OFF-NORMAL PERFORMANCE OF EBR-II DRIVER FUEL*

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ABSTRACT

The off-normal performance of EBR-II Mark-II driver fuel has been more than satisfactory as demonstrated by robust reliability under repeated transient overpower and undercooled loss-of-flow tests, by benign run-beyond-cladding-breach behavior, and by forgiving response to fabrication defects including lack of bond. Test results have verified that the metallic driver fuel is very tolerant of off-normal events. This behavior has allowed EBR-II to operate in a combined steady-state and transient mode to provide test capability without limitations from the metallic driver fuel.

I. INTRODUCTION

A. Overview

The EBR-II metallic driver fuel has performed well under combined steady-state and off-normal conditions over the course of twenty-two years of operation.[1-6] For the past five years, irradiation programs have required the driver fuel to undergo several prescribed transient events. About six operational transient overpower (TOP) tests and one extended transient overpower test have been conducted each year. Just recently, two demonstrations of loss of primary coolant flow from full power without reactor scram (LOFWOS) were performed.[7-8] These were, of course, the most demanding events which culminated a long sequence of preliminary, less-demanding tests. No premature breaches of the driver fuel were detected during the course of the off-normal events nor during subsequent extended steady-state operation. This report describes the qualification process and the results to date for four major categories of off-normal operation: transient overpower, elevated-temperature, run beyond cladding breach (RBCB), and operation with fabrication defects. Current observations confirm satisfactory off-normal performance and suggest that metallic fuel by nature is very flexible in application, tolerant to off-normal conditions, and benign in breach and beyond.

B. Significance

The initial perceived weakness of metallic fuel - low fuel temperature capability - is now understood to provide several benefits. Consider these insights as a prelude to the technical discussions which follow. Metallic fuel performance in comparison with ceramic fuels is essentially equivalent. All have been found to perform acceptably well under all conditions. Metal fuel may provide ancillary benefits such as less expensive reprocessing

and refabrication and possibly no need for failed fuel identification in the reactor system, but the significant benefits are reflected in core design and control systems which incorporate the inherently safe attributes of metal fuel. The magnitude of transient overpower events are greatly reduced for metallic fuel cores for two basic reasons. Since more U-238 can be loaded into the metallic core than the oxide core and because the neutron economy is improved as a result of the harder spectrum in a metallic fuel core, the reactivity required to offset burnup is substantially reduced. With limited individual control rod worths, the severity of a control-rod-induced transient overpower event is reduced.

For loss-of-flow events, the reactivity introduced upon collapse of the radial temperature profile in a metallic fuel element is much less than that for an oxide fuel element because the thermal conductivity of the metallic fuel is several times higher than the oxide fuel. Since the reactivity response is the product of the average radial temperature gradient and the sum of the Doppler and axial fuel expansion coefficients, which is similar for metal and oxide, the higher thermal conductivity substantially reduces the resulting overtemperature.

Because of these two benefits of metal fuel there exists more margin to sodium boiling for TOP and LOFWOS events than for oxide fuel. For loss-of-heat-sink events, the cladding properties (not the fuel) may limit reliability. If the cladding stress rupture strength is not limiting, the reliability may be controlled by fuel-cladding eutectic formation and growth for some fuel/cladding combinations. Eutectic formation may result in cladding wastage which in turn reduces lifetime even more than induced by stress-rupture at elevated temperature. This is particularly true for EBR-II with a uranium-base alloy fuel and Type 316 stainless steel cladding. Advanced metallic fuels contain zirconium which increases the reaction threshold temperature and also increases the solidus temperature.

The EBR-II driver fuel by demonstrating adequate off-normal performance and reliability has established the foundation for the advanced metallic fuel systems containing plutonium stabilized by zirconium which exhibit even greater safety margins in temperature - the primary parameter of influence.

C. Design Description and Burnup Limit

The EBR-II driver fuel consists of U-5 wt % Fs fuel sodium bonded to Type 316 stainless steel cladding in cylindrical geometry with an initial 75% fuel-smeared density (see references 1 and 2 for detailed descriptions of design and steady-state performance). The current reference design is limited to 8 at.% burnup because swelling and creep deformation of the Type 304 duct consumes the EBR-II fuel handling design tolerance (see reference 9, presented in this conference, for a complete discussion of this topic). An improved fuel design and advanced duct materials are expected to increase the burnup limit to about 14 at.% (see reference 10 for a discussion of the factors limiting metal fuel lifetime).

No reduction in exposure limit has been necessary for the range of transient overpowers which have been conducted or are anticipated. Life fraction analyses are performed for driver fuel exposed to temperatures above the eutectic formation temperature to determine if lifetime should be reduced.[8] Currently, steady-state irradiations and examinations are proceeding

to verify that lifetime following limited eutectic formation is not shortened sufficiently to reduce the current 8 at.% burnup limit.

II. OPERATIONAL TRANSIENT OVERPOWER PERFORMANCE

A. Issues

For metallic fuel, transient overpower conditions were initially expected to affect fuel behavior and reliability. Higher than normal powers or even high rates of power increase would increase fuel temperatures and temperature gradients both radially and axially. Additional cladding stresses due to increased fuel cladding mechanical interaction (FCMI) from thermal stresses, differential thermal expansion, transient fuel swelling, fuel phase transformation, and incremental gas release could generate premature breach of the cladding in the fuel-column region of the element rather than in the dimple restrainer as has been observed for end of life at steady-state conditions.[1-2] In addition, the transient operation of EBR-II required that the fuel be qualified to undergo repeated transients over its lifetime from either reduced or nominal power levels. The length of time at reduced power and the rate of increase of power were initially assumed to affect lifetime.

