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EXPERIENCE WITH LIFETIME LIMITS FOR EBR-II CORE COMPONENTS*

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INTRODUCTION

The Experimental Breeder Reactor No.2 (EBR-II) is operated for the U.S. Department of Energy by Argonne National Laboratory and is located on the Idaho National Engineering Laboratory where most types of American reactor were originally tested. EBR-II is a complete electricity-producing power plant now in its twenty-fourth year of successful operation. During this long history the reactor has had several concurrent missions, such as demonstration of a closed Liquid-Metal Reactor (LMR) fuel cycle (1964-69); as a steady-state irradiation facility for fuels and materials (1970 onwards); for investigating effects of operational transients on fuel elements (from 1981); for research into the inherent safety aspects of metal-fueled LMR's (from 1983); and, most recently, for demonstration of the Integral Fast Reactor (IFR) concept using U-Pu-Zr fuels. This paper describes experience gained at EBR-II in defining lifetime limits for LMR core components, particularly fuel elements.

EBR-II DESCRIPTION AND HISTORY

Designed in the late 1950s, built in the early 1960s, and brought to power in 1964, EBR-II is an unmoderated sodium-cooled reactor with a design power of 62.5 MWt, a closed intermediate sodium loop, and a conventional steam plant and turbine producing 18.5 MW of electricity. A schematic of the total plant is shown in Fig.1. EBR-II comprises a main reactor building, a sodium boiler building and a power plant; an adjacent Hot Fuel Examination Facility (HFEF) is used to examine irradiated subassemblies. Part of the HFEF was originally the Fuel Cycle Facility (FCF) where the U fuel was pyro-reprocessed 1965-69.

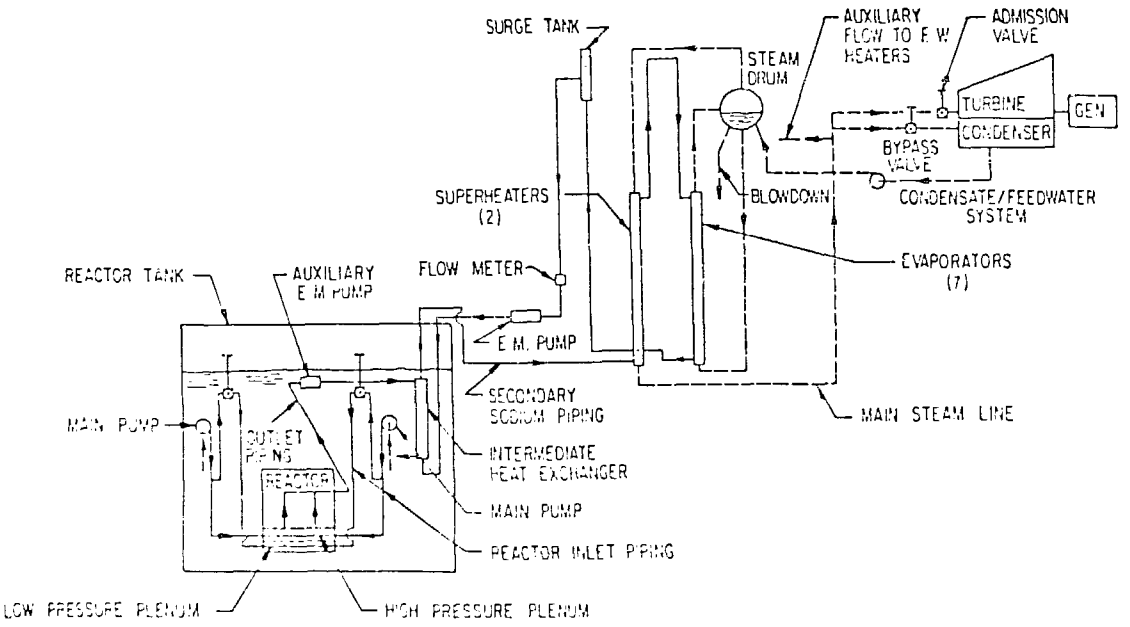

Fig.1
Schematic Layout of the Experimental Breeder Reactor No.2 (EBR-II)

