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EXAMINATION OF AN IRRADIATED PROTOTYPE
FUEL ELEMENT FOR THE ELK RIVER REACTOR

by

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EXAMINATION OF AN IRRADIATED PROTOTYPE FUEL ELEMENT FOR THE ELK RIVER REACTOR

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ABSTRACT

An examination was made of a prototype Elk River Reactor fuel element irradiated in the EBWR to a maximum burn-up of 1000 MWD/T. The assembly, which consisted of an array of Type 304 stainless steel tubes fueled with pellets of 97 w/o ThO_2 -3 w/o UO_2 , showed insignificant dimensional changes. Irregular vertical packing of the fuel pellets gave a "banded" appearance to the fuel rods, due to preferential oxide deposition. The brazing material used in the center strap of the assembly showed noticeable corrosion effects. The fuel pellets showed no thermal effects and exhibited little cracking. Their dimensional stability was excellent. The fuel exhibited good gas-retention characteristics.

INTRODUCTION

An investigation was undertaken to evaluate the performance characteristics of a prototype fuel element supplied by the Allis-Chalmers Manufacturing Company for the first core loading of the Elk River Reactor at Elk River, Minnesota. The irradiation program was carried out in the EBWR.

The fuel element assembly consisted of 25 fuel rods in a five-by-five array. Each rod contained ThO_2 - UO_2 fuel in an AISI Type 304 stainless steel tube with an inside diameter of 0.4125 in. + 0.0015 in. - 0.0000 in., a wall thickness of 0.0200 in. $\pm$ 0.0025 in., and an outside diameter of 0.4525 in. The fuel was a mixture of 97 w/o ThO_2 -3 w/o UO_2 compressed and sintered to 94% of theoretical density in the form of pellets, 0.4075 $\pm$ 0.002 in. in diameter by $\frac{1}{2}$ in. long. The uranium was fully enriched. The annulus between the pellets and the tube wall was filled with helium at one atmosphere pressure. The rods were welded closed at each end, leaving a 1-in. long gas space. The rods were held rigidly at the midplane of the element by stainless steel straps, brazed to each rod and spot welded to each other. They were held semirigidly in the end fittings by compression springs at the ends of each rod. The prototype element had EBWR-type end fittings so that it could be inserted into the core of that reactor. The fuel element assembly is shown before irradiation in Fig. 1.

Fig. 1. The Elk River reactor fuel element modified for irradiation in EBWR.

PREIRRADIATION EXAMINATION

A preirradiation examination of the fuel assembly before its insertion into the EBWR was made at the reactor site. The four faces of the assembly were coded to make later identification possible. The basis for the code was the welded lap seam in the top end fitting. The face on which the seam was visible was designated as number "1," with the other faces being numbered clockwise from this position when viewing the element from the top. The corner rods were numbered with two-digit numbers with the digits designating the faces intersecting at the rod. Hence, the corner rods were numbered 12, 23, 34, and 41. Other rods in the face were numbered from the low-numbered corner to the high-numbered corner, with the first digit being the number of the face.

The examination consisted of the following measurements:

1. the width of each face as measured from corner rod to corner rod across a face at seven positions along the length of the assembly;
2. the diameter of the four corner rods measured every 6 in. along the length of the rod; and

3. a qualitative measurement of the squareness of the rod bundle at the center strap as the assembly lay flat on each face.

The positions at which the width and diameter measurements were made are shown in Fig. 2.

Fig. 2. Positions at which assembly widths and rod diameters were measured

The squareness of the rod bundle was measured by determining the horizontal shift from the vertical assumed by any face as measured against the corner rods in that face. A vertical surface was placed against the corner rods, and the horizontal distance between the vertical surface and the deviating corner rod was measured with a centimeter scale.

Rods 12 and 23 were bowed inward toward the central axis of the element. The bowing was generally midway between the center strap and the end fittings, and was probably caused by the metal straps used to hold the element on its shipping pallet. In addition, rods 35, 36, 37, and one uncoded internal rod were bowed.

