INTERIM RESULTS OF END-OF-LIFE EXAMINATION OF MARK-IA
FUEL IRRADIATED IN EBR-II SUBASSEMBLY X068A

by

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ABSTRACT

A total of 36 breaches in Mark-IA fuel-element cladding has been obtained in the run-to-cladding-breaching program (subassemblies X068, X068A, and X068B). With the results from subassembly X068A and the initial indications of cladding breaches from subassembly X068B, the following conclusions may be made:

1. There appears to be one independent mode of cladding breaching of the Type 304L stainless steel cladding in which the failure probability, F , may be characterized in terms of maximum burnup (Bu_{max}) as

$$F(Bu_{max}) = 1 - \exp \left[- \left(\frac{Bu_{max}}{3.97} \right)^{15.7} \right]$$

Explanation of variance = 93%

Cladding-temperature and fuel-fabrication variables analyzed for subassembly X068 do not appear to be significant in the cladding-breaching mechanism.

2. Once a cladding breach occurs, the fuel "extrudes" through the breach.

3. The total diameter change of Mark-IA fuel elements is proportional to the cladding swelling. The creep/swelling

ratio is ≈ 2 over the axial distance that the fuel is in contact with the cladding for maximum burnups on the order of 3.5 at. % of the heavy atoms.

4. All cladding breaches were observed to be longitudinal and to occur axially between the core midplane and the top of the fuel column.

I. INTRODUCTION

Subassembly X068A (61 elements) was reconstituted with 51 encapsulated Mark-IA elements (see Appendix) previously irradiated in subassembly X068 to a maximum burnup of 2.4 at. %. Subassembly X068A contained five double-heat-treated and four centrifugally bonded, vendor-produced elements; the remainder of the elements were ANL-produced elements from fuel batches containing 85-705 ppm silicon. The silicon variation was used to obtain information on the influence of fuel swelling on performance of Mark-IA fuel elements. The interim examinations of this subassembly have been reported previously^{1,2}.

Subassembly X068A was inserted in the core at the beginning of run 47B and irradiated an additional 5601 MWd to 3.7 at. % maximum burnup. The subassembly was removed from the core at the end of run 51B. Irradiation data are summarized in Table I. The thermal operating conditions resulting from high and low ratios of power to coolant flow during reactor operation, calculated by using the BURNUP - COOLTEMP - HECTIC codes, are shown in Figs. 1 and 2 for subassemblies X068 and X068A, respectively.

LOW POWER / FLOW RATIO (R)

RUN 44A

CORE POSITION = 4A3

REACTOR POWER = 50 MWt

MEAN S/A POWER = 452.4 kWt

MEAN S/A FLOW RATE = 59.2 gpm

$R_{low} = 7.6$

TO REACTOR CENTER

HIGH POWER / FLOW RATIO (R)

RUN 45B

CORE POSITION = 4A3

REACTOR POWER = 50 MWt

MEAN S/A POWER = 444.5 kWt

MEAN S/A FLOW RATE = 56.8 gpm

$R_{high} = 7.8$

Fig. 1. Calculated Maximum Cladding Temperature and Coolant ΔT (°F) for Each Element in Subassembly X068 at the Low and High Power/Flow Ratio (R) during Irradiation. (Calculations were based on 50-MWt operation and 9000 gpm total coolant flow.)

Fig. 2. Calculated Maximum Cladding Temperature and Coolant $\Delta T(^{\circ}\text{F})$ for Each Element in Subassembly X068A at the Low and High Power/Flow Ratio (R) during Irradiation. (Calculations were based on 62.5-MWt operation and 9000 gpm total coolant flow.)

TABLE I. Summary of Irradiation Data
for Subassembly X068A

(Compiled by J. C. Case)

I. Pre-Irradiation Data

Date Assembled	December 29, 1970
Subassembly Type	Mark B61
Flow-test Results	~55 gpm
Installed in Reactor Basket	Feb. 22, 1971
Installed in Reactor Position	6B4, February 23, 1971
Initial Reactor Startup in Run	47B

II. Irradiation History

Previous Subassembly Residence X068

History

<u>Run Number</u>	<u>Position</u>	<u>Reactor Power, MWt</u>	<u>MWd</u>
47B	6B4	62.5	505
48C - 49D	6B4	62.5	2321
49F - 51B	6B4	62.5	<u>2775</u>
Total			5601