The reactor, primary components and pipes, and much of the fuel-handling equipment are submerged in a double-walled tank containing 340 m^3 of 370°C sodium. Two centrifugal primary pumps, rated at 0.347 m^3/s, take suction from the bulk sodium, circulating it in a single pass through the reactor, a single outlet pipe to the intermediate heat exchanger (IHX), and back to the tank. A DC electromagnetic pump on the outlet pipe, together with inherent natural convection, removes decay heat should a primary pump fail. The secondary system is an intermediate closed loop between the primary and steam systems that contains 50 m^3 of sodium moved from a surge tank to the IHX at 0.41 m^3/s by an AC electromagetic pump. The sodium is heated from 305 to 467°C in the IHX and flows through two parallel superheaters and seven parallel evaporators back to the surge tank. Superheated steam enters the power plant at 8.6 MPa; it can be dumped to the condenser or used at the turbine generator to produce electricity, which is distributed to a 138-kV commercial power loop.

The reactor was originally controlled by twelve fueled rods. Eight high-worth rods with B_4C followers are now used, allowing four positions to contain instrumented in-core test facilities. Any one of the remaining eight rods may be used for control while one rod can be driven by computer to provide automatic power control and shaping of power transients. Two fueled safety rods supply an independent shutdown method and are used during operation for reactivity shim control.

The original goal for EBR-II was to demonstrate feasibility of a sodium-cooled fast reactor operating as a power plant with adjacent fuel-reprocessing capability of 1000 kg/year. During the five years of operation in this mode, over 35000 U elements were made in the FCF, with recycle times as low as a month. This initial phase of operation showed that on-site reprocessing removed the need for large fuel inventories, and for spent fuel to leave the reactor site, advantages central to the current IFR concept.[1]

EBR-II next became the chief irradiation facility for LMR fuels and materials for eventual use in the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor(CRBR). A variety of experimental plutonium-bearing oxide, carbide, nitride and metal elements were irradiated 1970-80 to cover a range of designs and cladding materials. Blanket rods, absorber materials and rods, element spacing methods, subassembly types, and duct and structural materials, have also been investigated. Elements were at first sealed in sodium-filled capsules for fear of the consequences of failure, but soon most were irradiated in contact with primary sodium. Tests were begun cautiously with peak exposures only gradually raised. As favorable experience was gained, some endurance testing or run-to-cladding-breach (RTCB) testing was allowed. RTCB testing required the identification and removal of a breached element because the cover seals leaked small amounts of fission gas.

CRBR licensing involved showing that mixed-oxide fuel elements could be operated for a limited time beyond failure without serious degradation. This requirement necessitated intentional run-beyond-cladding-breach (RBCB) testing in EBR-II.[2] After preparations that included installing a cover-gas cleanup system (CGCS) and removing a delayed-neutron (DN) trip function, this mode of testing was begun in 1978 and has continued to the present day.

In 1981, after several years preparation and when FFTF came to power, EBR-II became a facility in which off-normal or operational reliability testing (ORT) of LMR fuel elements could be performed.[3] ORT includes RBCB testing, testing to simulate reduced-power operation and periodic mild overpowers, and in-situ testing of elements up to 100% overpower to determine failure thresholds. A collaborative program with Japan has been in progress since 1981 to determine the reliability of elements for the MONJU reactor.[4,5]

After a decade of testing that demonstrated the advantages of sodium in removing decay heat, a follow-on program to show the general safety characteristics of metal-fueled LMR's began in 1983. In 1986 it culminated in historic tests of loss-of-flow-without-scram and loss-of-heat-sink-without-scram reactor conditions, which proved to be entirely benign.[6] These tests, as well others involving IFR ternary-alloy metal fuel development, are discussed in parallel exchange papers.