IRRADIATION HISTORY

The assembly was inserted into the EBWR on December 11, 1958, in core position 72-43, with face number four toward the center of the reactor. On April 6, 1959, it was transferred to position 23-82. The fuel element orientation in this position was not known. Based on available flux plots for the core, it was estimated that the average unperturbed thermal neutron flux was $4.6 \times 10^{12} \text{ n}/(\text{cm}^2)(\text{sec})$ and a maximum of $8.0 \times 10^{12} \text{ n}/(\text{cm}^2)(\text{sec})$. The maximum cladding surface heat flux was estimated at $85,000 \text{ BTU}/(\text{hr})(\text{ft}^2)$. The integral $\int_{T_s}^{T_0} k dT$, evaluated at the position of maximum heat flux, was 7.7 watts/cm , where T_s is the temperature of the fuel surface and T_0 that at the center of the fuel. The assembly was removed from the core and placed in the EBWR storage pit on

July 31, 1959. The element was in the core for an irradiation period of 90 days, during which time the reactor was at an average power of 20 Mw.

POSTIRRADIATION EXAMINATION

A preliminary examination was made under water in the EBWR reactor vessel on February 11, 1959. A film deposit was noted, but no mechanical distortions were observed.

The element was removed from the EBWR storage pit and placed in the Argonne hot cell facilities for detailed postirradiation examination. Special handling of the element in the hot cell was used to avoid damage that might be misconstrued as being caused by irradiation. A special pallet was constructed and used to move the element in the hot cell.

Visual Observations

Visual inspection indicated that, of the visible rods, only those which were bowed before irradiation were bowed after irradiation. Figure 3 shows a view of face number two, with bowed rods in the lower half of the assembly.

Fig. 3.

Fuel element after irradiation in the EBWR. Outer rods in the foreground were bowed prior to irradiation.

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Further examination showed that one rod had a shallow $\frac{1}{8}$ -in. diameter dent about 8 in. from the top of the rod. The dent is of undetermined

origin. Another rod, number 36, was seen to have an irregular longitudinal line about the length of a single pellet. This line is shown in Fig. 4 as it was first seen. Upon closer examination, the line proved to be a surface imperfection in the stainless tube. An enlarged view of the imperfection is shown in Fig. 5.

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Fig. 4. Longitudinal surface imperfection in rod No. 36. Transverse lines are boundaries of oxide "band" outlining a pellet.

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Fig. 5. Enlarged view of surface imperfection in rod No. 36.

The center straps were fully intact after irradiation, although it appeared that excess brazing material had been leached out from between crevices in the strap, as shown in Fig. 6. The brazing material was

Nicrobraz No. 50. The end closure welds were not visibly affected during irradiation.

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Fig. 6. Area of center strap showing loss of exposed brazing material in crevice.

Considerable "banding" of the rods was evident, as shown in Fig. 7. These bands were apparently caused by the preferential deposition of oxide film at areas of highest heat flux. Some spaces between bands were as wide as $\frac{1}{4}$ in. and are believed to be caused by the absence of tight vertical packing of the fuel pellets, thereby leaving gaps filled with helium. The presence of these gaps was verified by preirradiation X-ray photographs supplied by the manufacturer. The bands were predominantly on the lower half of the assembly, which was in a region of higher flux during irradiation. The element had an overall pale rust color except at the end fittings, which tended to have an intermingling of bluish hues. The film formation was light, however, and no flaking was observed.

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Fig. 7. Lower half of assembly showing "banding" on rods of face No. 4.

