	X			
	3	X	X		X		X					
	4	X		X		X		X				
	5	X	X		X		X			X		
	6	X		X		X		X				
	7	X		X		X					X	X
	8	X		X		X		X				
	9	X		X		X		X	X			
	10	X	X		X		X			X		
	11	X		X		X					X	X
	12	X		X		X		X				
	13	X		X		X		X				
	14	X		X		X		X				
	15	X	X		X		X					
	16	X		X		X		X				
	17	X	X		X		X					
	18	X		X		X		X	X			
	19	X	X		X		X			X		

TABLE III (Contd.)

Fuel Element	Plate Number	Visual Inspection	Oxide Thick. Survey	Oxide Thick. Max. Only	Detailed Gamma Scan	Gamma Scan Only	Thick. Meas. Survey	Thick. Meas. Max. Only	Oxide Met. Sample	Thickness Measurement	B.U. Samples	Blister Samples
A172C	1	X	X				X			X	X	
	2	X	X		X		X		X	X	X	X
	3	X		X		X		X				
	4	X		X		X		X				
	5	X		X		X		X				
	6	X		X		X		X				
	7	X		X		X		X				
	8	X		X		X		X				
	9	X		X		X		X				
	10	X	X		X		X			X	X	
	11	X		X		X		X				
	12	X		X		X		X				
	13	X		X		X		X				
	14	X		X		X		X				
	15	X		X		X		X				
	16	X		X		X		X				
	17	X		X		X		X				
	18	X	X		X		X		X	X		X
	19	X	X		X		X			X		

TABLE III (Contd.)

<u>Fuel Element</u>	<u>Plate Number</u>	<u>Measure Length</u>	<u>Visual Inspection</u>	<u>Oxide Thick. Survey</u>	<u>Oxide Thick. Max. Only</u>	<u>Detailed Gamma Scan</u>	<u>γ Gamma Scan Only</u>	<u>Oxide Met. Sample</u>	<u>B.U. Samples</u>
XA20G	1	X	X		X				
	2	X	X		X		X		
	3	X	X	X		X		X	
	4		X		X		X		
	5	X	X	X		X		X	X
	6		X		X		X		
	7		X		X		X		
	8		X		X		X		
	9		X		X		X		
	10	X	X	X		X		X	X
	11		X		X		X		
	12		X		X		X		
	13		X		X		X		
	14		X		X		X		
	15	X	X	X		X		X	
	16		X		X		X		
	17	X	X	X		X			
	18	X	X		X		X		
	19	X	X	X		X		X	X

TABLE III (Contd.)

<u>Fuel Element</u>	<u>Plate Number</u>	<u>Measure Length</u>	<u>Visual Inspection</u>	<u>Oxide Thickness Max. Only</u>	<u>Detailed Gamma Scan</u>	<u>Gamma Scan</u>	<u>B.U. Samples</u>
A16D	1		X		X		X
	2		X		X		X
	3		X	X		X	
	4		X	X		X	
	5		X	X		X	
	6		X	X		X	
	7		X	X		X	
	8		X	X		X	
	9		X	X		X	
	10		X		X		X
	11		X	X		X	
	12		X	X		X	
	13		X	X		X	
	14		X	X		X	
	15		X	X		X	
	16		X	X		X	
	17		X	X		X	
	18		X		X		
	19		X		X		

found that these were caused by thin cladding at the dogbone. As shown in Figure 5, these pimples were generally quite small $\sim 1/32 - 1/16$ inch and exhibited a comet tail shaped discoloration streak on the downstream side which ranged from $1/4$ to 1 inch long.

Figure 3 - Axial view of element XA3G.

Figure 4- View of fuel plates 10 through 19 from element XA3G after disassembly.

Figure 5 - View of the top end of plate 7 from element XA8G showing a pimple and associated "comet tail" discoloration streak (arrow).

2. Burnup Determinations

The two basic methods employed to obtain measured information concerning fuel plate burnup in the elements examined from ATR Cores II and III consisted of: (1) analysing chemical burnup samples taken at known locations in the fuel plates, and (2) performing detailed gamma scanning on certain fuel plates to establish the distribution of gamma producing fission products in these fuel plates. The results of both of these methods were then combined to obtain detailed data concerning the number of fissions of uranium atoms which occurred per cubic centimeter of fuel plate core. These data are presented in terms of contour plots which will be developed in the following sections.

2.1 Chemical Burnup Sampling. During the post-irradiation examination of the Cores II and III elements, 1/4 inch burnup samples were punched from known locations in selected plates. Figure 6 shows the burnup sample location plan for each of the Core II and Core III elements examined except A16D and XA20G. The "A" and "C" planes noted in Figure 6 were nominally 3/8 inch from the edge of the fuel plate (after the fuel plates were machined free from the side plates). Both plate outlines sketched in Figure 6 represent fuel plates in the concave position. It can be seen that nominally 120 burnup samples (from each element) were taken from plates 1, 3, 5, 10, 15, 17, and 19 from XA3G and XA8G. Approximately 86 burnup samples were taken from plates 1, 2, 10, 18, and 19 from A1720. Those punchings that were selected (from the Cores II and III elements) and sent to the Analytical Chemistry Branch of Idaho Nuclear Corporation[a] for analysis are reported in Table IV. The remaining samples were held in reserve. The estimated ^{148}Nd percent fissioned values listed in Table IV were determined from Figure 7.