CORE STRUCTURE LIMITS

EBR-II was built before the phenomenon of stainless steel swelling was discovered. The original design clearances between components that allowed for thermal expansion and creep could not for long accommodate the volume increase and change of shape accompanying fast-neutron irradiation. Consequently, the major core components--subassembly ducts and control-rod thimbles--have had to be routinely replaced to avoid problems with fuel handling. This replacement has not, however, affected plant factor because it has been done during the regularly scheduled shutdowns for change out of experiments and driver fuel.

Subassembly diametral growth is limited by the clearance between ducts, which is approximately 0.025 in.(0.6 mm). But swelling in one subassembly duct may be partly offset by lower swelling in other ducts so that the allowable swelling for a given duct will depend on the number of ducts, their individual swelling (which is material dependent), and the core-wide cumulative value. A surveillance of duct swelling by calculation and by systematic measurement of push-pull forces on subassemblies during fuel handling has been used since 1975 to avoid local regions of high swelling and potential problems.

There are two further practical limits to duct swelling. The first is a diametral limit of 0.040 in (1 mm) imposed by the hexagonal openings in the in-vessel basket used to store subassemblies while they cool. The second is a limit on axial swelling dictated by the initial clearance between the top of a subassembly and its hydraulic holddown finger on the underside of the reactor cover. Measurements made during the handling of two early subassemblies[7] that proved troublesome suggested that the upper limit for length increase of ducts was 0.20-0.25 in (0.5-0.6 mm); above this value deformation of the subassembly upper adaptor would occur for no apparent length increase (Fig.2). A special gauge subassembly inserted at shutdown has also been used since that time to check this clearance for different core locations.[8] The thimbles or guide tubes for the control rods have been similarly replaced when diametral swelling has approached 0.040 in (1 mm). Although these components are not removed to the storage basket, this swelling value is one that experience has shown is a practical limit for easy withdrawal.

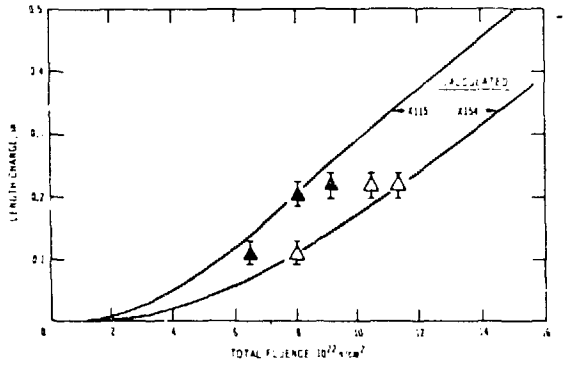

Fig.2
Measured and Calculated Lengths of Two EBR-II Subassemblies That Define Growth Limit

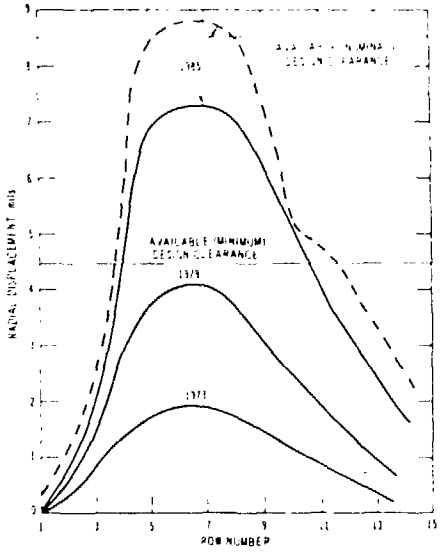

Fig.3
Calculated Radial Displacement of EBR-II Upper Grid Plate Showing Benefit of Clearances

A component whose swelling has been of greatest concern is the upper of the two plates in the grid-plenum assembly used to locate subassemblies in-core. The fear has been that swelling of the upper plate may produce a radial misalignment of the sets of holes that keep subassemblies vertical, leading to difficulty in inserting and withdrawing the lower pole pieces. Because the plenum assembly is impossible to replace such swelling might be a major life-limiting factor for EBR-II. Fortunately, fast flux at the upper grid plate is very low[9] and can be minimized by subassembly shielding. Figure 3 shows estimates made in 1975 of the relative displacement of the upper grid plate with time. They showed that available clearances between polepieces and holes could accomodate this displacement, at least until 1985; measurements with the gauge subassembly have since confirmed these values.[9] Additional clearance was incorporated in 1977 by reducing slightly the diameter of the lower pole pieces. With this correction no problems are foreseen during the remaining life of the reactor.