Table I

PRE AND POSTIRRADIATION MEASUREMENTS OF FUEL ASSEMBLY WIDTH

Position ^a	Preirradiation			Postirradiation			Change (in.)
	Reading 1 (in.)	Reading 2 (in.)	Average (in.)	Reading 1 (in.)	Reading 2 (in.)	Average (in.)	
1AA	3.5432	3.5425	3.5428	3.5340	3.5342	3.5341	-0.0087
1A	3.4394	3.4418	3.4406	3.4350	3.4351	3.4350	-0.0056
1B	3.4371	3.4285	3.4328	3.4086	3.4084	3.4085	-0.0243
1C	3.4599	3.4621	3.4610	3.4399	3.4511	3.4455	-0.0155
1D	3.3754	3.3790	3.3772	3.3659	3.3472	3.3566	-0.0206
1E	3.4390	3.4393	3.4391	3.4358	3.4347	3.4352	-0.0039
1EE	3.5080	3.5073	3.5076	3.4960	3.4988	3.4974	-0.0102
2AA	3.5094	3.5088	3.5091	3.5056	3.5080	3.5068	-0.0023
2A	3.4461	3.4478	3.4470	3.4391	3.4449	3.4420	-0.0050
2B	3.4285	3.4288	3.4286	3.4041	3.3972	3.4006	-0.0280
2C	3.4723	3.4708	3.4716	3.4872	3.4693	3.4782	+0.0066
2D	3.3041	3.2998	3.3020	3.2941	3.2908	3.2924	-0.0096
2E	3.4198	3.4201	3.4199	3.4190	3.4216	3.4203	+0.0004
2EE	3.5329	3.5370	3.5350	3.5297	3.5270	3.5284	-0.0066
3AA	3.5460	3.5427	3.5444	3.5211	3.5287	3.5249	-0.0195
3A	3.4438	3.4441	3.4439	3.4571	3.4511	3.4541	+0.0102
3B	3.4531	3.4515	3.4523	3.4392	3.4332	3.4362	-0.0161
3C	3.4635	3.4664	3.4650	3.4617	3.4631	3.4624	-0.0026
3D	3.4039	3.4029	3.4034	3.3669	3.3575	3.3622	-0.0412
3E	3.4398	3.4430	3.4414	3.4520	3.4526	3.4523	+0.0109
3EE	3.5073	3.5055	3.5064	3.5100	3.5096	3.5098	+0.0034
4AA	3.5125	3.5109	3.5114	3.5213	3.5221	3.5217	+0.0103
4A	3.4395	3.4384	3.4390	3.4375	3.4350	3.4362	-0.0028
4B	3.4042	3.4085	3.4064	3.3858	3.3806	3.3832	-0.0232
4C	3.4471	3.4477	3.4474	3.4440	3.4459	3.4450	-0.0024
4D	3.3735	3.3745	3.3740	3.3680	3.3761	3.3720	-0.0020
4E	3.4295	3.4298	3.4296	3.4257	3.4270	3.4264	-0.0032
4EE	3.5305	3.5342	3.5324	3.5191	3.5194	3.5192	-0.0132

^aThe first number designates the face. See Fig. 2 for measurement locations.

Fig. 9

Micrometer and mechanical ringstand arrangement used to measure corner rod diameter.

Physical Measurements

Measurements of the assembly width at corresponding position to the preirradiation measurements were made using the micrometer and mechanical ringstand arrangement

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Fig. 8. Micrometer and mechanical ringstand arrangement used to measure width of assembly were made.

shown in Fig. 8. The pre and postirradiation measurements are compared in Table I. The average variation in the pair of postirradiation readings from 28 positions was 0.0046 in. in 3.5 in., whereas for the preirradiation readings it was 0.0020 in. The rather large variation in the postirradiation results is attributed to the inherent difficulties under which these measurements were made and to the uncertainty of the readings from the position midway between the end welds and the center strap (positions B and D indicated in Fig. 2). At these positions the unsupported rods tended to bow inward under pressure from the micrometer. A minimum difference of 0.005 in. from the preirradiation measurements was considered to be a significant change. The micrometer was checked for accuracy before and after the series of measurements

The diameters of each of the four corner rods were measured using a micrometer 45° from the horizontal in the arrangement shown in Fig. 9. A comparison of the pre and postirradiation measurements of diameter is given in Table II. The average variation in the pairs of preirradiation readings was 0.0002 in., whereas it was 0.0014 in. in the postirradiation readings. A change of 0.002 in. from the preirradiation measurement was considered significant. The rather large number of negative changes in the rod diameters may be due to an elliptical shape of the rods. Two sets of readings taken 90° apart of rod number 34 give credence to this possibility.