Because there was no reactor facility available to transient test the EBR-II fuel under the appropriate conditions, several tests and analyses were performed prior to actual whole-core testing in EBR-II.

B. Preliminary Tests and Analyses

Qualification of the driver fuel began in 1979.[4] The phase transformation rate of unirradiated fuel was found to be instantaneous in going from the low-temperature alpha phase to the higher-temperature gamma phase and time dependent for the reverse.[4,11] The magnitude of strain was found to be one volume percent.

Transient fuel swelling due to internal fission gas pressure and creep rates of irradiated fuel were measured at elevated temperature. Experience under steady-state operating conditions had shown that fuel creep rates were sufficiently high to preclude cladding rupture after the fuel contacted the cladding. Since contact is established early in life, subsequent fuel swelling is accommodated by swelling of the matrix into the open, interconnected porosity in the fuel column. For transient conditions, however, it was necessary to determine how additional FCMI stress would partition between plastic deformation of the fuel and the cladding. The results showed that the fuel can creep at high rates at elevated temperature and thereby relax the FCMI stress.[4]

Sections of irradiated cladding were also transient tested to determine the failure stress at high ramp rates.[12] The section containing the dimple restrainer, the dominant site of failure under steady-state conditions, was found to be the weakest under transient conditions. The FCMI stress during a transient was calculated knowing the fuel strains from transient swelling, phase transformation, and differential thermal expansion. From the stress history, the cumulative damage of the cladding was calculated. The results showed that the early Mark-IA, high-smear-density (85%) design would not survive the transient but the standard Mark-II element design would.[4] These results were confirmed by early in-reactor tests which showed substantial cladding strain on Mark-IA elements subjected to repeated transients. In

contrast, repeated (141 cycles) power changes on Mark-II elements translated in and out of the core showed no additional diameter increase. Interchange of two assemblies 12 times between high- and low-power positions for extended exposure resulted in only minimal diameter increase.[4 13] These results led to two dedicated reactor transient test periods.

C. Whole-core Transient Demonstrations

The whole core of driver fuel and seven experimental assemblies of driver fuel operating at normal [600°C peak cladding temperature (PCT)], elevated (630°C PCT) and peak (660°C PCT) temperature went through 56 transients from 24 to 62.5 MWt in 60 seconds, held for 720 seconds, and then decreased back to 24 MWt in 60 seconds. The transients were repeated three times a day (every eight hours) every other day.

No cladding breaches were generated and no performance degradation was observed. The previously-naturally-breached fuel elements failed in the dimple and exhibited minimal degradation. Only one of the three elements leaked additional fission gas during the transient period. The remaining test elements covering a range of burnup and peak temperature exhibited no performance degradation and no additional cladding strain or fuel swelling was observed.[4]

Following this first demonstration of repeated transients at 1.6% power increase per second which was completed in early 1981, a second demonstration of repeated transients was conducted in late 1982.

The high-ramp-rate reactor power transients were about six times faster than the low-ramp-rate series of transients, which were obtained by manual control of a standard high-worth control rod. The fast transients required the use of a computer-controlled drive system to obtain the desired transient shape. The power transient consisted of a linear up ramp of 4 MWt/s from 25 to 62.5 MWt, a hold at 62.5 MWt for 12 min, and a down ramp of about 3.7 MWt/s to 25 MWt. The form of the transient was designed to expose the driver fuel to the highest possible rates of power increase and decrease with a hold at peak power sufficient to cause maximum damage with an extended hold at lower power between transients to allow recovery of the cladding and fuel and increase the burnup of the fuel.

After two trial transients to 50 and 56 MWt, the first power transient from 25 to 62.5 MWt in 9 s was completed on November 4, 1982. Successive transients were then run every day for a total of 13 full-power transients. The power increase was nominally from 10 to 29.5 kw/m, and the maximum PCT increase was 20°C/s.

Five test subassemblies containing driver fuel with burnups from 0 to 9.1 at.% were irradiated at three goal peak cladding temperatures (PCT) as shown in Table I. The elements in these tests were exposed to all combinations of irradiation conditions including steady-state, low-ramp-rate transient, and high-ramp-rate transient. Selected elements also underwent irradiation-temperature change between 600 and 630°C PCT by being interchanged between reactor row-2 and row-5 positions.

The highest-temperature test, XY-15, was positioned beneath the fuel-performance test facility[6] (FPTF) to monitor the coolant temperature,

TABLE I. Test Subassemblies for High-ramp-rate Whole-core Transients

Subassembly No.	Reactor Position	Element Source[a]	Number of Elements	Avg Peak Burnup,[b] at.%	Goal Peak Cladding Temperature, °C	Conditions
C-2866X	2F1	C-2704X	6	7.6	600	Above-nominal PCT
		C-2862X (X273B)	3[c]	9.1		
		C-2862X (B-3353)	7[c]	6.4		
		C-2862X (B-3375)	8[c]	4.4		
		C-2862X	5[c]	0.8		
		New	60	0.3		
		TED-monitor dummy	2			
C-2867X	2D1	C-2707X	6	7.1	630	Boundary of operation
		C-2862X (X273B)	3[c]	8.9		
		C-2862X (B-3353)	4[c]	6.5		
		C-2862X (B-3375)	3[c]	4.4		
		C-2862X	1[c]	0.8		
		C-2863X (B-3353)	2[c]	6.7		
		C-2863X (B-3375)	2[c]	4.2		
		C-2863X	4[c]	0.8		
		New	53	0.3		
		TED-monitor dummy	2			
		SS dummies	11			
C-2885X	5B4	C-2864X (X273B)	6[c]	9.0	600	Above-nominal PCT
		C-2864X (B-3353)	11[c]	6.4		
		C-2864X (B-3375)	11[c]	4.1		
		C-2864X	5[c]	0.7		
		New	56	0.3		
		TED-monitor dummy	2	0.3		
C-2886X	5D4	C-2865X (B-3282S)	6	7.9	630	Boundary of operation
		C-2865X (B-3353)	11[c]	6.3		
		C-2865X (B-3375)	11[c]	4.1		
		C-2865X (B-3347)	6[c]	2.7		
		C-2855X	6[c]	0.7		
		New	49	0.3		
		TED-monitor dummy	2			
XY-15	5F3 (FPTF)	8-3282S	6	7.7	660	Boundary of operation with uncertainties
		C-2863X (B-3282S)	2[c]	8.0		
		C-2863X (B-3353)	5[c]	6.5		
		C-2863X (B-3375)	5[c]	4.2		
		New	1	0.3		
		TED-monitor dummy	1			