This figure compares the ASTM 1 isotopic burnup results with the ^{148}Nd burnup results reported from the Analytical Chemistry Branch. It can be seen that this plot is a well defined linear relationship with only two slightly deviating points. This method of estimation was employed (rather than 100% ^{148}Nd burnup analysis) because of the much higher cost of the neodymium method compared with the isotopic methods.

The estimated random error ranges (from Analytical Chemistry Branch experience) in the isotopic values and ^{148}Nd values (as reported in Table IV) are 4-6% and 3-4% relative standard deviation, respectively. It can be seen in Table IV that the isotopic methods also have systematic errors associated with the determinations. That is, these values are always less than the burnup calculated by the neodymium method. This deviation seems to occur (in the isotopic method) because ^{238}U and ^{236}U contents are assumed to remain constant during burnup of the fuel. These values, however, do change during irradiation. The ^{148}Nd method is not subject to this same type of systematic error because heavy atoms other than uranium are not present in ETR and ATR fuel and this method takes into consideration total uranium fissions (i.e., for the change in atom percent of all uranium isotopes are calculated).

[a] Now Aerojet Nuclear Company.

Figure 6 - Burnup sample location plan - 1/4" diameter circular punch.

TABLE IV

Burnup Analysis Results for Elements Examined from Cores II and III

Element Number	Sample Number ⁽¹⁾	²³⁵ U Atoms-% Fissioned ASTM 1 Isotopic Method	U Atoms-% Fissioned ¹⁴⁸ Nd Method	Fission Density ⁽³⁾ from ¹⁴⁸ Nd Method-F/ccx10-21	Calculated
XA3G	1-9	33.4	37.8 ⁽²⁾	1.02	
"	1-10	30.0	34.0 ⁽²⁾	0.958 ⁽⁴⁾	
"	1-11	33.0	37.8	1.02	
"	3-9	26.7	30.2 ⁽²⁾	1.02	
"	3-10	23.3	28.7	0.973	
"	3-11	26.1	29.6 ⁽²⁾	1.00	
"	5-9	23.1	26.2 ⁽²⁾	1.07	
"	5-10	20.0	24.9	1.02	
"	5-11	23.3	26.5 ⁽⁵⁾	1.09	
"	10-11	23.0	26.1	1.07	
"	10-12	18.1	20.8 ⁽²⁾	0.852	
"	10-13	23.6	26.7 ⁽²⁾	1.09	
"	15-11	27.2	30.9 ⁽²⁾	1.27	
"	15-12	22.9	26.0 ⁽²⁾	1.07	
"	15-13	28.1	31.9 ⁽²⁾	1.31	
"	17-11	33.2	37.7	1.28	
"	17-12	29.1	32.8	1.11	
"	17-13	32.5	36.3	1.23	
"	19-1	11.4	13.3 ⁽²⁾	0.358	
"	19-2	24.3	27.7 ⁽²⁾	0.745	
"	19-4	34.3	38.8 ⁽²⁾	1.04	
"	19-8	35.5	40.1 ⁽²⁾	1.08	
"	19-11	37.9	42.8	1.15	
"	19-12	40.4	45.4	1.16 ⁽⁴⁾	
"	19-13	41.1	46.5	1.25	
"	19-15	34.2	38.7	1.04	
"	19-17	24.8	28.3	0.76	
"	19-18	12.5	14.7	0.395	
XA8G	5-10	18.4	22.9	0.939	
"	10-12	16.5	19.1 ⁽²⁾	0.514	
"	19-12	31.9	36.3 ⁽²⁾	0.976	
"	19-15	28.4	32.5 ⁽²⁾	0.874	
"	19-17	19.1	22.1 ⁽²⁾	0.594	
XA20G	19-17		24.2	0.651	
"	5-10		23.8	0.976	
"	10-12		20.3	0.832	
"	10-1 ⁽⁵⁾		5.0	0.205	
"	19-12		39.1	1.05	

TABLE IV (Contd.)

Element Number	Sample Number ⁽¹⁾	²³⁵ U Atoms-% Fissioned ASTM 1 Isotopic Method	U Atoms-% Fissioned ¹⁴⁸ Nd Method	Fission Density ⁽³⁾ Calculated from ¹⁴⁸ Nd Method-F/ccx10 ⁻²¹
A172C	2-10		38.4	1.29
"	10-12		26.6	0.894
"	10-1		10.6	0.356
"	1-10		42.3	1.42
"	1-1		14.9	0.501
A16D	1-1 ⁽⁵⁾		11.4	0.383
"	10-1 ⁽⁵⁾		4.5	0.151
"	1-10		35.8	1.20
"	2-10		32.1	1.08
"	10-12		21.3	0.716

- (1) The first number denotes the plate. The second number gives the position according to Figure 6.
- (2) These numbers were taken from Figure 7 and therefore represent estimated data.
- (3) These numbers were calculated from the ¹⁴⁸Nd percent fissioned values above by multiplying by the nominal U atoms/cc for each plate type.
- (4) These samples were repunched at 1/4 inch from the position indicated along the longitudinal centerline.
- (5) These samples were taken 2.75 inches from the top of the plate along the longitudinal centerline.

The fission density values calculated and listed in Table IV result from multiplying the ^{148}Nd burnup values by the nominal atoms U/cc for each ATR 7F plate type. The nominal values used for ^{235}U concentration in ATR 7F cores are⁽⁹⁾: 0.976 g ^{235}U /cc for plates 1, 2, 18 and 19; 1.232 g ^{235}U /cc for plates 3, 4, 16, and 17; and 1.488 g ^{235}U /cc for plates 5-15. These values convert to (assuming 93% enrichment) 2.69×10^{21} atoms U/cc for plates 1, 2, 18, and 19; 3.39×10^{21} atoms U/cc for plates 3, 4, 16, and 17; and 4.10×10^{21} atoms U/cc for plates 5-15.

