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MASTER

THE EFFECT OF HIGH POWER DENSITY ON

UO₂ ROD TYPE ELEMENTS -

THE SAXTON SPECIAL 2 X 2 ASSEMBLY

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ABSTRACT

Four stainless steel clad uranium dioxide fuel elements were irradiated to a peak power density of 23 kw/ft and a peak burnup of 12,000 MWD/MTU in the Saxton Reactor. Uranium dioxide fuels of 97-percent and 78-percent initial theoretical density were represented in this experiment. Post-irradiation destructive examination of the fuel elements included fission gas analysis, cesium-137 nuclide burnup analysis, and microstructural examination of the fuel. The performance of these high power density fuel elements was entirely satisfactory.

1.0 INTRODUCTION

An engineering proof test was performed in the Saxton Reactor to investigate the performance of uranium dioxide fuel operated at high linear power densities.* It was initially planned to operate the fuel elements at power densities of about 26-28 kw/ft to achieve center melting of the fuel. However, the highest power density achieved was 23 kw/ft and fuel center melting did not occur.

Four fuel elements were incorporated into a modified Saxton removable 2 x 2 assembly. The assembly was inserted during December 1963 and was irradiated until July 1966 to a peak burnup of 12,000 MWD/MTU. Post-irradiation examinations were performed at the Westinghouse Post-Irradiation Facility Hot Cells at Waltz Mill, Pennsylvania.

*This experiment was carried out with the co-operation of the Saxton Nuclear Experimental Corporation (SNEC).

2.0 DESIGN AND FABRICATION

The primary objective of this test was evaluation of the performance of UO_2 fuel irradiated at high power densities. Thus, in establishing the design of the 2 x 2 assembly, the main consideration was to fabricate fuel elements which would produce the maximum linear heat ratings achievable in the Saxton Reactor, commensurate with thermal and hydraulic limitations. This was accomplished by increasing the fuel pellet diameter by 40 percent, from 0.390 inches to 0.540 inches, and increasing the fuel enrichment from 5.4 to 8.3 w/o U^{235} . This permitted the fuel rods, when inserted in a peripheral core position, to operate at a peak linear heat rating of between 1.1 and 1.5 times the peak in the rest of the core with a surface heat flux not exceeding that of the normal Saxton fuel elements. Movement of this assembly to the central core position resulted in operation at between 1.0 and 2.4 times the peak kw/ft in the remainder of the core. The assembly was fabricated, according to Westinghouse drawing 647J1111 (Figure 1), with the aid of the fabrication flow diagram shown in Figure 2. The enclosure assembly and all structural components were fabricated from cold worked AISI Type 304 stainless steel. The individual fuel elements were fabricated according to Westinghouse drawing 881D819 (Figure 3).

The use of Zircaloy or annealed stainless steel rather than cold worked stainless steel as the cladding material was investigated. Calculations indicated a minimum wall thickness of 38 mils was required for either of these alternate materials versus the 23.5-mil thickness needed for cold worked stainless steel. The high design values of maximum heat flux proposed for this fuel assembly precluded the use of thick cladding materials because of the large temperature rise which would be generated across a 38-mil-thick cladding wall.

The fabrication parameters for the fuel elements are summarized in Table 1. Three of the fuel elements, two non-removable and one removable, contained UO_2 pellets of 97-percent theoretical density (T.D.). The fourth element, Rod 8B, contained 25 pellets of 97-percent T.D., and 17 pellets with 500 ppm boron in the form of boron carbide. The presence of this burnable poison in the fuel limited the maximum density achievable to 78-percent T.D. These low density pellets were loaded into the center of the fuel element where the peak neutron flux would exist. The quantity of boron in the fuel had an

Figure 1 - Saxton Reactor Plant
Removable Fuel Assembly

Figure 2. Flow Diagram for the Fabrication of the Saxton Removable 2 x 2 Fuel Assembly