[a] Elements from subassemblies in parentheses had experienced only steady-state irradiation before any transient testing.
[b] After transients.
[c] Previously underwent 55 low-ramp-rate whole-core transients.

flow, and DN activity. The temperature of the coolant leaving XY-15 was measured by the FPTF inlet thermocouples; a typical transient is shown in Fig. 1. The other test subassemblies, which occupied open-core positions, contained passive thermal monitors (thermal-expansion-difference, or TED, monitors) to verify predicted goal PCTs. In every case, the temperatures measured by the monitors were within 10°C of the goal temperatures.

Visual examination of the elements after the test revealed no unusual characteristics. No significant weight loss was observed by weighing the elements, which confirmed the observed in-reactor reliability. During transient operation, only minor fission-gas activity was observed. The source was considered to be experimental RBCB elements that had previously breached during steady-state operation and were being subjected to transients. No gas release was observed from the many high-temperature test elements that contained unique gas tags specifically included for identification.

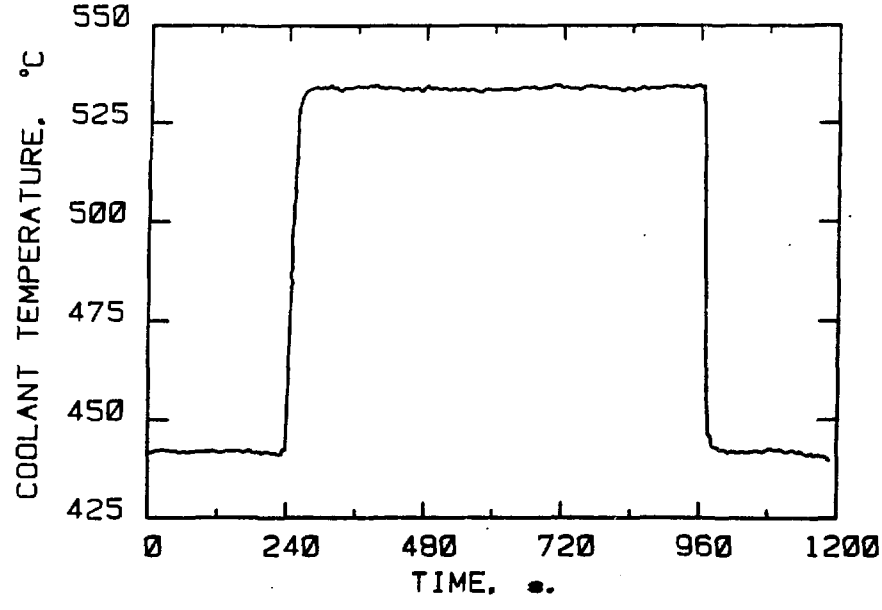

Fig. 1. Exit Coolant Temperature of XY-15 as Measured by FPTF during a Typical High-ramp-rate Transient

Helical profilometry of all the higher-burnup elements was performed to determine whether repeated transients at high ramp rates caused incremental cladding deformation. Figures 2 and 3 summarize the profilometery traces for each of the elements in two of the five test subassemblies. Since the traces are categorized by element source and therefore burnup, one can observe the cladding-diameter increase with burnup. The periodicity in the axial direction, particularly for high-burnup elements, is common for elements irradiated under steady-state conditions and results from element/element and element/spacer-wire interaction. The dimple restrainer near 360 mm is indicated by the abrupt indentation in the profilometer trace. The diameter increases are plotted collectively in Fig. 4.