FUEL ELEMENT LIMITS

As of April this year (Run 143) over eleven thousand experimental LMR fuel elements had been irradiated in EBR-II; they took a variety of forms and were contained in about 800 experimental subassemblies, as listed in Table I. Most tests were first designed for specific goal exposures. For example, proof tests of FFTF elements were designed for 80,000-100,000 MWD/tonne or 8-10 at.% burnup, with several interim examinations. As indicated earlier, however, to an increasing extent through the 1970's RTCB tests were performed to determine

the factors that eventually cause fuel-element failure, while RBCB testing began in 1978. Figure 4 shows how this change in mode of testing caused the incidence of failures in experimental elements to increase from the two chance failures before 1970 to about 15 intentional failures per year by 1979-80, a value that has remained about constant to the present time.

Table I
EBR-II Irradiations by April 1987

Experiment Type	Number of Elements, Capsules,etc	Peak Exposure
Mixed Oxide	3,555	19.9 At.%
Mixed Carbide/ Nitride	1,202	19.1/9.5 At.%
Metal Fuel (U-Fs/U-Pu-Zr)	6,938	18.9/10.0 At.%
Structurals/ Absorbers	1,604	2.98×10^{23} nvt 20×10^{21} cap./cm^3
Other Tests	586	

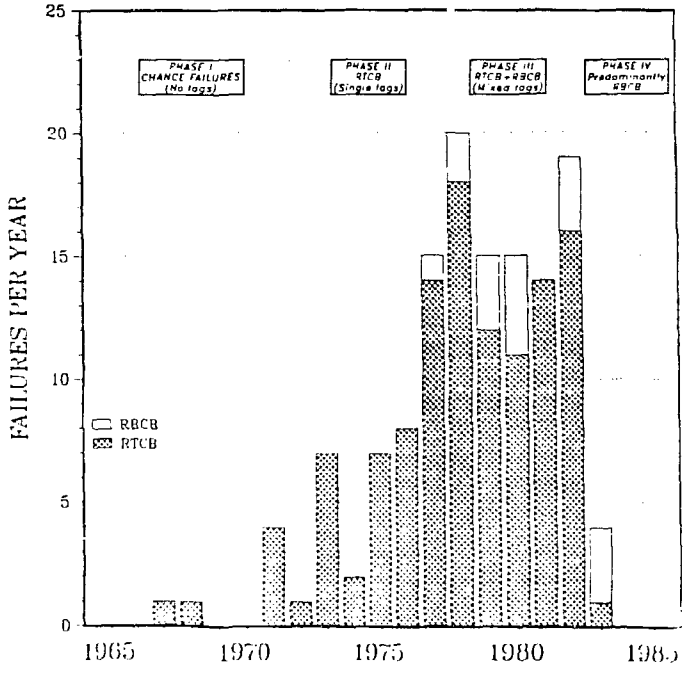

Fig.4
Failures in EBR-II Experimental Elements over 1967-1983

Normal Endurance Limits: When a high-burnup oxide element fails it typically produces a combined Xe-133/Xe-135 activity of ~6 μCi/mL in the argon cover gas. Until the CGCS was installed in 1977, leakage of this activity to the containment building dictated the practical endurance limit for elements in EBR-II. Mass-spectrometric identification of the xenon tag released by the failed element pinpointed its subassembly[10] which then had to be discharged. A secondary limit was the buildup of radioactive cesium in primary sodium: by late 1977 this rising activity necessitated the frequent changeout of aerosol filters, particularly in the fuel unloading machine. The problem was finally solved by installation of a carbon trap in the primary-sodium economizer. Used at shutdown, the trap rapidly reduced and maintained activities to pre-1974 values of about 20 nCi/gm.