NOTE	TITLE
1	END PLUG
2	SPRING
3	BUSHING
4	MARKER BODY
5	PELLET STACK
6	TUBE
7	MARKER RINGS
8	END PLUG
9	END PLUG

Figure 3. Saxton Removable 2 x 2 Fuel Rods

TABLE 1
FABRICATION PARAMETERS^a

Pellet Diameter	0.540 in.
Enrichment U ²³⁵	8.4%
Density (% Theoretical)	97.4%, 78% ^b
Pellet Stack Height	33.240 in.
Cladding I.D.	0.552 in.
Cladding O.D.	0.599 in.
Fuel Element Length Non-Removable	39.060 in.
Removable	41.030 in.
Cladding Material	AISI Cold Worked Type 304 SS
Wall Thickness	0.0235 in.
Cladding-to-Fuel Diametral Gap	0.012 in.
Weight of Fuel	1320, 1230 ^b gm
Weight of U	1165, 1084 ^b gm
Weight of U ²³⁵	98, 91 ^b

a. Nominal

b. Rod 8B

insignificant effect on the heat rating (it accounted for much less than one percent of the total absorption cross section of the fuel) but was included to assess the chemical compatibility of this form of boron with uranium dioxide as part of a separate burnable poison program. Rod 8B was included in the post-irradiation program to evaluate the performance of very low density (hence very low thermal conductivity) uranium dioxide at high power densities.

A major improvement incorporated into the design of this assembly is the accessibility of individual fuel elements that can be removed remotely or replaced while the fuel assembly is in the storage rack. Thus, fuel elements were available for examination or replacement without waiting for head removals or destruction of the containing assembly. This was accomplished by fabricating the assembly so it could be separated into two major parts: 1) the fuel bearing portion, which is sized to coincide with the reactor internals, and 2) its removable supporting structure.

The fuel bearing portion of the assembly consists of the conventional spring type grids, perforated enclosure and bottom end-plate. The as-fabricated fuel bearing portion of the 2 x 2 assembly is shown in Figure 4. The top end plate was made with extended posts and holes through which the removable fuel element end plugs protrude. The upper portion of each assembly can be removed from the fuel assembly, thus exposing the removable fuel element end plugs to a removal tool (Figure 5). A tool was also designed (Figure 6) to permit latching and unlatching of the assembly from the removable supporting structure.

Figure 4. Fuel Bearing Portion of the Saxton Removable 2 x 2 Fuel Assembly

SAXTON SPECIAL REMOVABLE 2x2 ASSEMBLY

SAXTON SPECIAL REMOVABLE ASSEMBLY UNLATCHING TOOL

3.0 REACTOR OPERATING CONDITIONS

The 2 x 2 high power density assembly was irradiated as part of the Saxton core for a total of 396 days. The reactor operating history is shown graphically in Figure 7 and is summarized in Tables 2 and 3. During this time the average thermal output of the core was 20.6 MW and the peak output was 23.2 MW. Significant thermal and hydraulic design parameters are given in Table 4.

TABLE 2

IRRADIATION HISTORY OF THE SAXTON REMOVABLE 2 X 2 FUEL ASSEMBLIES

<u>Date</u>	<u>Transfer</u>
December 19, 1963	Loaded in core position N-3 ^a
May 19, 1964	Removed from N-3. Placed in storage rack
July 6, 1964	Loaded in core position N-3.
May 19, 1965	Removed from N-3. Placed in storage rack
June 1, 1965	Loaded in core position N-1
July 6, 1965	Removed from N-1. Removed Rod 7. Loaded in core position N-1
September 8, 1965	Removed from N-1. Placed in storage rack
September 9, 1965	Inserted Rod 8B
November 22, 1965	Loaded in core position N-3
April 6, 1966	Removed from N-3. Loaded in core position N-1
July 21, 1966	Removed from core

a. N-1 and N-3 signify the central and peripheral core locations, respectively.

TABLE 3
SUMMARY OF IRRADIATION HISTORY

Fuel Rod 4

<u>Core Position</u>	<u>Time at Power (Days)</u>
N-3	181
N-1	47
N-3	86
N-1	<u>82</u>
Total	396

Fuel Rod 8B

<u>Core Position</u>	<u>Time at Power (Days)</u>
N-3	181
N-1	16
N-3	86
N-1	<u>82</u>
Total	365