The nominal values used for ^{235}U concentration in ATR regular cores are 1.22 g ^{235}U /cc of core or 3.36×10^{21} atoms U/cc. All of these values are based on a nominal core thickness of 20 mils.

Least squares plots of the centerline fission density (with 95% confidence limits on either side) versus normalized gamma amplitude for elements XA3G and XA20G are given in Figures 8 and 9, respectively. The burnup samples taken from the "A" or "C" planes were not included in Figures 8 and 9 because of the sharp peaking in burnup near the edge of the fuel plates. The standard deviations (S_y) of the centerline data points about the least squares fit line are 7.23×10^{19} ($S_y/\bar{y} = 8.05\%$) and 9.11×10^{19} ($S_y/\bar{y} = 12.3\%$) for XA3G and XA20G, respectively. The equations of these lines were used in the computer program CONTOR to convert gamma amplitude input data to fission density output in the form of contour plots.

2.2 Data Obtained from Gamma Scanning

The experimental set up used in the hot cell for gamma scanning is presented in Figure 10. Here it is seen that the fuel plate or subject is supported approximately 9 inches from the end of the collimator slit and moved at a constant rate during the scanning operation. The rate of speed employed when the plates were scanned in the length direction was 2.4 inches/min. and the speed during cross scanning was 1.75 inches/min. The collimator slit used during the Cores II and III element examinations was 0.50 inch x 0.019 inch x 11.75 inches deep. With the type of geometry shown in Figure 10 and using the dimensions above, it can be shown that the gamma signal emitted from the fuel plate during scanning comes from an area 1.25 inches long x 0.048 inch wide. During the centerline scanning operations, the slit length (0.50 inch) was parallel with the ends of the fuel plate (top and bottom). During all other scanning operations ("A" and "C" planes and cross scans), the slit length was parallel with the sides of the fuel plate. All plates were in the concave position with respect to the detector, when scanned. The scanning was carried out in this manner because of the flux peaking previously mentioned along the sides of the plates. This peak burnup along the sides of the plate (because of scanning geometry shown above) is always averaged over at least 0.048 inch. This type of slit orientation is designated to provide the best possible resolution at the areas of maximum burnup.

Figure 7 - Chart used for conversion of isotopic burnup to neodymium burnup.

Figure 8 - Correlation curve of fission density versus normalized gamma amplitude for fuel element XA3G with 95% confidence limits.

Figure 9 - Correlation curve of fission density versus normalized gamma amplitude for fuel element XA20G with 95% confidence limits.

Figure 10 - Gamma scan schematic.

2.3 Iso-Fission Density Contours

The computer program CONTOR receives values of amplitude from gamma scans as input. For each fuel plate, this input is generally about 140 - 150 points for each length scan and 30 - 35 points for each of nine cross scans. These values are then normalized by subtraction of a baseline amplitude. They are then stored as a matrix and converted to fission density by the equation of a line such as those given in Figures 8 and 9. Linear interpolation is then performed between the stored data sets to provide fission density contours at known intervals.

Iso-fission density plots of three plates from XA3G (plates 1, 3, and 15) and three plates from XA20G (plates 5, 15, and 19) are given in Figures 11 through 16. The contour interval (i.e. the amount of change in fission density between adjacent lines) is given at the top left hand side of each plot. The multiplication factor for the labeled values of fission density is given at the top near the center. Also, the approximate length and width of each fuel plate core is labeled in inches. All fission density plots are viewed in the concave direction and the top end of each plate is at zero inches.

The areas of maximum flux peaking and hence maximum burnup generally occur between 20 and 30 inches along the plate length and on either edge of the plate. This gradient can be seen by the closely spaced contour lines in these areas. It should be noted that these contour plots are primarily intended to provide a general visual indication of the burnup patterns which occur in these types of fuel plates.

2.4 Cumulative Accuracy of Analysis

A measure of the total accuracy of the burnup values given is approximately equal to the combined variance of the least squares fit line which related gamma amplitude to fission density and the variance of the points about the least squares fit line. The random error present in the ^{148}Nd burnup results should be part of the variance of the data points about the correlation line. A confidence interval about the correlation line may be calculated at any desired confidence. At a 95% confidence interval, the band about the correlation line at the peak burnup in elements XA3G and XA20G is $\pm 1.2 \times 10^{20}$ fissions/cc and $\pm 3.5 \times 10^{20}$ fissions/cc, respectively. Because CONTOR has predicted maximum fission density values in XA3G (plate 15) and XA20G (plate 15) on the "C" edge of

the fuel core between 24 and 25 inches from the top of the fuel plate of 15.0×10^{20} fissions/cc for both plates, it can be stated with 95% confidence that the true peak fission densities in these plates (assuming no instrument error) was not greater than 16.2×10^{20} fissions/cc and 18.5×10^{20} fissions/cc, respectively.

2.5 Plate-to-Plate Distribution of Burnup

The maximum longitudinal centerline and maximum peak plate-to-plate burnup distributions in elements XA3G, XA20G, and A172C are given in Figures 17 through 19, respectively. It can be seen from these figures that the plate-to-plate burnup distribution is quite different when standard ATR and ATR 7F elements are compared. Both of the 7F elements show large changes in plate-to-plate burnup in the regions of plates 1-5 and plates 15-19, maxima generally occur at plates 5 and 15, and a generally uniform burnup pattern is achieved between plates 8-13. The single ATR standard element which has been evaluated in detail shows steadily decreasing burnup values from plates 1 to 19. The data used to plot Figures 17 through 19 are given in Tables V, VI, and VII.