Estimated Average Burnup of Lead Pin -- 3.3 at. %

Estimated Peak Burnup of Lead Pin -- 3.7 at. %

Estimated Fluence, Reactor Midplane -- 1.6×10^{22} nvt (X068A only)

Estimated Fluence, Reactor Midplane -- 4.6×10^{22} nvt (X068 and X068A)

III. Normal Operating Conditions

Inlet Sodium at 700°F

62.5 MWt

Off-normal Performance: Actual coolant flow was ~55 gpm rather than requested ~48 gpm because of use of incorrect lower adapter.^a

IV. Post-irradiation Data

Last run, 51B, was completed September 20, 1971

Subassembly, was transferred from reactor to HFEF September 29, 1971

^aReference coolant-flow rate based on a core-pressure drop of 40.8 psi.

II. EXAMINATION

A. Gamma Scanning

All the capsules from subassembly X068A were gamma-scanned for indications of breached fuel-element cladding. The method consisted of gamma-scanning the capsule-plenum region for ^{137}Cs activity in the capsule sodium above the fuel element and for ^{133}Xe activity in the gas-filled portion of the plenum. This technique identified 18 suspects out of the 51 high-burnup elements--(i.e., no suspects were found in the ten low-burnup elements at 1.3 at. % maximum burnup)³.

B. Neutron Radiography

Results of neutron radiography have been obtained on all 18 elements from subassembly X068A suspected of having cladding breaches. The capsules were radiographed in groups of six, with at least one high-burnup capsule in a given group being one not suspected of containing an element with breached cladding. These elements were radiographed at two different orientations 90° apart. From the 18 capsules containing suspect elements, 11 elements with breached cladding could be verified by neutron radiography. The radiography results also showed discontinuous sodium distributions in the capsule plenums of 50% of the capsules containing elements with suspected cladding breaches.

The extent of element degradation may be seen in Fig. 3, which shows results of neutron radiography for six of the experimental capsules. All capsules shown in Fig. 3, except capsule 114, are suspected of containing elements with breached cladding. The degradation of the element from capsule 85 is so severe that the element could not be deencapsulated. Because of element degradation, eight of the eighteen suspect elements could not be deencapsulated.

Fig. 3. Neutron-radiography Results for the Core Regions of Capsules 36, 40, 85, 154, 77, and 114.

C. Cladding Breaches

All cladding breaches observed thus far are longitudinal and are axially located between the core midplane and the top of the fuel column. Of the ten deencapsulated (10/18) elements suspected of having cladding breaches, five were initially verified during visual examination⁴. The five elements are shown in Figs. 4-8. Leak-testing of the remaining five deencapsulated suspect elements showed positive indications of cladding breaching for all five by the collection of substantial amounts of ⁸⁵Kr. Leak-testing was first performed at 850°F, with the observation in element 77 of an axial cladding breach about 1/2 in. long, beginning at the core midplane. A probable defect noted at the top of the fuel column coincides better with the observation of no sodium above the fuel pin. The second leak test was performed on the remaining four suspects at 950°F. Visual examination showed cladding breaches in elements 44 and 40 at axial distances of 11-1/4 in. and 8-1/2 in., respectively, from the bottom of the fuel columns. Element 142 also contains a probable cladding breach near the core midplane. Of the deencapsulated elements with cladding breaches, only element 21 has a breach that was not observed during visual examination. Therefore, leak testing opened up the breaches in four of the five elements tested⁵.

D. Fuel Extrusion through Cladding Fissures

Three transverse samples for metallography were taken from two XJ68A capsules (85 and 103) in which the elements could not be deencapsulated. The sections were taken from axial locations showing the most element degradation in the neutron radiographs to determine whether fuel "extrudes" through cladding breaches after rupture. All three sections showed total

Fig. 4. Element from Capsule 70 of Subassembly X068A. (Pinhole breach at right is $9\frac{3}{4}$ in. from the bottom of the fuel column. The larger cladding breach (≈ 1 in. long) to the left is $10\frac{1}{4}$ in. from the bottom of the fuel pin. Top of the element is to the left.)

Fig. 5. Element from Capsule 76 of Subassembly X068A. (Large cladding breach, $\approx 1\frac{1}{4}$ in. long, begins $11\frac{1}{2}$ in. from bottom of fuel column. Top of the element is to the left.)