Fuel Rod 7

<u>Core Position</u>	<u>Time at Power (Days)</u>
N-3	181
N-1	16

TABLE 4

SIGNIFICANT THERMAL AND HYDRAULIC DESIGN PARAMETERS FOR THE
SAXTON REMOVABLE 2 X 2 ASSEMBLY

Core Power (MWth)	20.0
Inlet Temperature (°F)	520
Maximum Thermal Output (kw/ft)	26.5
Maximum Surface Heat Flux (BTu/hr-ft ²)	577,600
Hot Channel Factors	
F_q - based on core average kw/ft	6.44
$F_{\Delta H}$	4.56
Minimum DNB Ratios	
100% Power - 2000 psia	q'' - DNBR 2.59
1200% Power - 1800 psia	q'' - DNBR 1.86
Center Fuel Temperature	Molten

4.0 POST-IRRADIATION EXAMINATION

Three of the four elements in the 2 x 2 assembly were destructively examined at the Waltz Mill Post-Irradiation facilities. They were: 1) Rod 7, after irradiation for a short period in the central core position N-1, and 2) fuel Rods 8B and 4 at the termination of the planned irradiation period. Element 3 was not examined since it was identical to Rod 4.

4.1 Visual

A Bausch and Lomb stereomicroscope, with a range of magnification from 1X to 30X, was used to conduct the visual examinations. No unusual findings or evidence of excessive fretting erosion was observed at the areas of contact with the stainless steel spring clip fingers. The only surface markings were scratches caused by removing the rods from the subassembly and other subsequent handling operations.

4.2 Dimensional

A remote spiral trace profilometer was developed for the purpose of obtaining very accurate in-cell measurements on irradiated fuel elements. The use of this tool on fuel Rods 4 and 8B resulted in an increase in precision for diametric measurements from plus or minus 0.5 mil possible with a micrometer to plus or minus 0.2 mil. The results of these diameter measurements are shown in Figures 8 and 9. On the basis of these data, no conclusion could be made concerning changes from as-fabricated dimensions (which were accurate to plus or minus 0.5 mil).

4.3 Gamma Activity

Based on past experience with radial and axial flux distributions of the Saxton Reactor, it was decided that gamma scans of the high power density fuel elements were not necessary. In addition, a gamma scan was taken on a fuel element* which was irradiated in the 2 x 2 assembly during the time interval when

*The 2 x 2 assembly was utilized for a short time as a vehicle for other experimental fuel elements.

Figure 8. Diameter Measurements of Rod 4

Figure 9. Diameter Measurements of Rod 8B

Rod 8B was removed. From this scan, a peak-to-average flux ratio of 1.27 was calculated (Appendix A). This value is in agreement with past experience and was felt sufficiently pertinent to be applied to the other fuel elements irradiated in this assembly.

4.4 Axial Expansion Measurements

An axial expansion marker (Figure 10), designed to reveal the extremes of fuel column movement during irradiation, was incorporated into the high power density fuel elements. However, due to a fabrication problem, it was necessary to reopen the fuel elements for the purpose of making fuel column adjustments. During these adjustments and subsequent resealing of the fuel elements, the axial expansion markers were improperly aligned. Thus, no expansion data were obtained.

4.5 Burnup and Power Density Determinations

The burnup and power outputs achieved by fuel Rods 4 and 8B during the course of irradiation were estimated from analyses of the cesium-137 fission-product nuclide. The burnups were calculated using the following values:

1. Cesium-137 half life of 29.4 years.
2. Cesium-137 fission yield from all fissionable species (U^{235} , U^{238} , Pu^{239} , Pu^{241}) of 6.3 percent.
3. Energy release per fission (U^{235}) of 204 Mev.

Linear power densities were estimated on the basis of the known irradiation history for the individual fuel elements. Independent calculations of the power densities achieved by these fuel elements were made based upon flux wire data. The results of the burnup analyses are given in Table 5. No average burnup value for fuel Rod 8B is given because this element contained fuel of two different densities and the peak-to-average ratio calculated in Appendix A cannot be applied.

4.6 Fission Gas Release

Fission gases were collected after irradiation for mass spectrographic analysis. The total gas volume contained within each rod was determined within ten

Figure 10. Axial Expansion Marker

TABLE 5
SUMMARY OF BURNUP AND LINEAR POWER DENSITIES

<u>Fuel Rod No.</u>	<u>Peak Burnup (MWD/MTU)</u>	<u>Average Burnup (MWD/MTU)</u>	<u>Peak Linear Power Density (kw/ft)</u>	
			<u>Cesium Analysis</u>	<u>Flux Wire Data</u>
4	12,000	9,500	23.3	22.8
7 ^a	4,500	--	--	--
8B	10,000	b	--	17.2

a. No burnup samples taken.

b. Average burnup not meaningful due to different fuel densities contained in this element.

percent with the system volume known to an accuracy of plus or minus one cc. A summary of the gases collected, their isotopic distribution, and percent release is given in Table 6.