Figure 11 - Iso-fission density for plate 1 of XA3G.

Plate Width

Figure 12 - Iso-fission density for plate 3 of XA3G.

Figure 13 - Iso-fission density of plate 15 of XA3G.

Contour Interval = 5×10^{19}

Multiply Levels by 1×10^{19}

Figure 14 - Iso-fission density of plate 5 of XA20G.

Figure 15 - Iso-fission density of plate 15 from XA20G.

Figure 16 - Iso-fission density of plate 19 of XA20G.

Figure 17 - Plot of peak plate burnup and maximum burnup along the longitudinal centerline versus fuel plate location in XA3G.

Figure 18 - Plot of peak plate burnup and maximum burnup along the longitudinal centerline versus fuel plate location in XA20G.

Figure 19 - Plot of peak plate burnup and maximum burnup along the longitudinal centerline versus fuel plate location on A172C.

TABLE V

Plate-to-Plate Distribution of Burnup in XA3G

<u>Plate Number</u>	<u>A</u> Maximum Longitudinal Centerline Burnup-Fiss/cc x 10 ⁻²¹	<u>B</u> Maximum Plate Burnup Fiss/cc x 10 ⁻²¹	<u>$\frac{B - A}{B} \times 100\%$</u>
1	1.04	1.12	7.4
2	0.966		
3	1.07	1.36	21
4	0.945		
5	1.07	1.42	25
6	1.03		
7	0.913		
8	0.949		
9	0.940		
10	0.943	1.38	32
11	0.945		
12	0.962		
13	1.02		
14	1.06		
15	1.10	1.50	27
16	1.06		
17	1.16	1.44	19
18	0.987		
19	1.03	1.23	16

TABLE VI

Plate-to-Plate Distribution of Burnup in XA20G

Plate Number	A	B	$\frac{B - A}{B} \times 100\%$
	Maximum Longitudinal Centerline Burnup-Fiss/cc x 10^{-21}	Maximum Plate Burnup Fiss/cc x 10^{-21}	
1	Not Gamma Scanned		
2	0.896		
3	1.02	1.31	23
4	0.876		
5	1.07	1.43	25
6	0.955		
7	0.906		
8	0.856		
9	0.860		
10	0.856	1.31	34
11	0.876		
12	0.886		
13	0.876		
14	0.965		
15	1.03	1.50	31
16	0.896	1.31	24
17	0.995		
18	0.925		
19	0.918	1.12	18

TABLE VII

Plate-to-Plate Distribution of Burnup in A172C

Plate Number	A	B	$\frac{B - A}{B} \times 100\%$
	Maximum Longitudinal Centerline Burnup-Fiss/cc x 10^{-21}	Maximum Plate Burnup Fiss/cc x 10^{-21}	
1	1.42		
2	1.28	1.48	13
3	1.22		
4	1.12		
5	1.13		
6	1.02		
7	1.03		
8	0.990		
9	1.00		
10	0.863	1.12	23
11	0.952		
12	0.950		
13	0.920		
14	0.940		
15	0.937		
16	0.940		
17	0.926		
18	0.904	1.08	16
19	0.962	1.17	18

3. Corrosion

Detailed oxide thickness measurements were made on plates 1, 3, 5, 10, 15, 17, and 19 from XA3G, XA8G, and XA20G. Detailed oxide data were also obtained on plates 1, 2, 10, 18, and 19 from A172C. The remainder of the plates were generally scanned for maximum oxide thickness only. The maximum observed oxide thickness readings on the convex sides of all of the plates measured in the examination of the Cores II and III elements are given in Table VIII. Some measurements were made of the oxide film thickness on the concave side of the plates. These measurements proved to be unreliable, and are not reported in Table VIII. The 1/2 inch diameter eddy current probe used for the measurements is believed to be the source of the problem and a new probe with a smaller diameter is being developed. It can be seen that the maximum accumulation of oxide generally was in the range from 0.6 to 1.7 mils on the convex side of the fuel plates.

Figure 20 compares oxide thickness measurements made on the convex side of plates from XA20G employing the eddy current device (1/2 inch probe) and the range of oxide thicknesses measured from photomicrographs of mounted fuel plate sections. Since the high points in the oxide will determine the distance separating the eddy current probe from the cladding surface, it is anticipated that the thickest oxides observed metallographically should make the best correlation with the eddy current measurements.

Metallographic examination of the oxide films showed no significant difference in the average film thickness on the convex and concave sides of the interior plates. Both eddy current and metallographic measurements indicate, however, that on plates 1 and 19 the oxide is thicker on the surfaces which face the inside water channels of the elements than it is on the surfaces which face the outside water channels.

X-ray analyses of oxide scrapings from Cores II and III elements revealed that the oxide contained both bayerite and boehmite.

Metallography frequently showed the oxide to be made up of two layers. As is shown in the photomicrographs in the metallography section of this report, the inner layer contains cracks normal to the cladding surface. The outer layer does not contain these cracks but appears to be more porous than the inner layer. The boundary between the two layers is distinct, often consisting of a crack separating the layers. The outer layer is subject to spalling. This fact is evidenced by both the presence and absence of the outside layer within the same metallographic mount as shown in Figure 27.