Only in about twenty instances have failures been visually apparent in RTCB elements. Before 1978-79 these visible failures generally occurred on: (i) mixed-oxide elements that had either locally overheated or sustained local fretting wear of the cladding; (ii) U-Fs driver-fuel elements that had well exceeded their burnup limit; and (iii) carbide elements in which wedging of cracked fuel or embrittled cladding had had led to failure.[11] Breach sites in most failures were pinholes or hairline cracks of the cladding that were invisible during normal inspection. Invariably failures have been in the fuel column region, toward the upper hotter end where the cladding is weakest. In recent tests, in which more aggressive designs have been irradiated, fuel-cladding mechanical interaction has tended to produce obvious splitting of the cladding. Examples of typical RTCB failures are shown in Fig.5.

None of the RTCB elements has been observed to damage or cause premature failure of its neighbors, despite plenum pressures which exceeded 1000 psi. The occurrence of failure in the fuel-column region by generally small breaches clearly has not caused any significant pressure pulse. Although interpretations of the exact causes of failures have differed, the considerable evidence gained from about 125 RTCB failures has supported the view that fuel-failure propagation is an extremely unlikely event under normal reactor conditions. Although release of radioactive gas and cesium initially caused problems at EBR-II, these have been circumvented and cladding failure **per se** is not seen as a life limit to most types of LMR element.

Post-Failure Limits: After the CGCS and Cs trap had been installed continued operation with fuel-element failures became a practical proposition at EBR-II. Beginning in 1978 with tests of natural end-of-life failures to support CRBR licensing, and continuing from 1981 with a broader collaborative program with Japan, RBCB testing of mixed-oxide fuel elements has continued to the present time. Results of this testing have been reported elsewhere[2,4,12] and they are only briefly summarized here.

When the plenum inventory of a breached element has been released sodium may enter, contact fuel and begin to react with it to form $Na_3(UPu)O_4$ at the breach and elsewhere in the fuel-cladding gap. Both during and after this reaction DN signals readily indicate exposure of fuel to primary sodium. The reaction product produces local swelling that extends the initial breach; Fig.6 shows typical breaches after 5 days RBCB operation.