A determination of the squareness of the assembly was made in the same manner as the preirradiation measurement. A comparison of the pre and postirradiation results is given in Table III. The changes were not considered significant.

Table II

PRE AND POSTIRRADIATION MEASUREMENTS OF CORNER ROD DIAMETERS

Position ^a	Preirradiation			Postirradiation			Change (in.)
	Reading 1 (in.)	Reading 2 (in.)	Average (in.)	Reading 1 (in.)	Reading 2 (in.)	Average (in.)	
Rod 12 a	0.4531	0.4530	0.4530	0.4589	0.4584	0.4586	+0.0056
b	0.4521	0.4520	0.4520	0.4545	0.4511	0.4528	+0.0008
c	0.4519	0.4518	0.4518	0.4508	0.4500	0.4504	-0.0014
d	0.4517	0.4515	0.4516	0.4501	0.4541	0.4521	+0.0005
e	0.4513	0.4521	0.4517	0.4508	0.4508	0.4508	-0.0009
f	0.4516	0.4525	0.4520	0.4513	0.4511	0.4512	-0.0008
g	0.4520	0.4521	0.4520	0.4543	0.4533	0.4538	+0.0018
h	0.4525	0.4525	0.4525	0.4598	0.4630	0.4614	+0.0089
j	0.4518	0.4518	0.4518	0.4541	0.4512	0.4526	+0.0008
k	0.4518	0.4523	0.4520	0.4528	0.4513	0.4520	0.0000
Rod 23 a	0.4515	0.4515	0.4515	0.4503	0.4500	0.4502	-0.0013
b	0.4515	0.4514	0.4514	0.4491	0.4490	0.4490	-0.0024
c	0.4513	0.4514	0.4514	0.4489	0.4490	0.4490	-0.0024
d	0.4513	0.4513	0.4513	0.4480	0.4493	0.4486	-0.0027
e	0.4512	0.4511	0.4512	0.4488	0.4490	0.4489	-0.0023
f	0.4513	0.4515	0.4514	0.4516	0.4518	0.4517	+0.0003
g	0.4518	0.4519	0.4518	0.4490	0.4482	0.4486	-0.0032
h	0.4518	0.4519	0.4518	0.4483	0.4485	0.4484	-0.0034
j	0.4518	0.4515	0.4516	0.4520	0.4524	0.4522	+0.0006
k	0.4530	0.4527	0.4528	0.4493	0.4511	0.4502	-0.0026
Rod 34 a	0.4545	0.4546	0.4546	0.4525	0.4561	0.4543	-0.0003
b	0.4518	0.4520	0.4519	0.4496	0.4503	0.4500	+0.0019
c	0.4524	0.4521	0.4522	0.4495	0.4507	0.4501	-0.0021
d	0.4519	0.4520	0.4520	0.4493	0.4490	0.4492	-0.0028
e	0.4524	0.4517	0.4520	0.4491	0.4494	0.4492	-0.0028
f	0.4516	0.4524	0.4520	0.4524	0.4499	0.4512	-0.0008
g	0.4519	0.4515	0.4517	0.4500	0.4510	0.4505	-0.0012
h	0.4521	0.4523	0.4522	0.4501	0.4525	0.4513	-0.0009
j	0.4518	0.4519	0.4518	0.4553	0.4595	0.4574	+0.0056
k	0.4519	0.4519	0.4519	0.4560	0.4552	0.4556	+0.0037
Rod 41 a	0.4535	0.4529	0.4532	0.4535	0.4538	0.4536	+0.0004
b	0.4516	0.4515	0.4516	0.4501	0.4513	0.4507	-0.0009
c	0.4516	0.4515	0.4516	0.4541	0.4514	0.4528	+0.0012
d	0.4514	0.4514	0.4514	0.4503	0.4493	0.4498	-0.0016
e	0.4514	0.4515	0.4514	0.4516	0.4493	0.4504	-0.0010
f	0.4516	0.4518	0.4517	0.4508	0.4505	0.4506	-0.0011
g	0.4515	0.4518	0.4516	0.4490	0.4519	0.4504	-0.0012
h	0.4515	0.4515	0.4515	0.4583	0.4528	0.4556	+0.0041
j	0.4515	0.4517	0.4516	0.4520	0.4524	0.4522	+0.0006
k	0.4517	0.4521	0.4519	0.4521	0.4525	0.4523	+0.0004

^aSee Fig. 2 for measurement locations.