Oxide thicknesses from the metallography of element XA3G are plotted as a function of calculated⁽¹⁰⁾ beginning of life surface temperature in Figure 21. The inner layer thicknesses correlate reasonably well with the equation:

$$X = 332t^{0.778} e^{-8290/^{\circ}R}$$

which is equivalent to 3/4 of the rate of oxide buildup observed out of pile by Griess(11). In the equation X is the oxide thickness in mils, t the time of irradiation in hours, and $^{\circ}R$ the oxide surface temperature in degrees Rankine. The oxide buildup (~ 0.1 mil) which occurs during the hydraulic testing of the elements has not yet been characterized and was therefore not included in this relationship. In addition, because of the small number of data points employed, no great confidence can be placed upon the correlation with 3/4 Griess, per se. However, it is significant to note in Figure 21 that the outer layer exhibits a much weaker temperature dependence than does the inner layer. This suggests that the activation energy of the growth controlling mechanism is considerably lower in the case of the outer layer compared to the inner layer. It is of considerable interest to discover how and when this outside layer of oxide forms. Work along this line is continuing.

In Figure 21, a total oxide thickness of 0.92 mil occurs at the inverse temperature $1.392 \times 10^{-3} \text{ }^{\circ}R^{-1}$ ($259^{\circ}F$). The appropriate nominal heat flux here is $1.02 \times 10^6 \text{ Btu/hr-ft}^2$ (10) and, using a thermal conductivity of $1.3 \text{ Btu/hr-ft}^{\circ}F$ (11), a temperature increase across the oxide of $96^{\circ}F$ is calculated. This means that the cladding experiences temperatures up to $355^{\circ}F$. At this temperature, 6061-0 Al (the cladding material) retains about 90% of its room temperature strength(12).

TABLE VIII

Maximum Oxide Thickness Observed on Convex Sides
of Fuel Plates - mils

Plate Number	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>	<u>9</u>	<u>10</u>
XA3G	1.35		1.19		1.52	1.24		1.12	1.05	0.51
XA8G	1.13	0.92	0.95	0.94	1.17	1.00	1.05	0.97	0.85	1.05
XA20G	1.13	1.16	1.16	1.04	1.04	0.76	0.73	0.70	0.71	0.76
A172C	1.48	1.75	1.68	1.35	1.40	1.28	1.43	0.97	1.28	1.08

Plate Number	<u>11</u>	<u>12</u>	<u>13</u>	<u>14</u>	<u>15</u>	<u>16</u>	<u>17</u>	<u>18</u>	<u>19</u>
XA3G	1.04	1.40	0.98	0.99	1.02		0.84	0.67	0.51
XA8G	0.63	0.92	0.88	0.78	0.95	0.60	0.84	0.60	0.49
XA20G	0.56	0.75	0.73	0.52	0.68	0.65	0.68	0.61	0.72
A172C	1.12	1.08	0.83	0.62	0.87	0.62	0.83	1.03	1.32

Figure 20 - Comparison of metallographically determined oxide thickness with oxide thickness determined by eddy current device for XA20G.

ANC-A-166

Figure 21 - Temperature dependence of oxide thickness.

4. Fuel Plate Stability

Visual examination, plate length measurements, plate thickness measurements and density measurements were made to determine the stability of the fuel plates. Table IX gives both pre-irradiation and post-irradiation total plate length measurements for plates 1, 2, 3, 17, 18, and 19 from XA3G. These measurements indicate that essentially no plate lengthening took place and that any swelling which occurred during irradiation increased the plate thickness rather than the plate length.

A detailed gross thickness survey (before stripping) was performed on plates 1, 3, 5, 10, 15, 17, and 19 from XA3G and XA8G and plates 1, 2, 10, 18, and 19 from A172C. Generally, the remaining plates from these elements were surveyed only in the area of maximum thickness (20 to 30 inches from the top of the fuel plate). All thickness measurements were made by use of a dial indicator. On the average, the cumulative error in measurement by this technique was about ± 0.5 mil.

The maximum gross thickness readings obtained on the plates examined are contained in Table X. In this table, it can be seen that the maximum thickness readings for plate 1, plates 2 through 18, and plate 19 were 82.0 mils, 51.5 mils, and 100.4 mils, respectively. These maximum thicknesses were randomly distributed on the plates, did not necessarily occur in high temperature or peak burnup regions, and were not of a sufficiently high magnitude to restrict flow significantly.

It is apparent that the amount of plate thickening above the maximum specified plate thickness is within the magnitude that one would expect from normal growth and corrosion phenomena and that breakaway swelling or other gross deterioration mechanisms have not been initiated. It can also be seen that there is consistency of performance from one element to another.

In order to further define the growth rate of these commercially produced fuel plates which did not have pre-irradiation core density values, a new method was developed. Coupons which were nominally 3 inches x 1.5 inches x plate thickness were cut from the fuel plates in the core areas which cover a range of burnups. Samples taken from the top of the fuel plate which are low in growth and corrosion were used as standards to establish weight per unit area and volume per unit area values. A maximum growth rate which is theoretically possible, but which is small because of low fission density, was assigned to these standard plates. The standards were corrected for this growth value and compared to all other coupons on a " $\Delta V/V$ of fuel core basis" to give a maximum growth value. The data obtained are given in Table XI. The $\Delta V/V$ values are plotted in Figure 22 and compared with growth rates expected from these plates based on their nominal void contents.