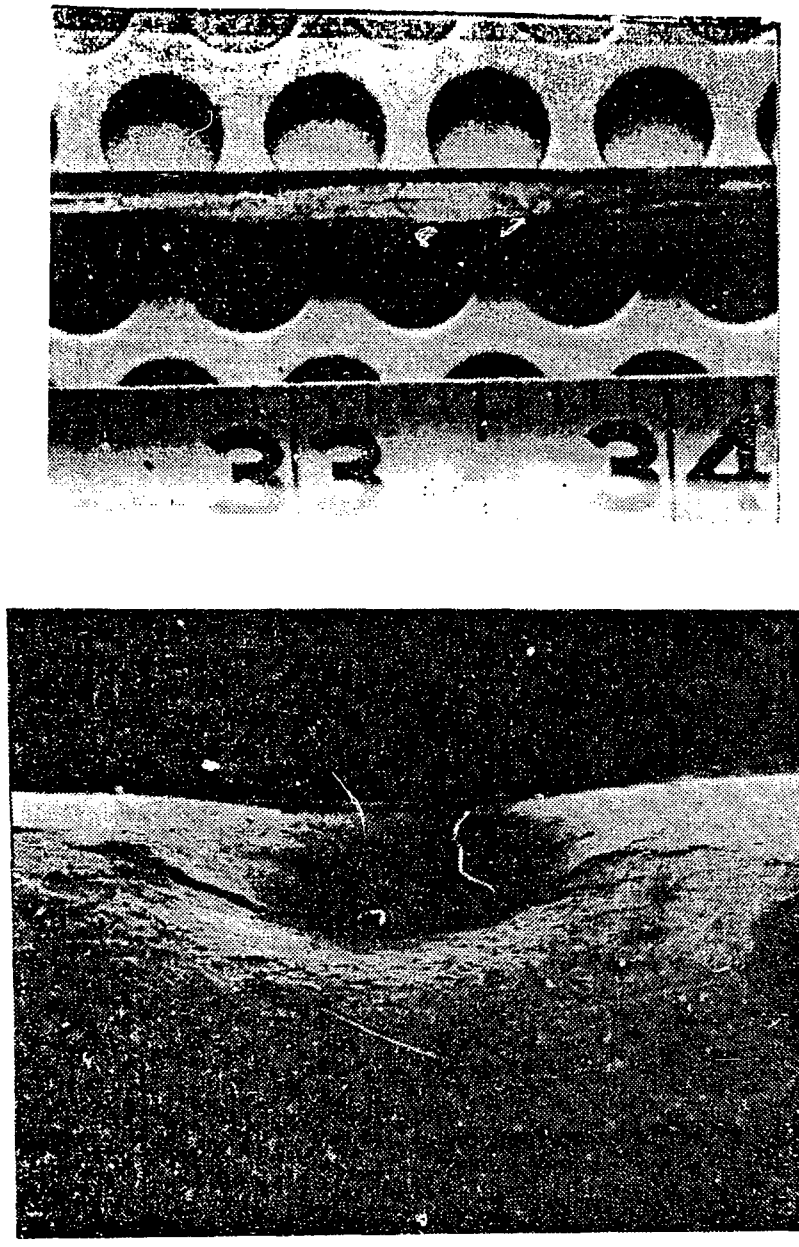

Fig.5
Examples of RTCB Element Failures in EBR-II
Upper: Failure in fretting-wear region on a high-burnup oxide pin
Lower: Cladding breach in the dimple region of a high-burnup U-Fs driver element

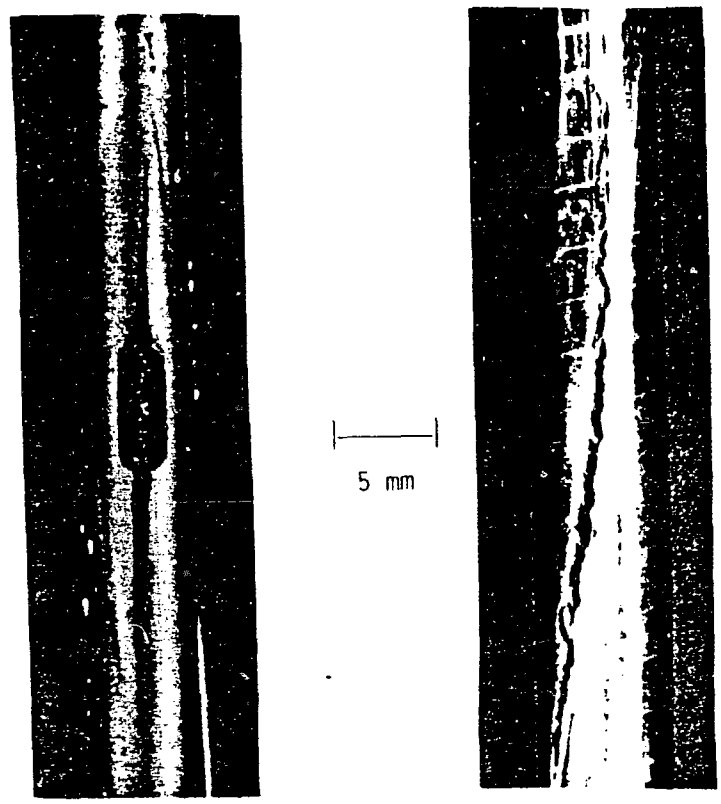

Fig.6
Typical breaches in RBCB tests in EBR-II
Left: At local prethinned region Right: End-of-life breach

Formation of the reaction product depends on time and temperature and is limited by the availability of sodium and oxygen and the product dissociation temperature of 1100-1150°C. Finally, however, a stable reaction product layer will form at the fuel surface to inhibit further reaction and splitting of the cladding. Fig.7 shows the morphology of the $Na_3(UPu)O_4$ reaction product at two stages of formation.