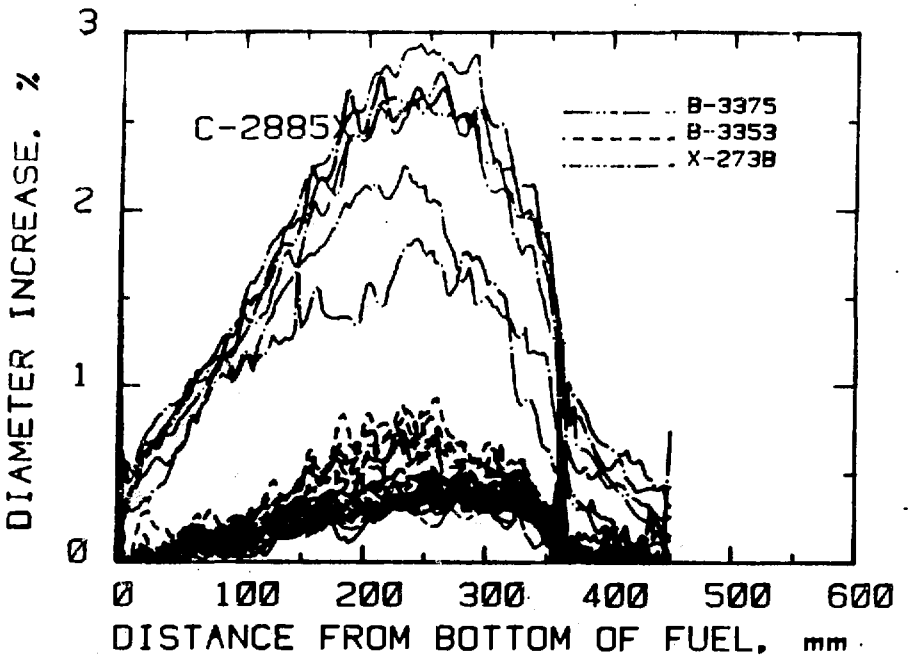

Fig. 2. Axial Profile of Diameter Increase of Test Elements in C-2885X

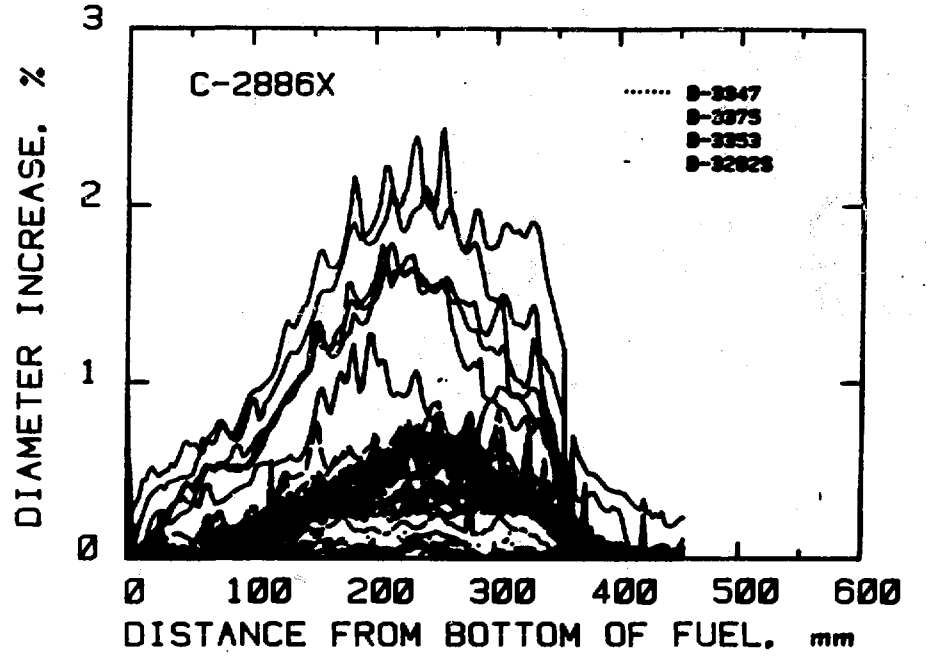

Fig. 3. Axial Profile of Diameter Increase of Test Elements in C-2886X

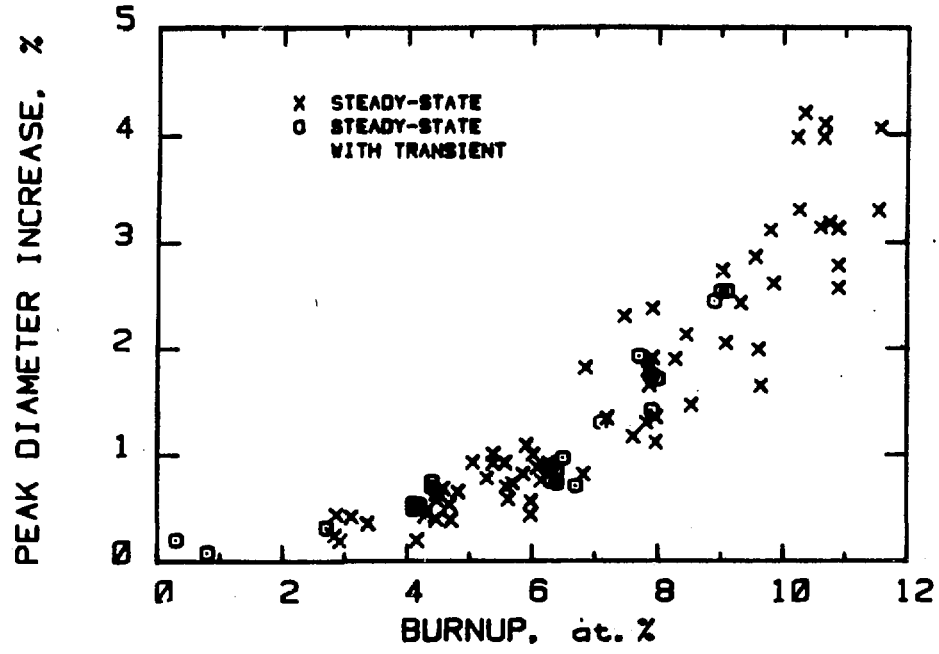

Fig. 4. Comparison of Peak Element-diameter Increase (average for several elements) Resulting from Steady-state Operation Only and Steady-state and Transient Operation Combined

The relationship of diameter increase for steady-state only and steady-state with transient operation is significant. On the basis of a large number of measurements (each datum point represents the average of several measurements), there is no additional diameter increase as a result of transient operation. Although large increases were not expected, stresses from transient fuel swelling, internal gas pressure, fuel phase-transformation strain, and differential thermal expansion may have caused some cladding strain in the temperature range of 400-660°C. Laboratory tests had shown earlier that at fuel temperatures above 650°C, fuel creep could effectively ameliorate transient fuel swelling, but there was an uncertainty at lower temperatures where small differences in the competing mechanisms could have had significant effect. The fact that there was no additional strain as a result of the transients indicates that there were no significant cladding stresses on either the power-increase or power-decrease phase of the transient. High reliability during transient operation can certainly be expected, since no measurable cladding damage has been observed.[15]

In summary, the high-ramp-rate whole-core transient tests demonstrated the reliability and performance of the EBR-II metallic driver fuel from 40 to 100% of element peak power, at rates up to 16% power increase per second of initial power for a minimum of 13 transients during the lifetime of the fuel. Although this is more than sufficient for the projected transient duty cycle there is no foreseeable limit to the number of transients, since there was no measurable damage after 13 of them. In addition, because several test subassemblies were operated at elevated temperature, there are no restrictions on operating indefinitely at the boundary of operation with

uncertainties. This provides an operating margin of 75°C in PCT for standard driver fuel. There are, therefore, no unresolved performance, reliability, or safety issues restricting EBR-II from unlimited transient operation.