It can be seen that growth was being accommodated in the voidage provided in the fuel core. With the exclusion of the control samples, the growth accommodation in plate 1 was slightly better than expected, plates 2 and 4 were within the expected range but plate 8 showed a higher (although not detrimental) growth rate. A possible explanation of this is that the actual void content in plate 8 was lower than the nominal or that the control sample was not truly representative on a weight-per-unit area basis.

All of the growth data showed higher rates than have generally been found in UAl_x fuels, this is due to the fact that plates produced by this supplier (Atomics International) have lower than normal voidage.

Blister testing was limited to the Core II fuel elements XA3G and A172C. Dogbone areas of element XA8G were also tested. Blister samples (see Table XII) were selected to cover a range of burnup 6.0×10^{20} fission/cc to 15.0×10^{20} fission/cc for UAl_x fueled plates and 6.0 to 13.0 for those using U_3O_8 . For this, portions of fuel plates were cut into blister test samples about 7 inches long by plate width. These were heated in air at $50^\circ F$ temperature increments for 30 minutes each. The plates were inspected after each increment until blistering was found to have occurred. Since, when blistering occurred, it was always at the high burnup area of the sample, these values of burnup were plotted against blister temperature (see Figure 23), and compared with previously established blister test data.

It was found that the behavior of these commercially fabricated fuel plates was similar to sample fuel plates irradiated in the G-12 loop of the ETR. The higher w/o UAl_x plates out-performed the intermediate and low w/o UAl_x and the UAl_x compositions out-performed the U_3O_8 fueled plates. In every case, the fuel blister resistance was better than the previously established minimums. Samples taken from the dogbone regions from element XA8G were tested to $1100^\circ F$ ($593^\circ C$) without the formation of a blister.

TABLE IX

Length Measurements for Fuel Plates in ATR
 "7F" Element Number XA3G

Plate Number	<u>1</u>	<u>2</u>	<u>3</u>	<u>17</u>	<u>18</u>	<u>19</u>
[a] As assembled pre-irradiation - in.	49.490	49.498	49.496	49.498	49.493	49.501
[a] As assembled post-irradiation - in.	49.492	49.498	49.397	49.488	49.491	49.496
Disassembled post-irradiation - in.	49.473	49.490	49.489	49.492	49.492	49.488
Length Changes:						
As assembled pre to post - in.	+0.002	0	+0.001	-0.010	-0.002	-0.005
% $\Delta l/l$	+0.004	0	0.002	-0.020	-0.004	-0.010
After removing side plates - in.	-0.019	-0.008	-0.008	+0.004	+0.001	-0.008
% $\Delta l/l$	-0.038	-0.016	0.016	+0.008	+0.002	-0.016

[a] Corrected for thermal expansion.

TABLE X

Maximum Fuel Plate Thickness Readings Obtained Before Stripping - mils^[a]

Plate No.	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>	<u>9</u>	<u>10</u>	<u>11</u>	<u>12</u>	<u>13</u>	<u>14</u>	<u>15</u>	<u>16</u>	<u>17</u>	<u>18</u>	<u>19</u>
XA3G	82.0	50.3	51.3	50.4	50.4	50.1	50.3	50.3	50.1	51.0	50.4	50.2	50.1	50.2	50.5	50.2	51.5	50.1	100.4
XA8G	81.6	52.0	50.2	50.4	49.8	50.4		50.1	49.8	49.3		49.5	49.2	49.3	48.9	48.3	49.3	48.9	99.8
Al72C	79.5	50.4	50.3	49.9	50.5	50.3	49.9	50.8	49.8	50.4	50.4	50.1	49.5	49.9	50.7	50.7	51.1	49.7	100.4

[a] The specified total plate thickness (IN-F-9ATR) prior to irradiation for plate 1, plates 2 through 18 and plate 19 are 80 +0 mils, and 100 +0 mils, respectively.

-2

-2

Figure 22 - Growth rates for fuel plate cores from elements XA8G and XA20G.

TABLE XI

Growth Data for Fuel Plate Cores
from Elements XA8G and XA20G

<u>Element Number</u>	<u>Plate Number</u>	<u>Distance from Top of Plate Inches</u>	<u>Average Fiss/ccx10⁻²⁰</u>	<u>Standards Core Growth % $\Delta V/V$</u>	<u>Maximum Core Growth % $\Delta V/V$</u>
XA8G	1	2-5	3.0	1.8	
XA8G	1	42-45	6.2		2.5
XA8G	2	2-5	3.0	1.8	
XA8G	2	10-13	7.5		2.3
XA8G	2	14-17	8.9		2.8
XA8G	2	31-34	8.6		0.4
XA8G	2	35-38	7.4		3.9
XA8G	2	43-46	4.3		2.2
XA20G	4	2-5	0.6	0.4	
XA20G	4	18-21	8.5		4.3
XA20G	4	34-37	6.3		3.0
XA20G	4	42-45	2.3		1.8
XA8G	4	18-21	10.6		2.9
XA8G	4	6-9	5.8		1.7
XA8G	8	2-5	2.6	1.7	
XA8G	8	26-29	9.1		6.4
XA8G	8	30-33	8.6		3.0
XA8G	8	34-37	7.8		2.4
XA8G	8	38-41	6.3		1.8
XA8G	8	42-45	4.5		1.5

TABLE XII

Results of Post-Irradiation Blister Tests Performed on Plates from ATR
Fuel Elements