The fuel-sodium reaction product layer has two major effects on RBCB element behavior. First, the layer acts as a somewhat lower thermal impedance than the fuel it replaces and its formation robs oxygen from the fuel interior, which lowers slightly the effective fuel thermal conductivity. The nett effect on thermal performance, however, does appear to be serious.[13] Secondly, the layer seems to act in a plastic manner that stops any significant loss of fissile material: only very small amounts (milligrams) have been measured in special filters placed above test assemblies[4] in a special

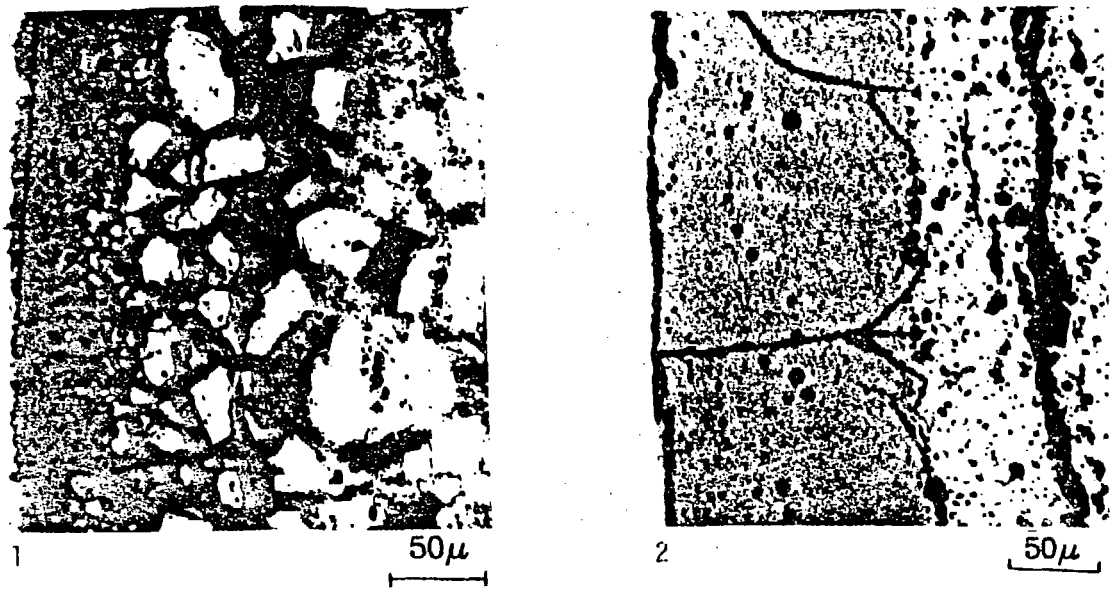

Fig.7
Appearance of the $Na_3(UPu)O_4$ Layer in RBCB Elements
Left: After 5 days RBCB Right: After 150 days RBCB

device known as the breached-fuel test facility (BFTF). These BFTF results have been encouraging, removing many of the worries about system contamination during RBCB operation. As with RTCB tests, major contaminants appear to be fission gases and cesium. A practical limit to RBCB operation at EBR-II (but not perhaps at other LMR's) is the Rb-88 activity that derives from short-lived Kr-88; this is not well controlled by the CGCS and can escape to the containment building. Operation with three concurrent RBCB elements appears to be the present limit.

SUMMARY

EBR-II has now been operated for 23 years as a true power plant. Although much of the last ten years has been devoted to aggressive modes of testing to determine fuel-element reliability, the availability factor of the plant has remained above 70%. This respectable record and the experience of component testing gained while it was being made suggests that the operability and availability of a liquid metal reactor are not substantially limited by the performance of core components.

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