It was possible to remove five whole pellets from rod No. 36 after removal of the rod from the assembly and sectioning it with a tubing cutter. The pellets came from an area of intermediate flux midway between the top weld closure and the center strap. Diameter measurements of these

pellets are given in Table IV. The as-fabricated diameters were 0.4075 ± 0.002 in. No significant dimensional changes were measured.

Table III

PRE AND POSTIRRADIATION MEASUREMENTS OF
FUEL ASSEMBLY SQUARENESS

Face No.	Horizontal Deviation from the Vertical (mm)	
	Preirradiation	Postirradiation
1	5	4
2	3	5
3	4	4
4	4	5

Table IV

POSTIRRADIATION DIAMETER MEASUREMENTS
OF ThO₂-UO₂ FUEL PELLETS

Pellet	Reading 1 (in.)	Reading 2 ^a (in.)	Average (in.)
1	0.4070	0.4085	0.4078
2	0.4092	0.4089	0.4090
3	0.4085	0.4090	0.4088
4	0.4096	0.4091	0.4094
5	0.4095	0.4090	0.4092

^aTaken 90° to Reading 1.

Gamma Scanning

A total of three rods from face number three was removed from the assembly for scanning of relative gamma activity. The rods were numbers 34 (a corner rod), 35, and 36. The results of the gamma scan are shown in Figure 10. The maximum flux is shown to have been slightly below the midplane of the fuel section. The results are in agreement with the previously observed patterns in the EBWR.

Fig. 10

Relative gamma activity of fuel rods.

Burnup Analysis

In order to determine the maximum burnup attained by the fuel element, a portion of a pellet from an area of maximum flux, in Rod 34, was taken for mass spectrographic analysis for the uranium isotopes. A similar analysis was made on unirradiated fuel from the same batch as that in the irradiation assembly. The results of these analyses are given in Table V.

Table V

PRE AND POSTIRRADIATION MASS SPECTROGRAPHIC ANALYSES OF URANIUM ISOTOPES

Analysis	U ²³³ , w/o	U ²³⁴ , w/o	U ²³⁵ , w/o	U ²³⁶ , w/o	U ²³⁸ , w/o
Preirradiation		0.945 ± 0.01	93.27 ± 0.05	0.47 ± 0.01	5.31 ± 0.05
Postirradiation	2.41 ± 0.02	0.95 ± 0.01	89.94 ± 0.06	1.05 ± 0.01	5.65 ± 0.06

Two methods were used to calculate the maximum burnup from the mass spectrographic analyses. The first method, after Kittel and Paine,⁽¹⁾ gives the fraction of total uranium burnup by either of the following equations:

$$B = \frac{F}{N_0} = \frac{(a_0 + b_0) - (a + b)}{(1 + \alpha) - (a + b)} \quad (1)$$

or

$$B = \frac{F}{N_0} = \frac{c - c_0}{\alpha + c} \quad (2)$$

where a_0 , a ; b_0 , b ; and c_0 , c , are the pre and postirradiation fractions of U²³⁴, U²³⁵, and U²³⁶, respectively; N_0 is the original amount of uranium; F is the amount of U²³⁵ fissioned; B is the total uranium burnup ratio; and

α is the ratio of capture to fission cross sections of U^{235} . The basic assumptions are that the burnup is due entirely to U^{235} fissions and that fission accounts for at least 95% of the total uranium lost.