Element No.	Plate No.	Sample Location Inches from Top of Plate	Temperature (°F)	Temperature (°C)	Burnup Fiss/cc x 10 ⁻²⁰	Remarks
XA3G	2	21-28	900	482	10.5	
"	7	0-7	1050	565	6.5	
"	7	7-14	1050	565	8.5	
"	7	14-21	1050	565	9.5	
"	7	21-28	1050	565	9.5	
"	7	28-35	1050	565	9.5	
"	7	35-42	1050	565	8.0	
"	7	42-48	1050	565	6.0	
"	15	25 3/4-32 3/4	1000	537	15.0	
"	16	21-28	900	482	11.0	
XA8G	7	top dogbone	>1100	598	~3.0	} Test terminated at 1100°F without blistering
"	11	top dogbone	>1100	593	~3.0	
"	11	bottom dogbone	>1100	593	~3.0	
A172C	2	0-7	950	510	9.0	
"	2	7-14	900	482	11.0	
"	2	14-21	850	454	13.0	
"	2	21-28	850	454	13.0	
"	18	0-7	850	454	7.0	Edge of sample Center of sample
"	18	0-7	900	482	6.0	
"	18	7-14	900	482	8.0	
"	18	14-21	850	454	9.0	
"	18	21-28	750	398	9.2

Figure 23 - Blister test results.

5. Metallography

Metallography samples were taken from the fuel plates to show various behavior effects on the microconstituents. Samples were examined for fuel stability, oxide buildup, blister initiation, and pimple formation characteristics.

Figure 24 shows typical UAl_x fuel areas at intermediate and high burnup at 50X and 200X. The fuel appears to be sound at both burnup levels. The fuel was examined for cracking from excessive gas bubble agglomeration or fuel particle growth, but this was not observed.

Figure 25 shows the oxide which formed on the surface of plate 16 of XA3G at a high burnup location. It was found that the oxide is present in two layers. The inner layer is 60% of the total thickness and is cracked intermittently. The outside layer accounting for the remaining 40% of the oxide has more porosity but does not exhibit cracking. Figure 26, on the other hand, shows the oxide from plate 2, an intermediate burnup area of the element. The outside layer here is about the same thickness as in the high burnup sample but the inside layer is considerably less giving an inside-outside percentage nearer 50 to 50. The outer layer is subject to spalling. This fact is evidenced by both the presence and absence of the outside layer within the same metallographic mount as shown in Figure 27.

When blistering occurs in the UAl_x plates as a result of the post-irradiation anneal tests, (Figure 28), cracks form simultaneously throughout the core at temperatures where the aluminum matrix would have very little strength. Figure 29 at 200X shows crack initiation in the matrix after the blister test. This suggests that the pressure of the fission gas collected in the recoil zones could not be restrained by the aluminum matrix at the high temperature. Similar crack initiation sites were observed in annealed sample fuel plates which were irradiated in the G-12 loop of the ETR.

As reported in another section of the report, the blister resistance of the U_3O_8 fuel used in these startup elements was not equivalent to that of the UAl_x . Metallography of U_3O_8 blistered samples was not performed but Figure 30 shows the fuel at low and intermediate burnup. As with the laboratory samples previously tested, there is an extensive reaction with the aluminum matrix.

The small anomalies at the end of fuel plate cores called pimples were also examined by metallography. The results of this effort can be summarized by noting that "pimple material," (Figure 31) was found in a U_3O_8 fuel element fabricated by Sylcor as well as in the UAl_x elements fabricated by AI and was associated with an opening in the fuel cladding which could allow stagnant water to be entrapped

in the fuel core during irradiation. The opening in the cladding of the U_3O_8 element was caused by lack of bond which resulted in a large blister rather than a pimple. It is believed that the "pimple material" which is composed of oxides of the core materials was produced by the corrosiveness of the stagnant dissociated water (caused by radiolysis). In the UAl_x fuel plates, the opening in the cladding was associated with thin cladding over the dogbone.

As-Polished

50X

As-Polished

200X

(a) Sample from Plate 9 along the longitudinal C plane and 28-1/2 inches from the top.
(0.9×10^{21} fiss/cc)

As-Polished

50X

As-Polished

200X

Sample taken from plate 15 transverse at the C plane and 24-3/4 inches from the top.
(1.5×10^{21} fiss/cc)

Figure 24 - Intermediate (a) and high (b) burnup metallography samples from element XA3G.

Outer Layer
Inner Layer

Fuel Plate

As-Polished

500X

Figure 25 - Oxide from high burnup location from XA3G.

Outer Layer
Inner Layer

Fuel Plate

As-Polished

500X

Figure 26 - Oxide from intermediate burnup location from XA3G.

Inner Layer Oxide

Fuel Plate

As-Polished

500X

Spalled Area - Outer Layer Missing

Outside Layer
Inner Layer

Fuel Plate

As-Polished

500X

Outside Layer Intact

Figure 27 - Oxide at an intermediate burnup area from element A172C.

As-Polished

(a) 10" from top of plate 7 after 1050°F.

50X

As-Polished

(b) 24 3/4" from top of plate 15 after 1000°F

50X

Figure 28 - Metallography of UA1x plates (a) intermediate burnup and (b) high burnup after blister testing.

As-Polished

200X

Figure 29 - Matrix crack initiation.

As-Polished

Low Burnup

200X

As-Polished

Intermediate Burnup

200X

Figure 30 - U_3O_8 fuel from element A172C showing a reaction with the matrix.

As-Polished

200X

(a) Material found in the core end of plate 17 element A96C (U_3O_8).

As-Polished

200X

(b) Material found in the core end of plate 11 element XA8G ($UA1_x$).

