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SUMMARY REPORT ON IRRADIATION OF PROTOTYPE EBR-II FUEL ELEMENTS

by

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ABSTRACT

This report summarizes the series of irradiations performed in the CP-5, MTR, and ETR on promising prototype EBR-II fuel elements. Type 304-clad uranium-zirconium and uranium-fissium alloy fuel pins were exposed to conditions approaching the anticipated parent core environment. Special facilities were designed to promote maximum fuel centerline temperatures ranging from 700°F to 1200°F, and fuel atom burnups from 0.4% to 1.56%. The exposure history and the effects of irradiation are described and illustrated photographically. These include dimensional and density changes, fission gas buildup, and fission product contamination of the sodium thermal bond, as a function of fuel burnup and operating temperature. The lesser physical changes evidenced by the stainless steel-clad uranium-fissium alloy support its superiority over the zirconium-uranium alloy, and ultimate selection for the initial core loading of the EBR-II.

I. INTRODUCTION

The Experimental Breeder Reactor-II (EBR-II), under construction at NRTS, Idaho, is a heterogeneous, enriched uranium-fueled, sodium-cooled reactor power plant designed to produce 20 Mw of electricity from 62.5 Mw of reactor heat in the form of 1300-psig steam. The plant includes an integral fuel-processing facility wherein the irradiated fuel is processed, fabricated, and assembled for return to the reactor.

The EBR-II is primarily an engineering facility to determine the feasibility of fast reactors for central station power plant application. Major emphasis has been placed on achieving high thermal performance at high temperatures, and high fuel burnup with a fast and economical fuel-processing system. The thermal performance of the reactor and the sizes of the system components are designed to permit direct extrapolation to large-scale central station power plants.

The proposed reactor consists of an enriched core surrounded by a fertile blanket of depleted uranium. The core section is comprised of hexagonal tubes containing small-diameter cylindrical pin assemblies.

The design of the pin assemblies is influenced by the desire for high thermal performance, high burnup, and simplicity of construction. The fuel pin is a loose fit in a thin-walled tube, and the resultant annulus is filled with static sodium to provide a heat transfer bond between the fuel pin and fuel tube. The heat generated within the fuel is ultimately removed by the primary sodium coolant flowing along the outside of the fuel tube.

The EBR-II probably will be the first reactor in the U.S. to operate directly on recycled fuel and, hence, to face the entire problem of heavy isotope buildup. The fuel element is amenable to reprocessing and fabrication by remote control methods as required by the pyrometallurgical oxidative drossing process selected. This method features simple process steps with uncomplicated equipment, but does not achieve complete removal of fission products. The ingot resulting from the melting operation contains uranium, plutonium, and the fission products Zr, Nb, Mo, Ru, Pd, and, presumably, Tc.

The name "fissium" has been applied to all alloys of uranium, plutonium, and fission products arising out of the pyrometallurgical process. The concentration of these elements increases with each fuel recycle until an equilibrium value is reached. This equilibrium value is a function of the fission yield of the isotopes involved, half-life, cooling time, cross section, and dragout via processing losses. As a result, there are many possible equilibrium alloys, depending upon: the ratios of the fissionable materials (U^{235}, U^{238}, Pu^{239}), on process operating conditions, on reactor cycle, etc. Certain of the variables are adjustable so that some control can be exercised over the composition of the recycled alloy.

However, it is apparent that between pure uranium and the equilibrium fuel of an infinite number of recycles, there exists an infinite number of alloys. As a result, it is planned to load the reactor with an alloy approaching one of the equilibrium alloys. The change in composition per pass in this case is slight and changes in properties are expected to be negligible. This plan alleviates the effects of the ingrowth of all fission products except technetium, but still leaves those problems arising from heavy element buildup. It is anticipated that the equilibrium alloy resulting from the operation of EBR-II with a uranium fuel loading may be as follows:

Element	wt-%
Zirconium	0.1-0.2
Niobium	0.01
Molybdenum	1.6-3.4
Technetium	0.5-1.0
Ruthenium	1.2-2.6
Rhodium	0.2-0.5
Palladium	0.1-0.3
Silver + Cadmium + Antimony	0.1

Accordingly, certain prototype fuel elements containing this equilibrium alloy were fabricated and irradiated in reactors operating at conditions simulating, as nearly as possible, the operating conditions anticipated in the EBR-II. The conditions of operation in EBR-II are such that the fuel will operate a central metal temperature of 1200°F, and will accumulate a burnup of 2% of total atoms before being unloaded from the reactor.

The general objective of the irradiation test program was to determine the behavior of these prototype fuel elements first at burnups up to 1%, and then up to 2% of total atoms. Minor variations in the fissium composition were evaluated, and similar tests were conducted on 2% zirconium-uranium alloys to provide a basis for comparison. The irradiation program can be considered as a "go - no go" evaluation, since the primary objective was to establish the feasibility of the fissium alloy.

II. FABRICATION OF IRRADIATION TEST FUEL PINS

The reference design EBR-II fuel element (Figure 1) is composed of a right circular cylinder (pin) of fuel alloy (0.144 in. in diameter by 14.22 in. long) which is enclosed in a stainless steel tube (0.174-in. OD; 0.009-in. wall thickness). The resultant annulus between the pin and the inside diameter of the tube (0.006 in.) is filled with static sodium to provide a thermal bond. The sodium bond extends a nominal 0.6 in. above the top of the fuel pin. An inert gas space (2.35 in.) is provided above the sodium to accommodate expansion of the sodium and accumulation of fission gases. The restrainer is a vertical rod with pedestal positioned to prevent axial growth of the fuel alloy beyond 2% of its original length. The center volume provides a void for collection of fission product gases. End fittings are welded at each end of the tube to effect a seal. A helical wire spacer, tack welded at the ends, is wound around the periphery of the tube to complete the element.

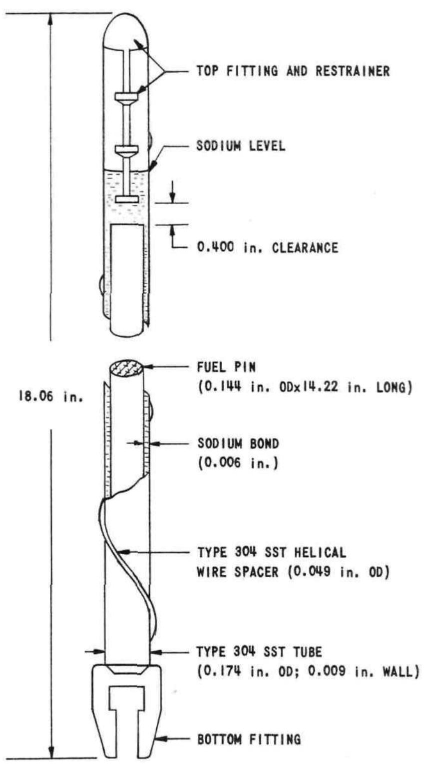


FIG. 1
REFERENCE DESIGN EBR-II FUEL ELEMENT

The compositions of the respective fuel alloys used to fabricate the test fuel pins are listed in Table I. The fissium alloy compositions were evolved from a series of prototype EBR-II pyrometallurgical

TABLE I

Designation and Composition of Alloys Irradiated

Alloy Designation	Composition, %						
	U	Zr	Mo	Ru	Rh	Pd	Nb
U-Zr	98	2					
3% Fissium	96.9	0.1	1.54	1.16	0.19	0.1	0.01
4% Fissium	96	0.16	2	1.2	0.24	0.4	
5% Fissium	95	0.2	2.5	1.5	0.3	0.5	
7.5% Fissium	92.49	2.56	2.48	1.97	0.3	0.2	

drossing operations.[1] The selection of the specific fissium alloys listed in Table I was premised on their remarkably good behavior during initial screening tests. The scope of these tests included laboratory thermal-cycling studies,[2] and subsequent short-term irradiations.