The second method, from IDO-16620,⁽²⁾ also gives the fractions of total uranium burnup by the following set of equations:

$$B = \frac{(1 + \alpha)(b_0 - b)}{b_0(1 + \alpha - b)} \quad (3)$$

or

$$B = \frac{\frac{c}{b} - \frac{c_0}{b_0}}{\frac{c}{b} + \frac{\alpha}{1 + \alpha}} \quad (4)$$

where the nomenclature in this case has been changed from the original reference to agree with that shown in equations 1 and 2.

In most cases α is a variable parameter dependent upon the reactor and upon the location within the reactor. For the Elk River assembly location in the EBWR, a value of 0.172 was recommended for α .⁽³⁾ The post-irradiation analyses were normalized to 100% excluding the U^{233} fraction to bring the postirradiation analysis into stoichiometric agreement with the preirradiation analysis. The normalized values were used in calculating the burnup. The results of the burnup determinations by the various methods are given in Table VI. The only relatively fair agreement of the values may be attributed to the unknown contribution to fission by the U^{233} isotope. The average value of the four determinations is 1000 MWD/T (2.80×10^{19} fissions/cc). With the maximum to average flux ratio of 1.74:1, the average burnup of the fuel was then about 570 MWD/T (1.61×10^{19} fissions/cc).

Table VI

SUMMARY OF BURNUP DETERMINATIONS

Method	MWD/Ton of Oxide	Fissions/cc ($\times 10^{-19}$)
Eq. 1	1030	2.86
Eq. 2	760	2.12
Eq. 3	1260	3.54
Eq. 4	960	2.68
Average	1000	2.80

Fission Gas Release

Two fuel rods, numbers 34 and 36, were punctured for measurement of released fission gas. The rods were placed, one at a time, in a vacuum chamber which was then evacuated, the cladding punctured, and the released gas collected and measured in a calibrated volume outside the hot cell. After the volume of total gas in the annulus had been determined, a sample of the gas was taken for mass spectrographic analysis. The total volumes of gas and the volume of fission gases (consisting of xenon and krypton) released from each rod are given in Table VII.

Table VII

RESULTS OF FISSION GAS MEASUREMENTS

	Total Gas in Annulus, cc (STP)	Fission Gas in Annulus, cc (STP)	% Release of Fission Gas
Rod 34	6.5	0.11	0.68
Rod 36	8.8	0.04	0.30

The percentage release of theoretically formed fission gas for rod number 34 was calculated using an average burnup value of 570 MWD/T (1.61×10^{19} fissions/cc). The relative gamma activity scan, Fig. 10, indicates a lower average burnup for rod 36. A value of 470 MWD/T (1.32×10^{19} fissions/cc) was used in calculating the theoretical gas yield in this rod. A higher release rate from rod 34 is to be expected because of its higher operating temperature.

Metallography

Rod number 34, a corner rod, was sectioned, with a tubing cutter, about 3 in. below the midplane in the region of the highest flux. The fracture section is shown in Fig. 11. This pellet achieved a burnup of 1000 MWD/T with a cladding surface heat flux of 85,000 BTU/(hr)(ft²). Areas of the section are in relief, and lines appearing to be radial cracks are in most cases boundaries of the relieved areas. No central recrystallization of heat-affected zones were observed.

The five pellets that were removed whole from rod number 36 are shown in Fig. 12. No cracks of any sort were noted in these pellets.

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Fig. 11. Mating transverse fracture surfaces of fuel pellet from region of maximum flux.

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Fig. 12. Whole fuel pellets from region of intermediate flux.

CONCLUSIONS

The fuel element retained its structural integrity during irradiation. Dimensional changes were insignificant. The brazing material used on the center straps, Microbraz No. 50, showed moderate corrosion effects in the EBWR process water. Irregular vertical packing of the fuel pellets caused a nonuniform deposition of oxide film on the fuel tubes.

The $\text{ThO}_2\text{-UO}_2$ fuel pellets showed no signs of central melting, recrystallization, or heat-affected zones, and exhibited excellent dimensional stability. The fuel also showed good gas retention characteristics.

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