III. ELEVATED TEMPERATURE PERFORMANCE

A. Issues

Metallic fuel has traditionally been viewed as having limited capability at elevated temperature because of the lower fuel melting temperature than ceramic fuel and the formation of a lower-melting temperature eutectic between the fuel and cladding compositions. Since the thermal conductivity is about four times higher than oxide and the fission gas porosity only reduces that conductivity by less than fifty percent, the fuel centerline temperature is significantly less for all power conditions. Recent rapid transient tests of Mark-II driver fuel have demonstrated that substantial fuel melting at four times normal power are required for failure.[16] The lower melting temperature of the fuel cladding eutectic is only exceeded for loss-of-flow-without-scram conditions where the primary coolant temperature increases. EBR-II has recently conducted several LOFWOS transients including two from full power with a slightly decreased (28°C) initial inlet coolant temperature.[7-8] These resulted in the limited formation of molten eutectic fuel-cladding alloy in some driver assemblies. Some of these assemblies are currently being irradiated to determine if there was any shortening of life.

B. Tests and Analyses

Several tests have been performed to characterize the formation and growth of molten eutectic alloy. For the Mark-II driver fuel with Type 316 SS cladding, the formation temperature of 715 ± 5°C was determined from diffusion couples heated for 300 h.[17] To address the formation and growth in irradiated elements, tests were conducted on samples[18] and elements[19] by heating them in a furnace. The results demonstrated that growth of molten eutectic only progressed rapidly at temperatures much above the eutectic temperature and that the formation of intermetallic compounds could retard the growth. In addition, the molten phase preferentially grew into the fuel.

To establish a true operating limit for the driver fuel, three assemblies of driver fuel composed of fresh elements and elements preirradiated to 8 at.% burnup were irradiated at temperatures near or above the eutectic formation temperature. The objectives of the tests included 1) characterization of the formation and growth of eutectic, 2) determination of the penetration rate of the eutectic into the cladding, 3) determination of the time to cladding breach as enhanced by cladding wastage and increased cladding stresses at elevated temperatures, 4) characterization of breach and loss of fuel, 5) evaluation of release of fission products and quantification of contamination, and 6) determination of enhanced delay neutron (DN) activity due to elevated temperature and direct release of DN precursors from molten eutectic at the breach. The first two tests, XY-13 and XY-14, demonstrated that irradiated elements operated just at the eutectic temperature were reliable for at least 12 hours, the length of the test. In fact, destructive examination showed that only the prior fuel-cladding chemical interaction layer formed eutectic which consists of about four percent of the cladding wall. The cladding interface was protected by a continuous layer of UFe_2.

The following test, XY-22, was designed to provide breach at elevated temperature. The experiment contained 55 elements with burnups from 0 to 8 at.%, and six nonfueled positions which contained multiple passive temperature monitors (TED) for verification of peak temperatures. The assembly was positioned under the FPTF with dedicated temperature, flow, acoustic, and DN monitors. After 19 hours of preconditioning to allow for buildup of an active fission gas inventory, reactor power was increased to obtain the goal temperature. After 42 minutes the FPTF DN detector signal increased seven times above background indicating breach and the reactor was shut down. Shortly after shutdown, fission gas activity was detected in the reactor cover gas. Only 2% of the fission gas was released and there was minimal fuel loss. Post-test analysis of the TEDs verified the peak cladding temperature of 793°C. The hottest element was the one and only element breached. The breach was in the dimple restrainer, the same location for breach under normal steady-state conditions and in the same manner with a crack growing from the outer to the inner surface but with eutectic residue in the crack, Fig. 5. Another dimple also contained a crack which had not completely penetrated the wall.

Destructive examination was performed on five elements to characterize eutectic growth and cladding attack with the following results:

Element Position	Peak Burnup, at.%	PCT, °C	Cladding Loss, %
16	7.7	793	4
32	7.5	759	4
24	4.0	793	17
33	2.4	778	25
23	0.0	769	33

The eutectic in the high burnup elements was essentially confined to the prior fuel-cladding-chemical-interaction (FCCI) zone. The combination of the formation of UFe_2 on the cladding side of the molten alloy, and nearly complete interconnection of fission gas porosity on the fuel surface, Fig. 6, essentially isolated the fuel from the cladding and prevented further interaction which requires mass transfer of fuel and cladding species.

In the low burnup and fresh elements there was limited or no protection by fission gas porosity and although a UFe_2-type boundary layer was present, significant cladding wastage was observed as shown in Figs. 7 and 8. The additional solid fission products such as zirconium, molybdenum, and ruthenium at high burnup may play a role in retarding eutectic growth. It is obvious though from Fig. 8 that in unirradiated elements, the molten alloy settles lower in the annulus between the fuel and cladding, and that the fuel slug is drawn toward the cladding surface continually supplying fresh fuel and cladding to form even more liquid alloy. The eutectic alloy that settled in the bond sodium annulus then froze at a slightly lower elevation which was below the solidus temperature of the alloy.