The fabrication history of each fuel pin is summarized in Table II. The uranium-zirconium specimens were originally cast about 30% over-size and then centerless ground or swaged to finish size. Specimens designated "wrought and heat treated" were swaged to size at 800°C and cooled in air. Two specimens were coextruded with a 0.005-in. zirconium clad. Full-length fissium alloy pins were precision cast to size in Vycor tubes. The final diameter of all specimens was 0.144 in., the reference diameter of the EBR-II fuel pins. The length of each specimen was 14.22 in.

III. IRRADIATION TEST FACILITIES AND PROCEDURES

A. Facilities

The irradiations were carried out in three thermal reactors: the Argonne Research Reactor (CP-5) at Argonne, Illinois; and the Materials Testing Reactor (MTR) and Engineering Test Reactor (ETR) at NRTS, Idaho. The majority of the irradiations was performed in CP-5. While it is recognized that the EBR-II is a fast reactor and that irradiations in a thermal flux will reflect a consequent deviation in burnup profile, the difference is not considered serious in elements of such small dimensions.

1. CP-5

Two facilities were designed primarily for irradiation of EBR-II fuel pins in the CP-5 reactor.[3] Both facilities featured variations of the same basic system, dictated by their location in the reactor and the method employed to remove the heat generated within the test specimen.

TABLE II

Fabrication History of Irradiation Test Fuel Pins

Test No.	Fuel Alloy Composition, %							U^{235} Enrichment, %	Mode of Fabrication
	U	Zr	Mo	Ru	Rh	Pd	Nb		
CP-5-1	98.7 98.0	1.3 2.0						15 15	Chill cast. Two slugs with 1.3% Zr; and five slugs with 2% Zr.
CP-5-2	98.0	2.0						17	
CP-5-3	98.0	2.0						20	Wrought and heat treated.
CP-5-4	98.0	2.0						17	
CP-5-5	98.0	2.0						20	Cast. Three sections butt-welded to one section
CP-5-6	98.0	2.0						20	Wrought and heat treated.
CP-5-7	96.0	0.16	2.0	1.2	0.24	0.4		20	
CP-5-8	96.9	0.1	1.54	1.16	0.19	0.1	0.01	20	Cast fissium.
CP-5-9	95.0	0.2	2.5	1.5	0.3	0.5		30	
CP-5-11	98.0	2.0						11	Coextruded with 4-mil Zr clad
CP-5-12	98.0	2.0						11	
CP-5-13	95.0	0.1	2.5	1.5	0.3	0.5		30	Cast fissium.
CP-5-14	95.0	0.1	2.5	1.5	0.3	0.5		20	Cast fissium. Cut into three pieces. Monitor wire attached to center piece.
CP-5-15								20	
CP-5-16								20	
CP-5-17								20	
CP-5-18	95.0	0.1	2.5	1.5	0.3	0.5		20	Cast Fissium
ANL-6-68-1 (MTR-3)								5	
ANL-6-70-2 (MTR-2)								5	
ANL-6-71-3 (MTR-1)								5	
ANL-40-1 (ETR-1)	92.53	2.5	2.5	1.97	0.3	0.2		10.5	Cast Fissium
ANL-40-2 (ETR-2) ANL-40-3 (ETR-3)	93.0	2.5	2.5	1.5	0.3	0.2		10.5	Cast Fissium

The initial irradiations were performed in the air-cooled facility shown in Figure 2. This facility was installed in one of the central aluminum thimbles in CP-5. The thimble was modified to provide for the cycling of air down through the inner aluminum tube assembly, around the test section, up through the annulus betweeen the aluminum tube and the thimble walls, and out to the existing reactor air-filter and exhaust system. A suction pressure of 14 in. Hg was maintained by a Roots-Connersville blower to produce the air flow required to cool a fuel element generating approximately 5 kw of heat.