There are several apparent anomalies in the fission gas values. The argon content of these elements is due to the argon welding atmosphere utilized in the fabrication of these rods. The higher argon content of fuel Rod 8B is due to the low density, hence, a large open-porosity fraction in the fuel. The lower values of Kr and Xe in Rod 7 are due to the lower burnup experienced by this element. A rough estimate of peak burnup in Rod 7 is 4500 MWD/MTU which is less than 1/2 the burnup experienced by Rods 4 and 8B. The high fission gas content of Rod 8B is due to the lower fuel density. Xe and Kr isotopic percentage agree well among the fuel elements.

Prior to fabrication, several pellets were analyzed for H_2 and H_2O content and the results indicated that the total H_2 content was less than 10 ppm. Assuming all of this H_2 was released, this would produce only a fraction of what is found in Rod 7. Therefore, the high hydrogen content of this element has two possible explanations:

1. the moisture content of the UO_2 sample was not representative of the fuel in Rod 7, and
2. the UO_2 adsorbed moisture during a fabrication step after the samples were chosen.

4.7 Fuel Metallography

Samples of a complete cross section were removed from each of the fuel elements in the region of axial peak power.* Low magnification macrophotographs were prepared from these cross sections. The micrographs of fuel Rods 4 and 8B are in the as-polished condition and Rod 7 is in the etched condition as shown in Figures 11, 12 and 13 respectively.

*As indicated by the gamma scan in Appendix A.

TABLE 6
FISSION GAS ANALYSES

<u>Fuel Rod No.</u>	<u>4</u>	<u>7</u>	<u>8B</u>
Total Volume (cc at STP)	90.2	34.2	115.5
H ₂ (cc at STP)	0.09	3.87	0.07
He	3.29	0.73	10.72
N ₂	0.23	0.04	0.45
O ₂	0.09	0.01	0.16
A	12.87	13.73	21.71
Kr	10.14	2.25	12.09
Xe	63.48	13.56	70.29
Kr ⁸² (Volume %)	0.04	--	--
Kr ⁸³	13.94	14.30	13.97
Kr ⁸⁴	27.33	27.11	27.27
Kr ⁸⁵	6.81	6.84	6.83
Kr ⁸⁶	51.88	51.74	51.93
Xe ¹²⁸ (Volume %)	0.01	--	0.01
Xe ¹³⁰	0.03	--	0.03
Xe ¹³¹	11.82	12.19	11.75
Xe ¹³²	18.95	19.02	18.68
Xe ¹³⁴	33.04	33.79	32.83
Xe ¹³⁶	36.15	35.00	36.70

TABLE 6 (Continued)

Summary of Fission Gas Yields^a

<u>Element No.</u>	<u>4</u>	<u>7</u>	<u>8B</u>
% Fission Yields Isotope			
Kr ⁸³	0.545	0.566	0.545
Kr ⁸⁴	1.06	1.33	1.06
Kr ⁸⁵	0.266	0.271	0.266
Kr ⁸⁶	2.02	2.02	2.02
Xe ¹³¹	2.89	3.61	2.88
Xe ¹³²	4.63	4.54	4.60
Xe ¹³⁴	8.06	8.06	8.06
Xe ¹³⁶	8.82	10.36	9.01
% Xe ¹³⁵ -to-Xe ¹³⁶ Conversion			
	38.0	62.0	41.0
% Kr and Xe Fission Yield ^b			
Xe	22.0	22.7	22.0
Kr	3.89	4.19	3.89
% Fission Gas Release			
Xe	24.0	--	~30.0
Kr	24.0	--	~30.0

a. Normalized to Kr⁸⁶ = 2.02%^[1]

b. Corrected for Xe¹³⁵-to-Xe¹³⁶ conversion

Figure 11 - Photomicrograph of
Fuel Element #4

PHOTOMACROGRAPH
ETCHED CONDITION

CLAD

SMALL EQUIAXED GRAIN

EQUIAXED GRAIN
GROWTH

COLUMN GRAIN GROWTH

CENTRAL VOID

100X

Figure 12 - Photomicrograph of
Fuel Element #7

4X

PHOTOMACROGRAPH
AS POLISHED CONDITION

CLAD

SMALL
EQUIAXED
GRAIN

EQUIXED
GRAIN
GROWTH

COLUMN GRAIN-GROWTH

CENTRAL VOID

100X

Figure 13 - Photomicrograph of
Fuel Element #88

A general examination of the strip microphotographs illustrate the following structural features:

1. A region of fine equiaxed UO_2 grains indicative of the pre-irradiated structure;
2. A region of equiaxed grain growth;
3. A region of columnar grain growth; and
4. A central void.

Even though fuel Rod 8B operated at a lower power rating than Rod 4, its microstructure indicates more severe thermal conditions. The great difference in overall structure between Rods 4 and 7 and Rod 8B is due to the different initial fuel densities represented by these elements. The low density has a two-fold effect on the irradiated fuel structure. First, at these power ratings the low fuel density results in the formation of a significant central void. Secondly, the low fuel density decreases the thermal conductivity, resulting in a steeper temperature gradient than in fuel Rod 4. The central void in Rod 8B accounts for approximately six percent of the total fuel pellet volume resulting in an average fuel density after irradiation of 84-percent T.D. This density was assumed for the center temperature calculations. Thermal conductivity was adjusted for porosity by the use of the Maxwell-Eucken expression.*

$$*K_P = K_{100} \frac{(1-P)}{1 + \beta P}$$

where

P = porosity fraction

K_P = thermal conductivity of porous fuel

β = an empirical constant related to the pore shape^[2]

Radii corresponding to various microstructural features such as equiaxed grain growth, columnar grain growth, and central void were measured to establish temperature profiles. The UO_2 fuel center temperatures were estimated using the $\int K d\theta$ technique.^[3] The expression utilized to calculate the temperature within the fuel was:

$$\int_{T=0}^{T_C} K(\theta) d\theta = \int_{T=0}^{T_S} K(\theta) d\theta + \int_{T_S}^{T_C} K(\theta) d\theta$$

where

T_S = temperature of fuel surface

T_C = temperature of fuel center

Table 7 summarizes the results of the evaluation of the fuel microstructures.

4.8 Cladding Examination

Detailed cladding examination was not performed since the conditions experienced by the stainless steel cladding for this experiment did not differ from standard Saxton rods. The performance of stainless steel cladding under equivalent and more severe conditions has been previously evaluated in other reports.^[4,5]

TABLE 7
SUMMARY OF FUEL MICROSTRUCTURE EXAMINATION

<u>Fuel Rod No.</u>	<u>Estimated Maximum Power (kw/ft)</u>	<u>Equiaxed Grain Growth (r/a)</u>	<u>Columnar Grain Growth (r/a)</u>	<u>Central Void (r/a)</u>	<u>Estimated Center Temperature</u>
4	23.0	0.78	0.68	0.07	2340
7	17.2	0.74	0.63	--	1925
8B	18.0	0.92	0.85	0.26	2200

5.0 CONCLUSIONS

The performance of uranium dioxide fuel operated at peak power densities up to 23 kw/ft and burnups to 12,000 MWD/MTU was satisfactory. The low density fuel achieved a burnup of about 10,000 MWD/MTU at a peak power density of 17.2 kw/ft. In addition to the lower thermal conductivity exhibited by the low density fuel, a much higher fission gas release was observed.

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APPENDIX A

Fuel Rod E-4 was irradiated in the 2 x 2 assembly in the location vacated by fuel Rod 7 as part of a burnable poison engineering test. The fuel was vibratory compacted six w/o U²³⁵ enriched and contained 500 ppm natural boron carbide. It was irradiated for a total of 199 days, 113 of which were in the central core location. During the post-irradiation examination of this fuel element, a single channel analyzer scan for gamma radiation of energies greater than 0.5 Mev was taken. This scan is illustrated in Figure A-1. A peak-to-average ratio of 1.27 was calculated from this scan.

The shape of the gamma scan is not believed to be significantly affected by the burnable poison which was initially uniformly distributed. Furthermore, the post-irradiation examination of the fuel showed that no axial migration of the boron had occurred.

Figure A-1. Gamma Scan of Element E-4