An analytical measure of reliability was developed to predict the likely loss of life of standard driver fuel exposed to eutectic temperatures during its normal steady-state lifetime. Since the time at elevated temperature was predicted to be less than two minutes, EBR-II was able to demonstrate complete LOFWOS from full power without breach and minimal loss of element lifetime.[8]

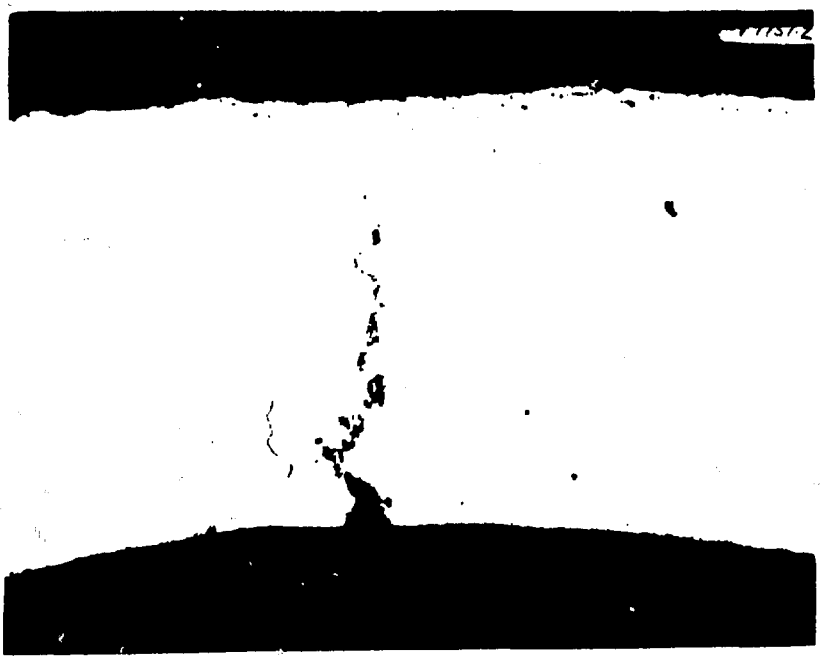

Fig. 5. Cladding Breach of E16 at 7.7 at.% Burnup in the Dimple After 42 minutes at 793°C PCT. The widest opening of the crack is on cladding outside surface; only limited molten alloy can be observed on the cladding inner surface 200X.

Fig. 6. Molten Alloy at the Fuel-cladding Interface in E16 at 7.7 at.% Burnup Near Top of Fuel After 42 minutes at 765°C PCT

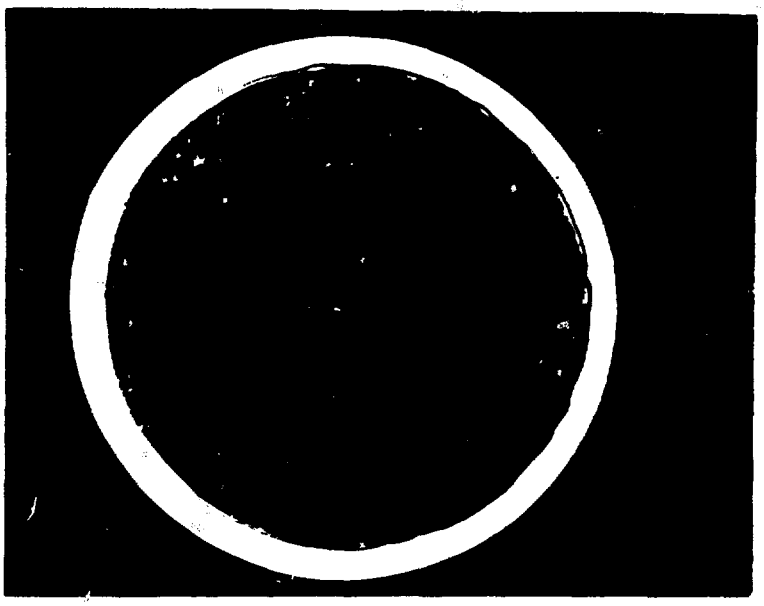

Fig. 7. Uniform Cladding Attack and Liquid Phase Throughout the Fuel in E33 at 2.5 at.% Burnup Near Top of Fuel After 42 minutes at 778°C PCT

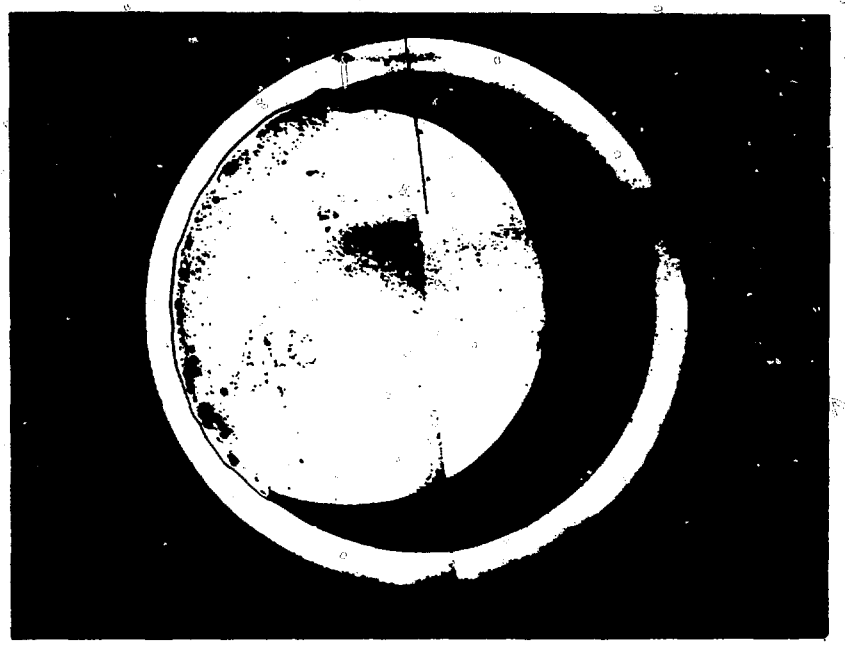

Fig. 8. Directed Cladding Attack in E23 at 0 at.% Burnup Near Top of Fuel After 42 minutes at 769°C PCT