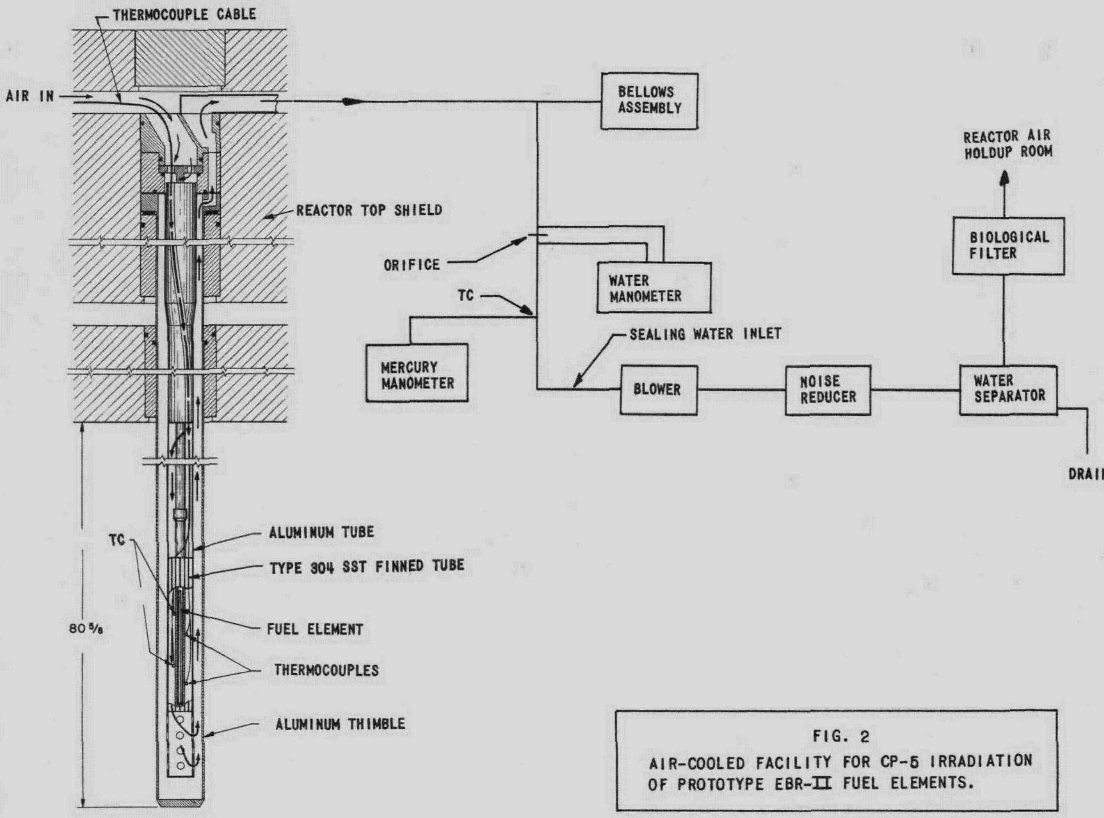
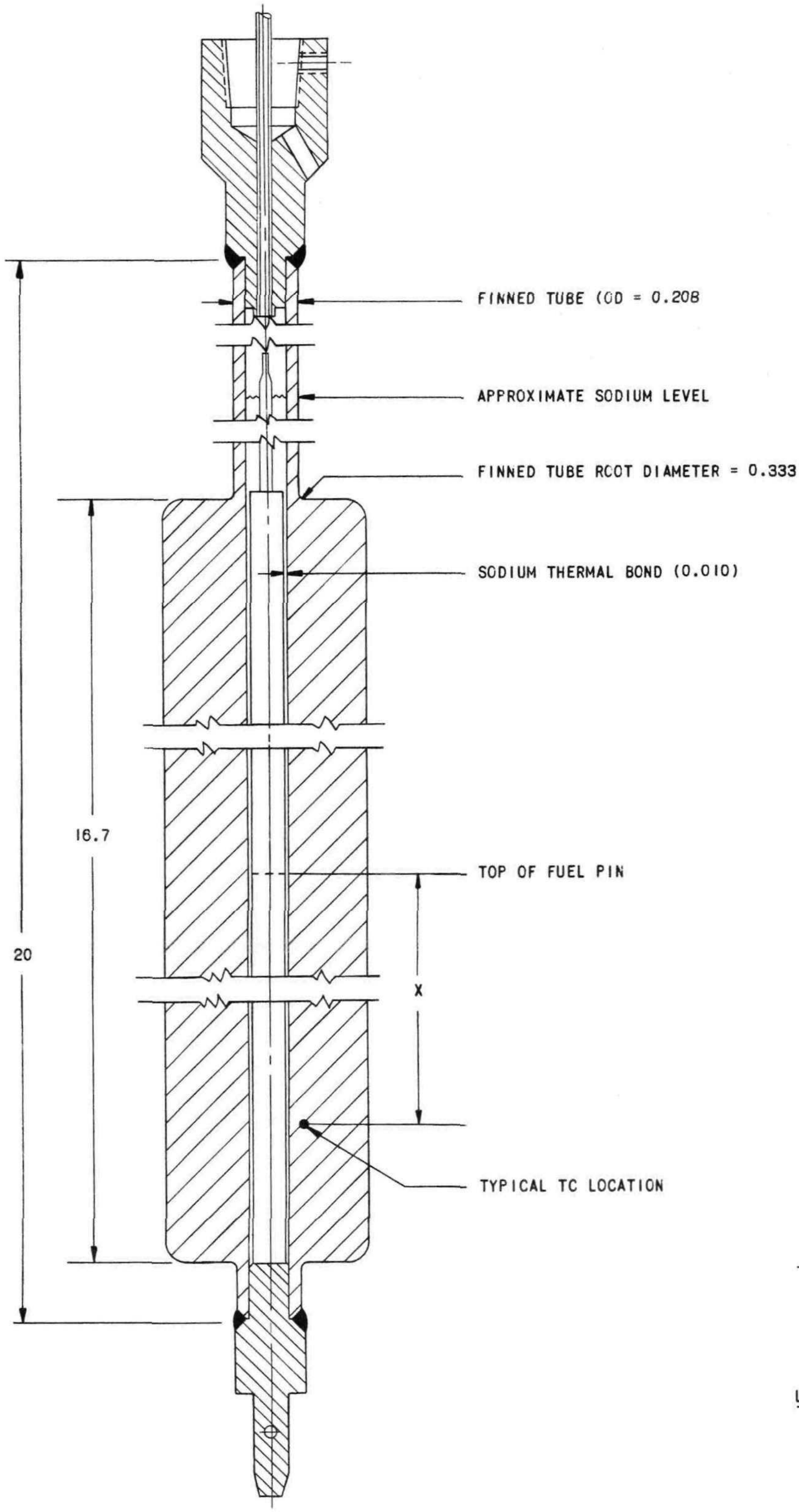


FIG. 2
AIR-COOLED FACILITY FOR CP-5 IRRADIATION OF PROTOTYPE EBR-II FUEL ELEMENTS.

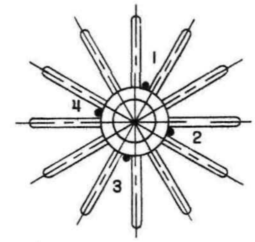
The test section (Figure 3) comprised a fuel element encapsuled within, and sodium bonded to, a Type 304 finned tube.

After installation of the fuel element within the finned tube, eight thermocouples were welded to the tube surface (in the root between fins) to permit continuous recording of temperature profiles pertinent to the fuel element and capsule. Thermocouples were also installed to permit recording the cooling air temperature immediately upstream and downstream of the finned tube assembly.

The balance of the instrumentation external to the reactor allowed measurement of air flow, temperature, and pressure. The air



TC NO.	X, in.
1	2.5
2	5.5
3	7.5
4	10.0



LOCATION OF THERMOCOUPLES

FIG. 3
CP-5 IRRADIATION TEST ASSEMBLY OF EBR-II
PIN-TYPE FUEL ELEMENT WITHIN FINNED TUBE.

flow was measured by a water manometer which bridged an orifice in the exhaust piping upstream of the blower. Air temperature and pressure were measured by means of a thermocouple and a mercury manometer installed downstream of the orifice.

The facility was designed to operate unattended; hence safety circuits were incorporated to protect the experiment in the event of equipment malfunction. Two of the eight thermocouples attached to the finned tube were interconnected to temperature limitrols which operated to scram the reactor in the event preset high or low temperatures were exceeded. A bellows assembly was installed in the exhaust line from the thimble. The bellows was designed to expand upon loss of vacuum (due to loss of air flow) and trip a microswitch to cause reactor shutdown. Another switch was interconnected with the reactor safety shutdown system. In the event of a reactor shutdown, this switch immediately shut off the blower fan in the test facility. This precaution was taken to prevent a rapid temperature drop across the test section that would be promoted by the combined loss of heat generation in the fuel pin and continued flow of coolant air. The switch also featured a 3-min. time delay mechanism to prevent the resumption of blower (and reactor) operation until the sample had cooled sufficiently.

The second facility (Figure 4) was designed for installation in one of the experimental thimbles located at the periphery of the reactor core. In brief, the encapsulated fuel specimen was suspended by a stainless steel tube within a pressure vessel partially filled with heavy water. The heavy water served as the capsule coolant during irradiation, and as shielding during unloading operations. Heavy water was used in lieu of light water to prevent contamination of the reactor heavy water moderator in the event of a leak in the pressure vessel.