**Fig. 6. Element from Capsule 94 of Subassembly X068A.
(Cladding breach is about $\frac{3}{8}$ in. long and begins
13- $\frac{3}{4}$ in. from bottom of the fuel column.
Top of the element is to the left.)**

**Fig. 7. Element from Capsule 110 of Subassembly X068A.
(Cladding breach is about $\frac{1}{2}$ in. long and begins
13- $\frac{1}{4}$ in. from the bottom of the fuel column.
Top of the element is to the left.)**

Fig. 8. Element from Capsule 154 of Subassembly X068A.
(Initial cladding breach at right begins 6.75 in. from
bottom of fuel column and appears to be a series of
pinholes. The "teardrop" type of cladding breach
at the left is most likely due to a fission-gas
bubble entrapped between the element and capsule.
Top of the element is to the left.)

breaching of the cladding (Figs. 9-11). Fuel extrusion through the cladding breaches was extensive for both capsules, as shown in Figs. 9 and 11, as was degradation of the fuel and cladding caused by overheating effects. The overheating effect, if any, appears to be more localized in the section (shown in Fig. 10) taken 7 in. above that shown in Fig. 9. The photomicrographs in Figs. 9-11 clearly show why such elements could not be deencapsulated.

E. Postirradiation Diameter Measurements

The peak diameter changes for the ten elements with breached cladding and the eight without are tabulated in Table II, along with other information of interest. Preirradiation diameters were measured on all elements. The diameter changes are, on the average, greater for elements with breached cladding; however, the average burnup of these elements is also greater. The maximum burnup range for the entire subassembly was 3.3-3.7 -at. %; no cladding of elements with a burnup below 3.5 -at. % maximum burnup was breached--i.e., the eight suspect elements that could not be de-encapsulated had maximum burnups ≥ 3.5 at. %.

Two elements and capsules (E157 and E10) were sectioned to provide swelling information on the cladding and adjacent capsule sections from immersion-density measurements. These elements are representative of the hot and cold thermal operating conditions of the subassembly (Table II) and differ in neutron fluence by only 3%. The results are shown in Figs. 12 and 13, which show that the swelling behavior of the capsules and element cladding differ significantly, with the cladding swelling being higher than that of adjacent capsule material. The differences in swelling behavior cannot be explained by differences in thermal conditions for capsule and cladding alone. In general, the creep/swelling ratio of the cladding is

Fig. 9. Polished Metallography Section of Capsule 103
5-1/2 in. above Bottom of the Fuel Column.

Fig. 10. Polished Metallography Section of Capsule 103
12-1/2 in. above Bottom of the Fuel Column.

Fig. 11. Polished Metallography Section of Capsule 85
9-1/2 in. above Bottom of the Fuel Column.

Fig. 12. Total and Swelling Change in Diameter of Element from Capsule 10. [The change in diameter of capsule material is also shown. All swelling data are from immersion-density measurements assuming $(\Delta D/D_0)_{\text{swelling}} = 1/3(\Delta V/V_0)_{\text{swelling}}$.]

Fig. 13. Total and Swelling Change in Diameter of Element from Capsule 157. [The change in diameter of capsule material is also shown. All swelling data are from immersion-density measurements assuming $(\Delta D/D_0)_{\text{swelling}} = 1/3(\Delta V/V_0)_{\text{swelling}}$]

TABLE II. Experimental Data for Deencapsulated Elements from Subassembly X068A

Capsule Number	Fuel Type ^a	Cladding-breaching Suspect	Peak Burnup, at. %	Peak Cladding Operating Temp., °F	$\frac{\Delta D_{max}}{D_o}$, %
1	AGC-319-CB	No	3.3	908 - 943	2.58
10	ANL-142-IB	No	3.4	892 - 926	2.42
45	ANL-144-IB	No	3.4	918 - 956	2.43
78	ANL-143-IB	No	3.4	956 - 1000	2.20
157	ANL-142-IB	No	3.5	984 - 1033	3.59
88	ANL-142-IB	No	3.5	923 - 961	2.54
136	ANL-144-IB	No	3.6	986 - 1036	4.40
102	ANL-143-IB	No	3.6	982 - 1032	3.51
154	ANL-142-IB	Yes	3.5	904 - 940	2.95
142	ANL-144-IB	Yes	3.5	969 - 1016	3.67
77	AGC-319-DHT	Yes	3.5	970 - 1016	3.59
21	ANL-142-IB	Yes ^b	3.6	990 - 1040	4.61
44	ANL-142-IB	Yes	3.6	993 - 1044	4.64
110	ANL-144-IB	Yes	3.6	982 - 1031	4.24
94	ANL-142-IB	Yes	3.6	934 - 974	3.12
76	ANL-142-IB	Yes	3.6	988 - 1038	3.48 ^c
70	ANL-143-IB	Yes	3.6	988 - 1038	3.96 ^c
40	AGC-319-DHT	Yes	3.7	921 - 960	4.73

^aCode: Manufacturer - Batch Number - Additional Description of fuel.