IV. RUN-BEYOND-CLADDING-BREACH PERFORMANCE

A. Issues

Since the metallic fuel and sodium are compatible, there is no interaction with the bond sodium or the primary system sodium coolant. The remaining questions regarding RBCB operation are related to fuel motion and loss, breach propagation, detectability and system contamination.

Experience at EBR-II has shown that RBCB operation of metallic driver fuel is completely benign. Intentional RBCB operation of experimental elements has been very uneventful for both steady-state and transient operation. Unexpected breach of standard driver fuel below the exposure limit has been suspected at times based upon apparent fission gas release to the reactor plenum, but no defect has ever been verified. Releases are often difficult to assign to specific assemblies because of the extensive RBCB program on oxide fuel in EBR-II and the general likelihood of multiple breaches as a consequence of the experimental fuel irradiation programs. The activity from releases from driver fuel are insignificant compared to the substantial releases from RTCB and RBCB oxide fuel experiments.

B. Tests and Analyses

RBCB testing of driver fuel has been conducted to supplement the experience gained from natural end-of-life breaches, initially with elements in capsules[20] and subsequently with open-core tests.

The earliest naturally breached driver fuel element in EBR-II operated for about forty days beyond breach.[21] The defect was located at the spacer-wire-to-tube weld; the defect was caused by interaction with the hexagonal duct during final assembly. Since the defect was in the plenum region, intermittent halogen fission product release resulted from exchange of primary and bond sodium. A subsequent, but early, lower-weld defect exhibited some loss of bond sodium and provided useful information regarding detection, but showed no abnormal performance.[22] A summary of early fission product releases and confirmation testing with predefected elements has been reported.[23]

Several driver fuel RTCB qualification assemblies with Mark-II elements have run one to three days beyond breach with no performance concerns. One experiment, X208, generated seven breaches above 10 at.% burnup over a period of 22 days of operation beyond the first breach. An instrumented fuel assembly also accumulated multiple (5) breaches. All of these breaches were located at the restrainer dimple at the top of the fuel column.[24] Therefore, no fuel is exposed or lost through the very small defect. Bond sodium is slowly expelled from the plenum followed by some plenum fission gas. Usually not all the fission gas is released through the tight crack in the dimple. There has never been any indication of breach propagation as confirmed by the lack of spatial and time relationship when multiple breaches results. A limited number of upper weld breaches and another spacer-wire weld defect have confirmed that there are no related performance issues. Three dimple breaches and one upper weld natural breach were included in the low-ramp-rate whole-core transient tests; other than minimal additional gas release from one of these, there was no effect of RBCB operation. Plenum breaches are totally benign - an irradiation for 130 days confirms that the only potential impact may be accumulation of fission gas in the reactor system.

These results, however, do not deal with the issues related to fuel column failure. Only one early Mark-II element breached in the fuel column and it ran beyond breach only one day. Less than 1% of the fission gas was released and about 22% of the Cs-137 was accounted for in the primary coolant. The longitudinal crack in the cladding was so small as to be invisible at 10X and therefore released no fuel.[25]

To evaluate the early RBCB behavior of Mark-II fuels after the fuel had swelled into contact with the cladding, an element which had reached moderately high burnup (7.1 at.%) was sectioned 330 mm above the bottom of the fuel and the upper 20.6-mm length of cladding was removed to expose 2.6 cm^2 of fuel surface. The element was irradiated at full power under normal conditions for 3.9 full power days. Figures 9 and 10 show the condition of the fuel before and after irradiation. The fuel swelled diametrically about 5% as measured from photographs but the unrestrained, porous fuel remained intact with no adverse effects. The limited swelling suggested that a natural fuel column breach would exhibit only slow crack extension with extended RBCB operation. Without reaction with the coolant, only fuel swelling would drive fuel loss and may be unable to cause blockage. The delayed neutron signal was characteristic of a recoil source with no signal amplification.

Currently, tests are being conducted to evaluate RBCB behavior with a fuel column breach. The robust forgiving nature of the Mark-II element was demonstrated when an element at 8.1 at.% burnup survived 36 days after the cladding was thinned to only 25% of the original wall, Fig. 11. To accelerate breach, a subsequent test was initiated with the cladding of one element (8.4 at.% burnup) thinned to 8% of the original wall thickness. The element did not breach upon reactor startup as expected and continues to be irradiated. Further results will be reported when available.

C. Extrapolation

Experience with lower weld, top weld and plenum breaches indicate totally benign behavior. The normal end-of-life breach in the dimple restrainer is quite benign with only bond sodium and fission gas being released. Only one RTCB element generated a small DN signal upon breach. However, since the driver fuel design is evolving toward the Mark-IIA with a spherical rather than a wedge-shaped indentation and an increased plenum, fuel column breach may result at very high burnup. The RBCB behavior at high burnup with the additional solid fission product accumulation may increase fuel cladding mechanical interaction and result in more exposed fuel. Additional tests have been initiated to evaluate this behavior.

V. FABRICATION DEFECTS

A. Issues

The primary concern related to fabrication focused on the presence of bond sodium defects or the lack of sodium bond. Fabrication practice has conservatively required bond level and bond quality inspection. Early experiments and recent results have shown that this may not be a critical issue.