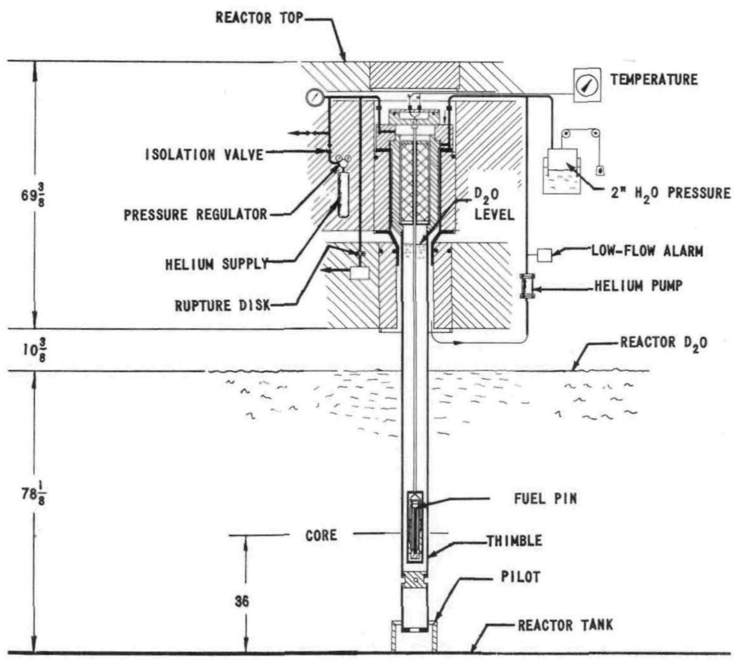


FIG. 4
EBR-II FUEL PIN IRRADIATION FACILITY INSTALLED IN CP-5 EXPERIMENTAL THIMBLE.

The fuel capsule is shown in Figure 5. The fuel pin was thermally bonded with sodium to the inner surface of a stainless steel tube. Four

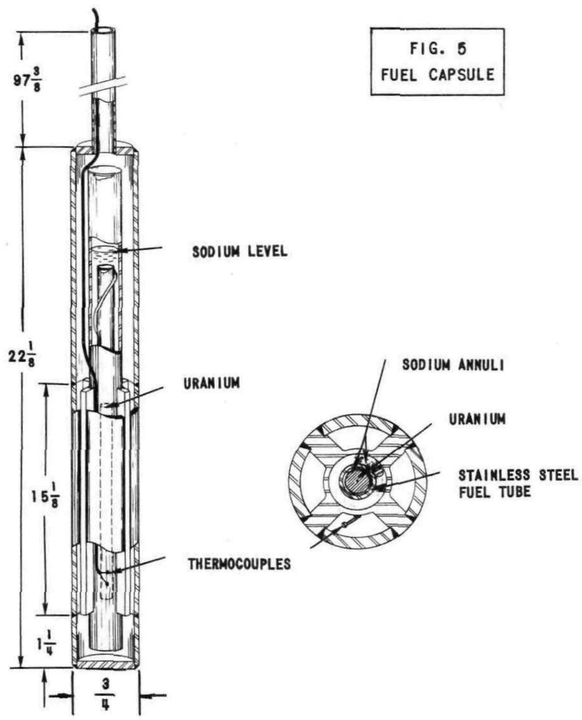


FIG. 5
FUEL CAPSULE

longitudinal fins were welded to the outer surface of the tube, and to cylindrical segments. The segments, in turn, were welded to form a cylindrical capsule (0.75 in. OD by $22\frac{1}{8}$ in. long). The intervening space between the finned tube and capsule contained air and thus presented a relatively high-resistance path for heat conduction. The path for heat flow was from the fuel pin, through the radial fins, to the outer surface of the fuel capsule. The capsule was immersed in the heavy water (in the pressure vessel), and local boiling occurred on the outer surface. The degree of local boiling, and hence, the fuel temperature, was controlled by pressurizing the heavy water in the pressure vessel. The reactor heavy water moderator, in which the pressure vessel was suspended, provided the final heat sink for the heat generated by the fuel specimen during irradiation.

The pressure vessel was a stainless steel tube (1.25 in. or 1.50 in. OD; 0.065 in. wall) welded to a modified thimble shielding plug, and designed for 1100 psig at 600°F. The vessel was partially filled with D_2O to a depth of 5 ft above the reactor core, and was pressurized with helium from a high-pressure container. The operating range of pressure was from 150 to 800 psig. The maximum operating pressure was set 300 psi below design pressure to prevent premature failure of the rupture disk which protected the system. Rupture disks were selected over safety valves in order to improve the gastight integrity of the system. Other measures designed to ensure a gastight pressure system included: (1) an all-welded vessel with a flanged closure sealed with rubber "O" rings; (2) flare-type fittings in the system tubing (copper: 0.25 in. OD; 0.065 in. wall); and (3) the use of rubber "O" rings for valve seats and stem packing.

In the event of a disk failure, the helium with some entrained steam would discharge to a shielded container (designed to condense the steam) which exhausted to the reactor active vent system. To prevent over-pressurization of the vessel in the event of failure of the pressure regulator, the vessel was charged with helium during reactor operation and then isolated from the gas supply.

The minimum pressure (150 psig) was established by laboratory tests which indicated that burnout of the capsule occurred at 50 psig and an operating power level of 10 kw. A safety factor of 100 psi was added. Further, a minimum pressure must be maintained to ensure a pressure differential between water in the vessel and the exterior coolant so that during operation the necessary temperature gradient can be established for outward flow of heat.

The surface temperature of the capsule was controlled by varying the pressure and thus the saturation temperature of the water in the vessel. Variation of the pressure from 150 to 180 psig permitted control of the central metal temperature of the fuel specimen over a range of 150°F. The generated heat and temperature of the fuel were equivalent to values estimated for fuel elements during operation in the EBR-II. However, since the temperature was controlled at the outer capsule surface, which was at a relatively low temperature (~450°F), the fins were sized to provide the thermal resistance necessary to raise the central metal temperature to the desired operating level (1100 to 1200°F). With the known values of thermal resistance and the temperature difference Δt from the measuring point (fin root) to capsule surface, the heat generation (and thus the burnup) and central metal temperature was calculated.

The instrumentation consisted of a multipoint temperature recorded equipped with a high-temperature scram, and a pressure recorder which featured a "high" and a "low" scram.

2. MTR and ETR

During the early stages of the irradiation test program in CP-5 it became apparent that the specimen EBR-II fuel pins would have to be exposed to a much higher neutron flux in order to sustain the desired high burnups within a reasonable time. Accordingly, three pins were prepared for exposure in the water channel at MTR, and three pins were preparedfor exposure in the in-pile test facility at ETR.

The construction of the irradiation test facilities in both reactors precluded the installation of thermocouples to register the fuel pin temperatures. Therefore, the irradiation test capsule simply comprised a fuel pin surrounded by a nickel-cobalt wire foil. This assembly was installed within and sodium bonded to a Type 347 stainless steel tube (23.625 in. long; 0.500-in. OD; 0.085-in. wall thickness) with welded and plug closures. The sodium bond extended 1.5 in. above the top of the fuel pin. The height of the sodium level provided sufficient space for expansion at temperature.

The capsule was positioned vertically in the reactor test facility at a point where the neutron flux was predetermined as sufficient to generate the desired heat and consequent temperature in the fuel specimen. The heat was removed by the reactor coolant flowing along the outer surface of the capsule. The nickel-cobalt wires were used subsequently to determine the total integrated flux sustained by each pin. The flux pattern was used in conjunction with the fuel enrichment to calculate the average operating temperatures.

B. Procedures

Each irradiation in CP-5 was prefaced by an axial flux distribution measurement (with bare and cadmium-covered gold foils) of the empty test hole in the reactor. This was done to establish the vertical position of the fuel pin in a flux region that would give the desired maximum central metal temperature. The capsule was then installed in the test hole, and the reactor was brought up to power.

The temperatures registered by the thermocouples were observed continuously during the startup procedure. In the event of erratic behavior of the thermocouples, the reactor was shut down immediately and the capsule was removed. In each such instance, subsequent examination of the capsule revealed a failure of the test specimen; and in each instance the failure was attributable to faulty bonding of the fuel pin to the inner wall of the capsule.