CB = centrifugally bonded (AGC process)

DHT = double-heat-treated (660°F/1 hr. followed by 550°F/1 hr after initial centrifugal bond)

IB = impact-bonded at 500°C (ANL process).

^bCladding breach not observed in visual examination.

^cThe actual $\Delta D_{max}/D_o$ values (6.97 and 5.47%, respectively) for the elements from capsules 76 and 70 occurred at the cladding breaches. The values listed for these two elements are the largest diameter changes not associated with the breaches.

≈ 2 for Mark-IA elements along a substantial axial portion of the element. Hence, swelling and creep probably are interrelated.

Optical metallography has been done on sections adjacent to the breaches shown in Figs. 4-8 and on selected sections from elements 10 and 157. The general observation is that grain-boundary precipitation increases proceeding axially up the fuel column, regardless of the element operating temperature. Secondary intergranular cracking, up to one mil, was also observed in sections adjacent to cladding breaches (Figs. 4-8) at both the fuel and coolant surfaces. This cracking was nonexistent in the cladding of elements 10 and 157, which did not breach. Typical of the secondary cracking is that shown in Fig. 14. Examination by scanning electron microscopy is in progress; this work must be completed for final determination of the fracture mode.

III. ANALYSIS AND DISCUSSION

The statistics on cladding breaches of the different types of fuel elements are tabulated in Table III. Although there appear to be significant

TABLE III. Data on Fuel Elements with Breached Cladding for Maximum Burnup >3.5 at. % for the Types of Fuel in Subassembly X068A

<u>Fuel Type</u>	<u>Silicon Content, ppm</u>	<u>Number of Breaches per Total of Type</u>	<u>90% Range for Cladding Ruptures, %</u>	<u>Median Rank, %</u>
AGC-319-CB	360	1/3	1.6 - 63.1	20.6
AGC-319-DHT	360	3/4	24.8 - 90.2	61.4
ANL-142-IB	350	3/16	33.3 - 66.6	46.9
ANL-143-IB	705	3/9	9.7 - 54.9	28.6
ANL-144-IB	85	3/6	15.3 - 72.8	42.1

≈ 250X

**Fig. 14. Typical Secondary Cracking on the
Outside Diameter of the Cladding Adjacent
to Primary Breach.**

differences between some of the fuel types, too much statistical overlap and too few data points on some of the element types prevent valid comparisons.

Analysis of the data for subassembly X068A indicates that cladding breaches were not observed on elements with maximum burnups <3.5 at. % and that elements in outside-row (cold) positions had the same probability of cladding breaching as inside-row (hot) elements. Therefore, the variation in thermal operating conditions of the fuel elements in X068A does not appear to be a significant factor in the statistics on cladding breaching for elements in subassembly X068A.

At this point, the most significant response parameter is the maximum burnup for predicting probabilities of end-of-life cladding breaching of the Mark-IA fuel element. These cladding-breaching statistics (in terms of maximum burnup) are tabulated in Table IV, along with the preliminary results obtained thus far on subassembly X068B. The elements in this subassembly have recently been deencapsulated, but statistics in cladding breaching were generated only from gamma-scanning results on the capsules. The results in Table IV give the following probability of cladding breaching $F(\text{Bu max})$ as

$$F(\text{Bu max}) = 1 - \exp \left[- \left(\frac{\text{Bu max}}{3.97} \right)^{15.7} \right]$$

Explanation of variance = 93%.

for the Mark-IA fuel elements in EBR-II.

IV. CONCLUSIONS

Final analysis of performance of Mark-IA fuel elements will require examination of deencapsulated elements from subassembly X068B. Work is in

TABLE IV. Statistics on Cladding-Breaching for
Mark-IA Run-to-cladding-breaching Experiments

<u>Maximum Burnup, at. %</u>	<u>Fraction Breached</u>	<u>Median Rank, %</u>	<u>Subassembly</u>
3.5	2/21	7.8	X082A ^a
3.5	6/51	11.0	X068A
3.6	14/51	26.5	X068A
3.7	18/51	34.5	X068A
3.8	21/43	47.6	X068B
3.9	23/43	52.3	X068B
4.0	31/43	70.7	X068B
4.1	35/43	79.9	X068B
4.2	36/43	82.3	X068B

^aX082A experiment with unencapsulated ANL fuel.

progress to firmly establish the modes of cladding breaching and to characterize grain-boundary-precipitation phenomena in Types 316 and 304L solution-annealed stainless steel.