Other concerns have focused on fuel composition and structural quality. The composition has been shown to be quite homogeneous and merits no further discussion. Fuel porosity and hot tears from casting have no performance implications although the fuel must meet the required fissile mass per length limitations.

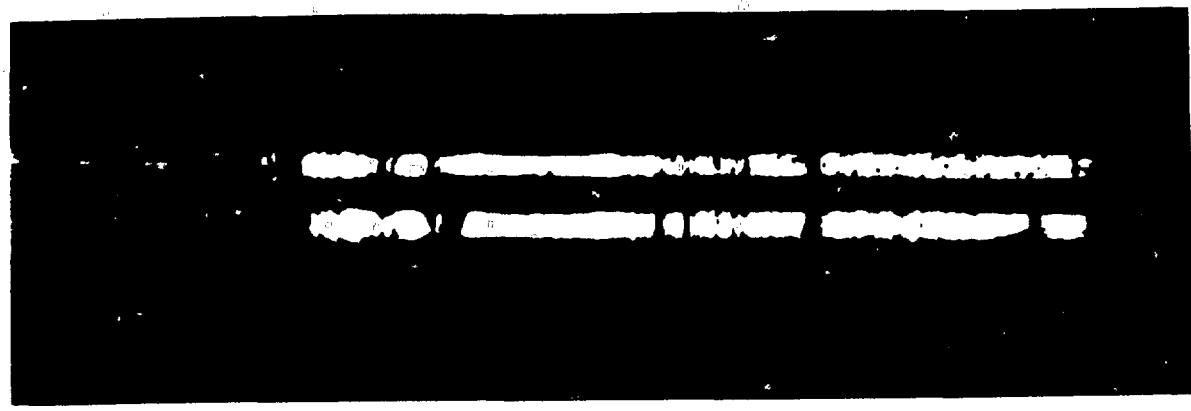

Fig. 9. Exposed Fuel at 7.1 at.% Burnup Before RBCB Operation

Fig. 10. Exposed Fuel at 7.1 at.% Burnup After 3.9 Days of RBCB Operation

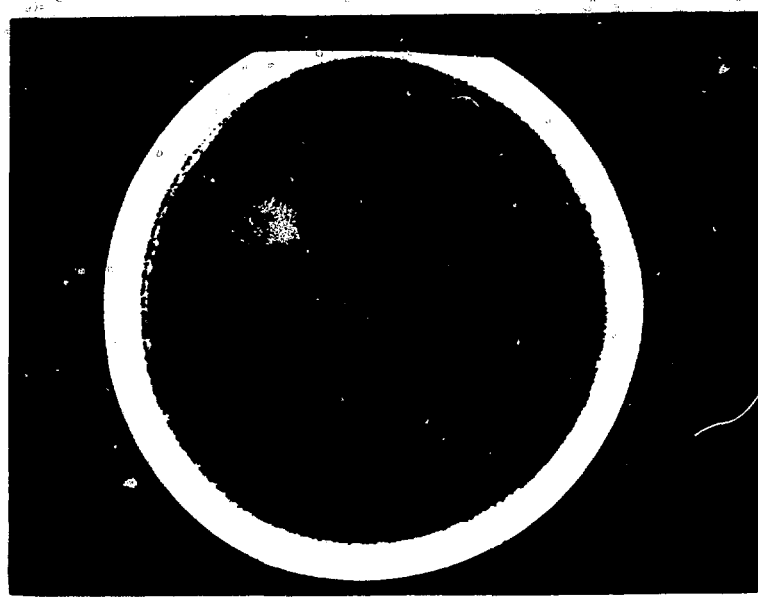

Fig. 11. Cross Section of Element at 8.1 at.% Burnup After 36 Days of Irradiation with Only 25% of the Original Wall

Concerns with the cladding include primarily the closure welds and secondarily with the quality of the finished product. Welds are inspected and tested and the element is visually inspected for surface defects.

The fuel specification has been critically evaluated and relaxed. The design is very forgiving and requires no strict tolerances on smear density, cladding dimensions, or bond level. The fuel specification allows more flexibility on impurity content and structure. Crystallographic texture in the fuel is no longer considered significant and is only evaluated upon any significant change in the fabrication process.

B. Lack-of-Bond Tests and Analyses

Birth defects were simulated by intentionally fabricating Mark-IA elements with part or all of the bond sodium around the circumference eliminated. The elements were irradiated for 1.2 equivalent full power days and found to perform satisfactorily.[26] The elements were reirradiated at normal operating conditions to 2.5 at.% burnup.[27] The fuel restructured into a "bamboo" configuration where the unbonded fuel slumped to contact the cladding at periodic intervals as a consequence of the reestablished thermal bond. The cladding diameter increased in the regions of fuel contact and showed little change where there was less fuel. The elements did not breach and those with a partial bond performed superior to normal elements.

Since EBR-II is in transition to a U-Pu-Zr driver fuel of larger diameter, similar tests are now being conducted, including total absence of bond sodium.

VI. CONCLUSIONS

The EBR-II driver fuel has demonstrated robust reliability over repeated transient overpowers and elevated-temperature operation resulting from LOFWOS. Run beyond cladding breach has indicated totally benign behavior. Fabrication defects including lack of bond have not limited reliability or performance for the Mark-1A design. Tests with current designs are expected to confirm these observations at high burnup.

The performance base of the Mark-II U-Fs elements is now being extended to the inherently safe and economical Integral Fast Reactor concept with U-Pu-Zr fuel. The goal is to demonstrate that EBR-II driver fuel of the advanced LMR design[28] will retain all the inherently safe characteristics of the current Mark-II driver fuel while being even more robust and forgiving to high burnup.

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