These observations led to the adoption of the following procedure: After loading the specimen, the reactor power was increased, stepwise, to the desired operating level. Provided no malfunction of the thermocouples was observed during the first twelve hours of operation, the irradiation was adjudged to be progressing satisfactorily. The operating data listed in Table III for Capsule No. CP-5-1 are typical of each subsequent irradiation.

TABLE III

Operating Conditions for Irradiation Capsule No. CP-5-1

Total sensible heat generation	5.0 kw
Cooling air flow rate	76.3 cfm
Inlet air temperature	70.0°F
Rise in air temperature	345°F
Average heat flux at OD of fuel pin	300,000 Btu/(hr)(ft^2)
Maximum surface temperature of fuel alloy	1040°F
Maximum center line temperature of fuel alloy	1100°F

The temperature distribution along the fuel pin is a resultant of the effects produced by the vertical flux gradient along the test facility, and the method of cooling employed. Figure 6 shows the central metal temperature gradient for Capsule No. CP-5-11 irradiated in the air-cooled

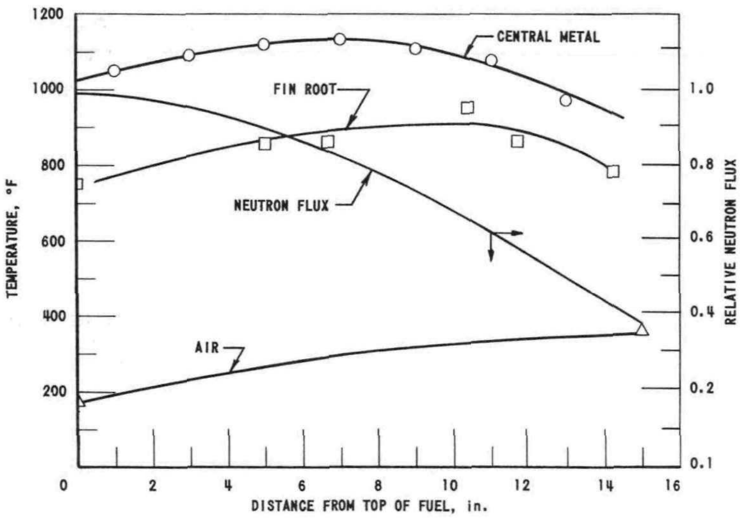


FIG. 6
AXIAL NEUTRON FLUX AND TEMPERATURE DISTRIBUTIONS FOR CAPSULE NO. CP-5-11.

facility. Despite the decrease in neutron flux from top to bottom, the central metal temperature of the fuel pin actually increases at the middle, and then decreases slightly at the bottom of the fuel pin. This effect is produced by the rise in temperature of the cooling air which was directed downward around the finned capsule.

The power generation in fuel pins containing ~70 gm uranium metal varied from a low of 2.16 kw for Capsule No. CP-5-8 to a high of 6 kw for Capsule No. CP-5-11. In addition, the fuel pins were subjected to a large number of thermal cycles as a consequence of frequent, unavoidable shutdown of the CP-5 reactor. The abrupt discontinuance of power generation in the fuel pin resulted in a rapid drop from operating to reactor ambient temperature (~175°F). However, despite the number of thermal cycles experienced by each fuel pin (~66 cycles/month), subsequent examination revealed that none was warped significantly.

Several fuel pins irradiated in the MTR and the ETR suffered cladding failures near the top of the fuel. The nature of the failures supports the theory that they were promoted by reactor shutdowns, which permitted the capsule sodium to freeze and, hence, to expose the upper portion of the fuel pin. Subsequent startups resulted in excessive fuel temperatures with consequent deterioration of the cladding until an effective sodium heat transfer bond was re-established.

IV. POSTIRRADIATION PROCEDURES AND RESULTS

Upon completion of irradiation, each capsule was withdrawn from the reactor into a shielded cask and transferred to a high-level cave for examination and evaluation.

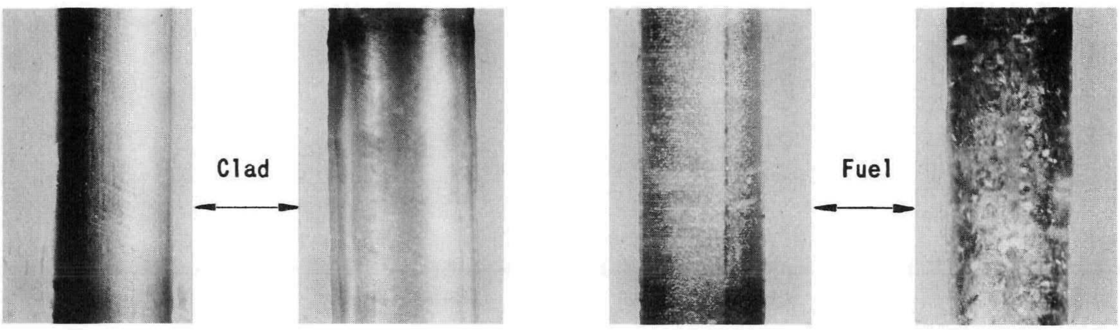
Normally, the fuel pin could be removed by (1) cutting off one end of the capsule with a remotely operated saw; (2) heating the capsule to melt the sodium bond; and (3) inverting and tapping the capsule. However, in instances of fuel pin failures, removal of the pin could be effected only by slitting the capsule longitudinally with a saw operated by a milling machine.

The fuel pin was then washed with water to remove any adherent sodium, and the cladding examined carefully with a magnifying periscope. All failures, defects, and other visible changes in the cladding were noted and photographed. Subsequent evaluations of each pin included: (1) total neutron flux dosage; (2) fission gas release; (3) dimensional and density changes; (4) burnup of fuel; and (5) fission product contamination of the sodium bond. The pertinent data are listed in Table IV.

Table IV

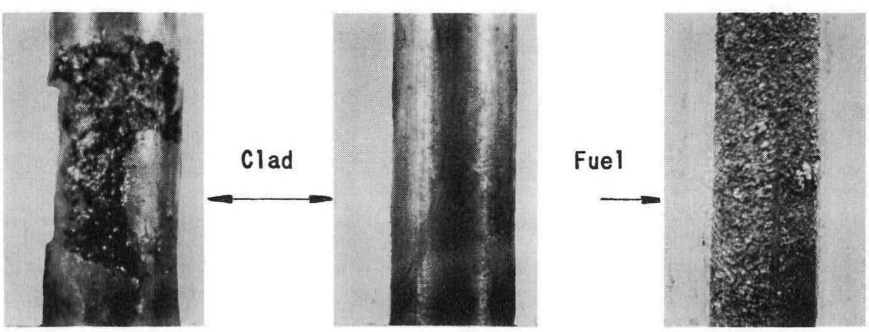
SUMMARY OF EBR-II FUEL PIN IRRADIATIONS

Fuel Alloy and History	Average Density, gm/cc Before	Average Density, gm/cc After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
			CAPSULE NO. CP-5-1			
2% Zr-U; 1.4% Zr-U; cast	17.290	16.423	183	1050	0.40	5.0



COMMENTS: Fuel loaded in seven 2-inch slugs. Bumpy surface observed on two bottom slugs containing 1.4% Zr.