The statistics on cladding breaching generated in subassemblies X068A, X068B, and X082A, indicate that there is one independent mode of cladding breaching, which may be characterized by burnup or time at power.

APPENDIX

EBR-II Mark-IA Driver-fuel Element

A standard EBR-II Mark-IA driver-fuel element consists of a fuel pins 13.50 in. long by 0.144 in. dia, of U - 5 wt % fissium alloy enriched to 52.2 wt % ²³⁵U, which is contained in a jacket, or cladding, of Type 304L stainless steel, with an OD of 0.174 in. and an ID of 0.156 in. A sodium bond, which occupies the annulus separating the fuel and the cladding, serves as a heat-transfer medium. Before irradiation, the level of the sodium bond is nominally 0.65 ± 0.15 in. above the fuel column. Above the sodium is a reservoir, about 0.67 cm^3 , filled with argon.

As burnup proceeds and the fuel swells, bond sodium is displaced upward and reduces the volume of this gas reservoir. However, the gas release ($\approx 2\%$) and fuel swelling ($\approx 17\%$) are relatively small for the burnups reached in this experiment, so that plenum pressures are of the order of 2000 psi.

Details of the typical Mark-IA element are shown in Fig. 15. There are two minor modifications to this drawing for elements encapsulated in B61 capsule hardware -- there is neither spacer wire on the element nor spade. Instead, there is a special end-plug fitting, which makes the total length of the element 17-5/8 in., instead of the 18-3/32 in. shown in Fig. 15. In this experiment, 61 encapsulated elements were packed into the hexagonal tube of a B61 subassembly. Table V lists specifications for the Mark-IA fuel elements used.

Fig. 15. Mark-IA Driver-fuel Element.

TABLE V. Mark-IA Fuel Elements Used in X068 Program

Fuel Pin	AGC- and ANL-produced fuel pins
Identification	Serial number on spade
Chemical and Isotopic Composition	
U, wt %	95.00 ± 1.00
^{235}U in U, wt %	52.18 ± 0.50
$^{234}\text{U} + ^{236}\text{U} + ^{238}\text{U}$ in U, wt %	47.82 ± 0.50
$^{234}\text{U} + ^{236}\text{U}$ in U, wt %	<1.00
Fissium Alloying Elements ^a	5.00 ± 1.00
Oxygen/Metal Ratio	N.A.
Fuel Form	Cast pin
Fuel Density, g/cm ³	17.9
Normal Fuel Smeared Density, g/cm ³	15.3
Fabrication Fuel Dia., in.	0.144 ± 0.001
Normal Fuel Length, in.	13.500 ± 0.032
Fuel Weight, g	64.5 ± 1.5
Volume of Sorbed Gas and Water Vapor	N.A.
Cladding (AISI Type 304L Annealed Stainless Steel)	
ID, in.	0.156 ± 0.0005
Thickness, in.	0.009 ± 0.0005
Sodium Bond	
Volume, cm ³	0.79
Level, in.	0.65 ± 0.15 above fuel pin
Fuel-Cladding Gap, in.	0.006
Plenum	
Volume, cm ³ (cold)	0.67 ± 0.35
Plenum/Fuel Volume Ratio (cold)	0.037
Gas	Argon
Element	
Nominal Weight, g	85 (with spade)
Nominal Length, in.	17-5/8 (without spade)
Cold Activity (at 1 ft), mrem/hr	
Gamma	0.4
Neutron	<0.2

TABLE V. (contd)

^aSilicon varied from 85 to 705 ppm in this experiment. Normally, fissium alloying constituents plus uranium consist of the following elements, in total wt %:

Molybdenum	2.44 $\pm$ 0.17
Ruthenium	1.95 $\pm$ 0.25
Rhodium	0.28 $\pm$ 0.05
Palladium	0.19 $\pm$ 0.04
Zirconium	0.09 $\pm$ 0.06
Niobium	0.015 $\pm$ 0.012
Silicon	0.050 $\pm$ 0.027
Impurities	2550 ppm maximum
Uranium	95.00 $\pm$ 1.00

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