Figure 31 - Photomicrographs showing the similarity between the pimple materials produced in (a) a U_3O_8 fuel plate compared to (b) a $UA1_x$ fuel plate.

IV. DISCUSSION AND CONCLUSIONS

The experimental data which served for a comparison basis in this performance evaluation have been developed over a number of years as part of the Reactor Fuels and Materials Development Program. One of the purposes of this program has been to extend the fuel cycle life and increase the power in test reactor fuels. The culmination of one phase of this work would be to reach the point where the ATR fuel cycle is reactivity limited rather than materials-performance limited. To move in this direction, it was necessary to characterize the behavior of the materials in the hostile environment of the reactor core and to anticipate failure mechanisms. Test methods, analytical methods and examination methods were developed to monitor behavior and evaluate performance.

It was found that fuel core growth rates under normal reactor conditions correlated with fission yield volumetric changes, if the fission products were considered to be in the solid form and growth is accommodated in fabrication-produced voids. In order to monitor this effect, immersion density techniques were developed. Growth rate vs fission density curves for characteristic fuels were produced.

It was found that the fuel plate growth occurs only in the thickness direction up to the levels tested. This is due to the fact that there is sufficient cladding to restrain growth along the plane of the plate and that a fission product matrix reaction provides plasticity and a fission rate controlled creep movement within the fuel core. The plastic behavior also reduced the tendency for matrix cracking and allows the growth to be accommodated internally in the fabrication voids. These effects have been monitored by external dimensioning and metallography.

Calculations have shown that if certain of the fission products, Kr and Xe, were to be agglomerated and behave as a gas their volumes and or pressures would be difficult to contain within the fuel cores. These materials are controllable because they are dispersed. The gaseous atoms may remain interstitial or collect in a defect lattice or be restrained by Van der Waal or surface tension forces. The effects of gases have been monitored by blister testing, metallography, and electron microscopy. As a result, blister resistance curves (blister formation as function of temperature and fission density) have been produced.

Corrosion of the fuel plate surfaces results in an aluminum oxide formation which is primarily boehmite. This oxide acts as a heat transfer barrier and causes the fuel plate temperature to rise as it builds up during the cycle. The increase in temperature is accompanied by a decrease in cladding strength. If clad strength should be lowered sufficiently, growth and thermal stress could cause plate buckling. This temperature rise due to oxide buildup has been monitored and characterized by the use of instrumented plates in the ETR G-12 loop.

The solutions to many of the developmental problems have yielded to a form of metallurgical analysis which recognizes that: if one starts with a material of a definite composition and subjects it to a specific history (process fabrication and/or application) a characteristic material structure (macro, micro, or submicro) can be produced which will have a corresponding mechanical behavior and a predictable performance.

The data correlations which have led to the present understanding of the ATR fuel behavior were generated by studying well characterized sample fuel plates which were produced to establish those compositions, process and fabrication histories, structures and properties necessary to double the original design life of the fuel. Because of this previous understanding and background information applied to the performance results of the ATR fuel elements, the following conclusions can now be made:

1. The areas of maximum flux peaking and hence maximum burnup occurred between 20 and 30 inches from the top of a given plate and at the edge of the fuel core. The plate to plate peak burnup distribution is quite different in a standard ATR element from a 7F element. The 7F elements show large changes in plate to plate burnup in the regions of plates 1-5 and 15-19; maxima generally occur at plates 5 and 15 (with 15 being the higher) and a generally uniform burnup pattern is achieved between plates 8-13. The ATR standard element showed a steadily decreasing burnup value from plates 1 through 19.
2. The fuel exhibited good dimensional stability. Essentially no plate lengthening took place. Growth or swelling which occurred during irradiation increased the plate thickness rather than the plate length. The amount of plate thickening above the maximum specified plate thickness is within the magnitude that one would expect from normal growth and corrosion phenomena, so breakaway swelling or gross deterioration mechanisms have not been initiated. There was consistency of performance from one plate to another and one element to another.
3. A comparison of the performance of this commercially produced fuel with that of sample fuel plates demonstrated that the growth accommodation of the UAL_x commercial fuel was impaired by the fact that the supplier of these particular elements used fabrication techniques which resulted in low void contents. Although growth accommodation was sufficient for stable performance to 1.5×10^{21} fissions/cc the latest fuel specifications have a requirement for voidage to accommodate additional growth and thus burnup.

Under ATR operating conditions the commercial UAl_x fuel exhibited plastic fuel behavior as did the miniature fuel plates and micro-cracking in the matrix did not occur.

The blister resistance of these commercially fabricated fuel plates was similar to that of sample fuel plates irradiated in the G-12 loop of the ETR. The higher weight percent UAl_x plates out-performed the intermediate and low weight percent UAl_x and the UAl_x out-performed the U_3O_8 fueled plates. In every case the fuel blister resistance was better than the previously established minimums.

4. The safe operating limit of the current fuel element designs had not been exceeded and there were no indications that the limits have been approached.

5. The only performance anomaly detected was the formation of "pimples" at the rate of 1 per 100 plates at the core ends (dogbone area). The pimple material is composed of oxides of the core materials and was produced by the corrosive action of stagnant water which had penetrated the (out of specification) cladding at these areas.

6. The oxide found on elements from Cores II and III frequently exhibited a double layer structure, but the total thickness observed metallographically would not cause temperature increases great enough to reduce significantly clad strength. The inner layer of oxide showed a temperature dependence of the type predicted by Griess, whereas the outer layer shows a much weaker temperature dependence.

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