Fuel Alloy and History	Average Density, gm/cc Before	Average Density, gm/cc After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
			CAPSULE NO. CP-5-2			
2% Zr-U; wrought and heat treated	--	--	212	879	0.43	5.5

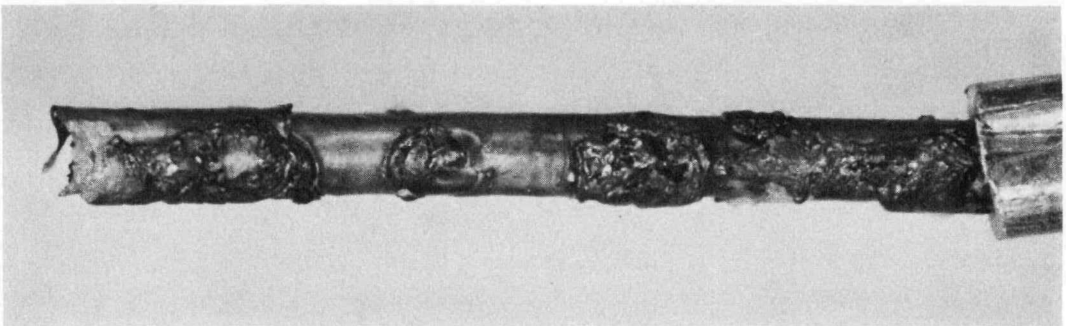


COMMENTS: Melt through upper portion of clad attributed to void in capsule sodium bond.

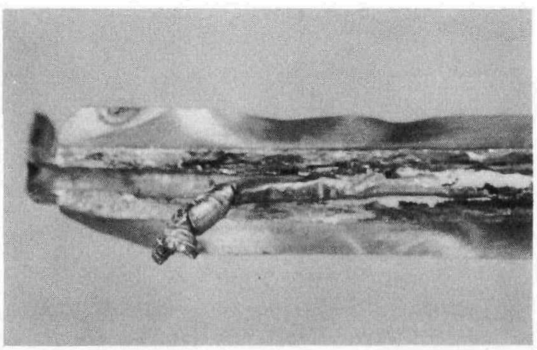
Table IV (Cont'd.)

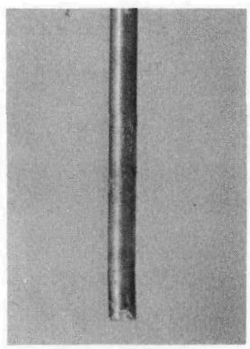
SUMMARY OF EBR-II FUEL PIN IRRADIATIONS

Fuel Alloy and History	Average Density, gm/cc Before	After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
			CAPSULE NO. CP-5-3			
2% Zr-U; wrought and heat treated	--	--	268	910	0.74	6.2



Fuel Alloy and History	Average Density, gm/cc Before	After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
			CAPSULE NO. CP-5-4			
2% Zr-U; wrought and heat treated	18.220	15.946	294	850	0.73	5.4



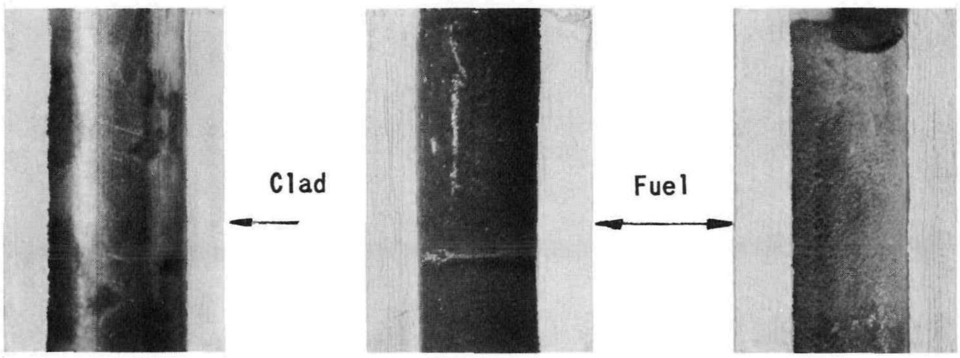


COMMENTS: Melting failure occurred at top of element due to faulty capsule sodium bond. Bottom section was reasonably sound.

Table IV (Cont'd.)

SUMMARY OF EBR-II FUEL PIN IRRADIATIONS

Fuel Alloy and History	Average Density, gm/cc Before	Average Density, gm/cc After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
CAPSULE NO. CP-5-5						
2% Zr-U; cast, butt welded sections	18.180	17.860	219	810	0.77	5.2



COMMENTS: Element and bond in sound condition. Surface of fuel roughened with many small pits. Fuel ruptured in one area by sub-surface casting defect.

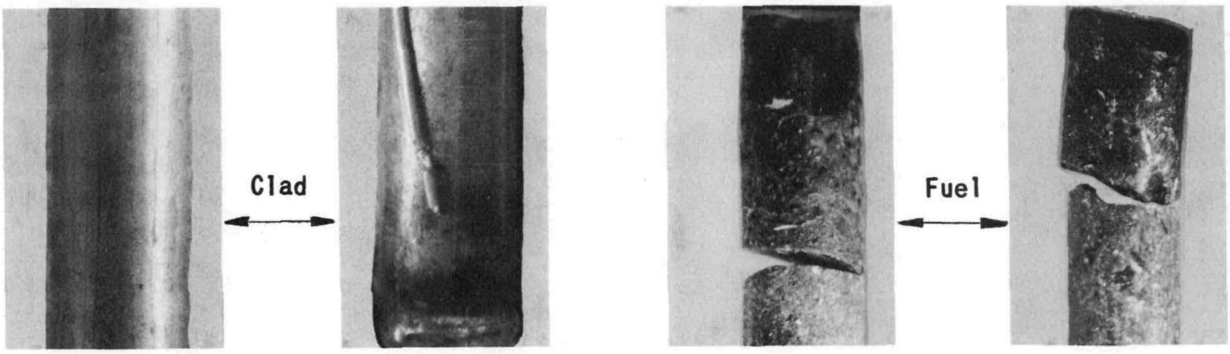
Fuel Alloy and History	Average Density, gm/cc Before	Average Density, gm/cc After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
CAPSULE NO. CP-5-6						
2% Zr-U; wrought and heat treated	18.230	--	--	825	0.65	5.5

COMMENTS: Fuel melted due to inadequate sodium bond. Element stuck in capsule; not decanned.

Table IV (Cont'd)

SUMMARY OF EBR-II FUEL PIN IRRADIATIONS

Fuel Alloy and History	Average Density, gm/cc Before	After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
			CAPSULE NO. CP-5-7			
4% fissium; cast.	18.017	17.550	352	750	0.60	2.9



COMMENTS: Alloy extremely brittle. Fuel fractured considerably during decanning operations. Surface of fuel coarse and wrinkled.

Fuel Alloy and History	Average Density, gm/cc Before	After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
			CAPSULE NO. CP-5-8			
3% fissium; cast.	18.238	18.080	271	698	0.50	2.2

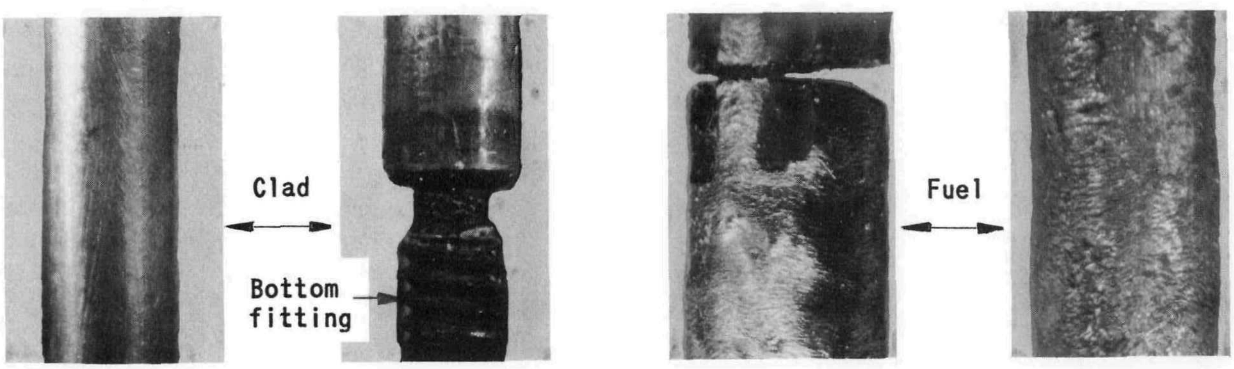
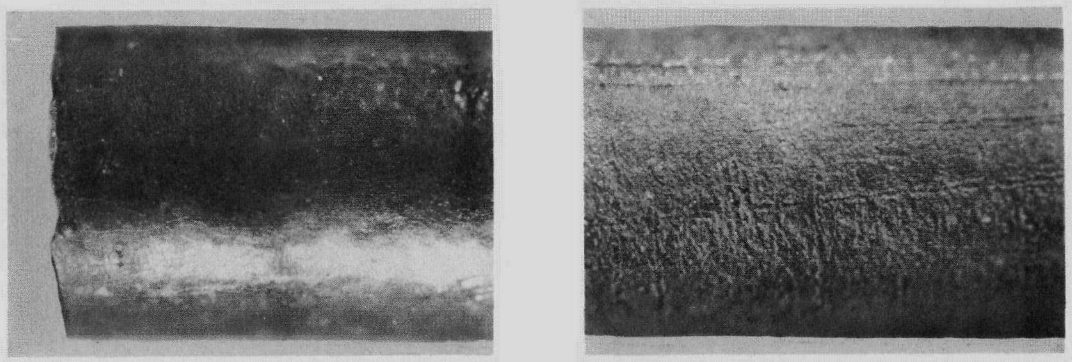


Table IV (Cont'd.)

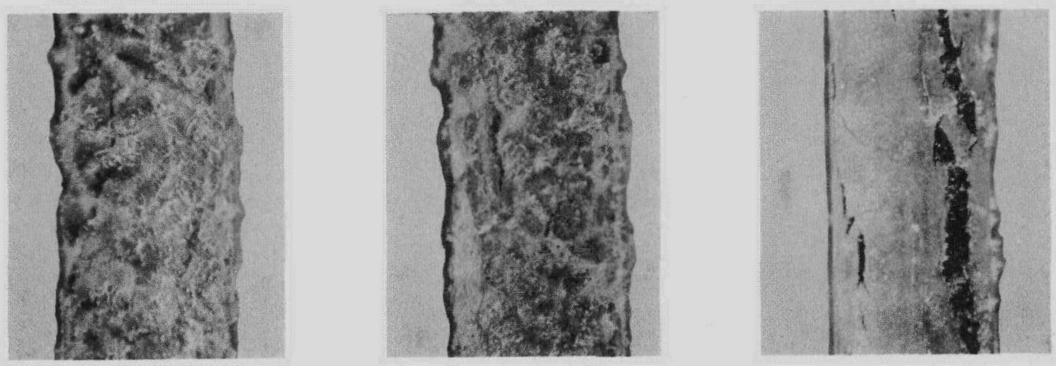
SUMMARY OF EBR-II FUEL PIN IRRADIATIONS

Fuel Alloy and History	Average Density, gm/cc Before	After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
			CAPSULE NO. CP-5-9			
5% fissium; cast.	17.981	15.805	394	1150	0.92	3.9



COMMENTS: Roughness of fuel surface more uniform, not as irregular as in Capsules NO. CP-5-7 and CP-5-8.

Fuel Alloy and History	Average Density, gm/cc Before	After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
			CAPSULE NO. CP-5-11			
2% Zr-U; coextruded with 4-mil Zr clad	--	--	289	1130	0.70	6.0

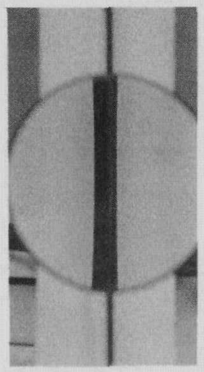


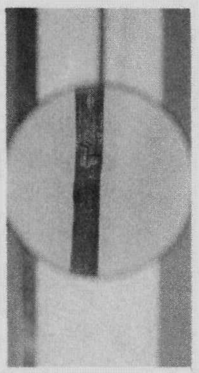
COMMENTS: Upper portion (~ I in. long) of element melted where fuel had extended above the top of the capsule cooling fin area. Zirconium clad and underlying uranium badly fissured, with pronounced diametral expansion.

Table IV (Cont'd.)

SUMMARY OF EBR-II FUEL PIN IRRADIATIONS

Fuel Alloy and History	Average Density, gm/cc Before	After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
			CAPSULE NO. CP-5-12			
2% Zr-U; coextruded with 4-mil Zr clad	--	--	196	800	0.50	3.2





COMMENTS: Over-all appearance of element good. Small rupture of clad promoted by extrusion defect.

CAPSULE NO. CP-5-13

COMMENTS: Irradiation of cast 5% fissium alloy pin terminated during approach to power. Pin melted due to insufficient capsule sodium bond.

CAPSULE NO. CP-5-14

COMMENTS: Irradiation of cast 5% fissium alloy pin terminated upon detection of water leak into thermocouple chamber.

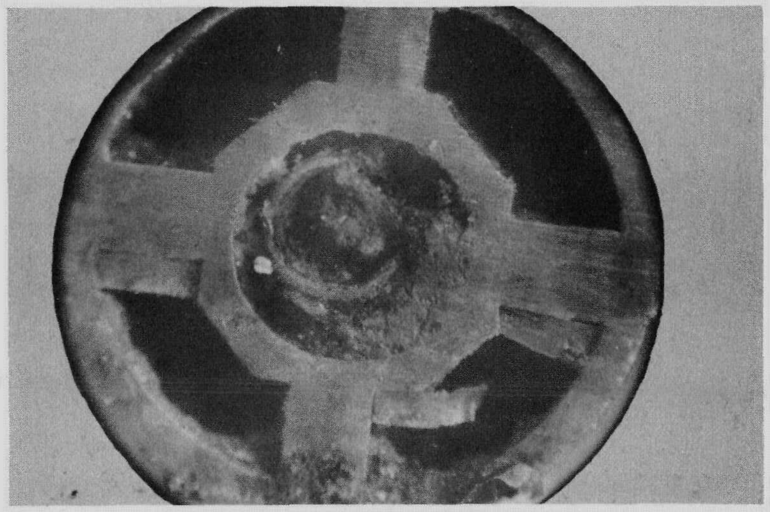
Table IV (Cont'd.)

SUMMARY OF EBR-II FUEL PIN IRRADIATIONS

Fuel Alloy and History	Average Density, gm/cc Before	After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw

CAPSULE NO. CP-5-15

5% fissium; cast



COMMENTS: Element badly damaged and stuck in capsule. Temperature data indicated that element had moved up and out of effective cooling (finned) region of capsule.

CAPSULE NO. CP-5-16

Fuel Alloy and History	Before	After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
5% fissium; cast.	18.070	--	349	925	0.84	3.8

COMMENTS: Element was removed from capsule but was not declad. Over-all appearance was good.

Table IV (Cont'd.)

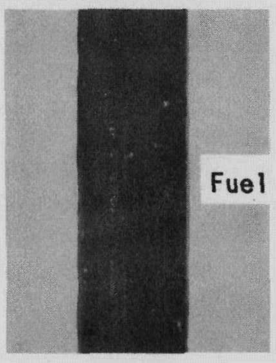
SUMMARY OF EBR-II FUEL PIN IRRADIATIONS

Fuel Alloy and History	Average Density, gm/cc Before	Average Density, gm/cc After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
CAPSULE NO. CP-5-17						

COMMENTS: Irradiation of cast 5% fissium alloy pin terminated upon detection of water leak into thermocouple chamber.

CAPSULE NO. CP-5-18

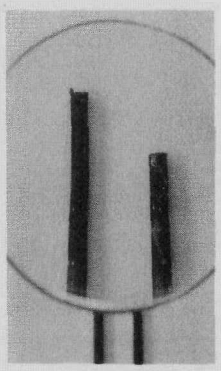
Fuel Alloy and History	Before	After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
5% fissium; cast.	18.110	16.780	162	1150	0.54	4.2



COMMENTS: Element was in sound condition, without any marked defects visible on the clad or the fuel.

CAPSULE NO. MTR-1

Fuel Alloy and History	Before	After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
5% fissium; cast.	--	--	--	423	0.57	4.0

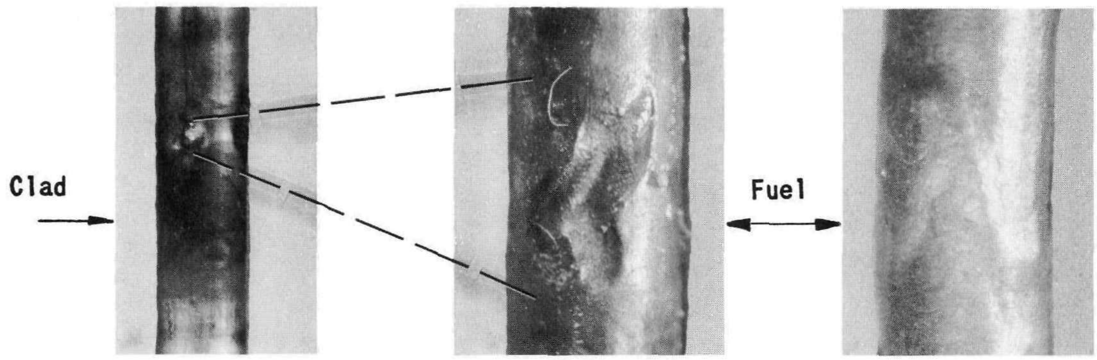


COMMENTS: Element was in good condition. Cladding was not removed.

Table IV (Cont'd.)

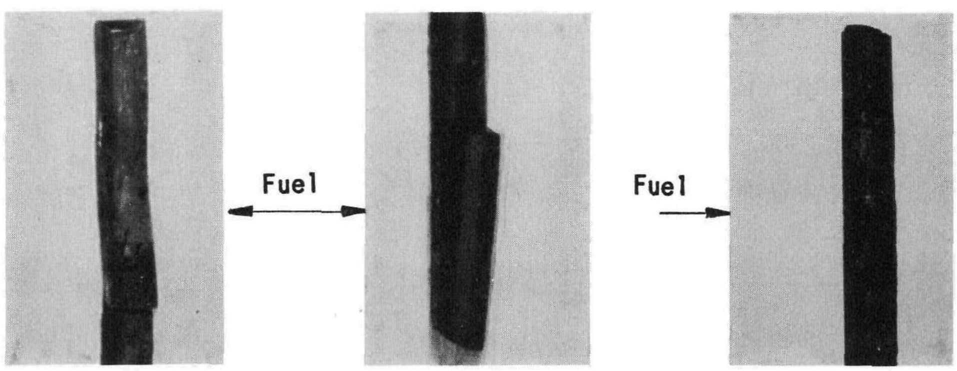
SUMMARY OF EBR-II FUEL PIN IRRADIATIONS

Fuel Alloy and History	Average Density, gm/cc Before	After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
CAPSULE NO. MTR-3						
5% fissium; cast.	**18.080**	**17.360**	**--**	**794**	**1.24**	**9.1**



COMMENTS: **Over-all appearance of element good. Small melt through clad 1/2 in. below top of fuel. Severe temperature in this region evidenced by discoloration of clad and appearance of underlying fuel.**

CAPSULE NO. ETR-1						
7.5% fissium; cast.	**17.490**	**16.900**	**--**	**932**	**0.71**	**9.3**

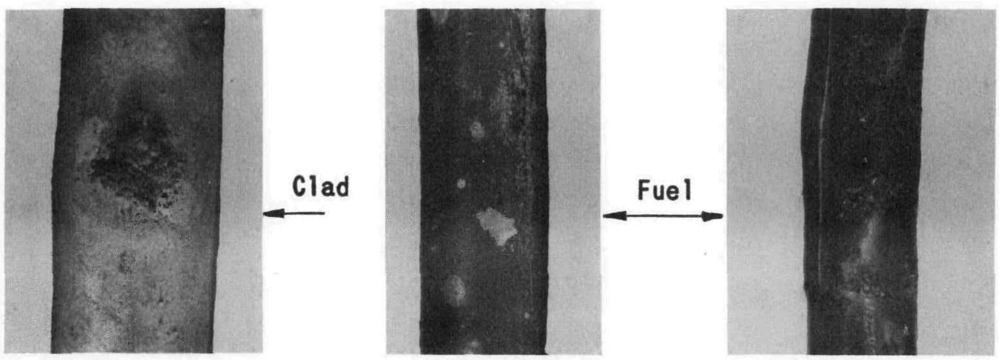


COMMENTS: **Cladding was in good condition. Surface of fuel was smooth, with some cracks. Fuel very brittle and fractured easily during inspection.**

Table IV (Cont'd.)

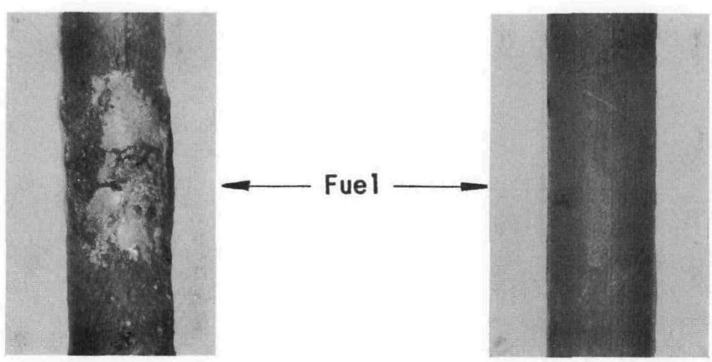
SUMMARY OF EBR-II FUEL PIN IRRADIATIONS

Fuel Alloy and History	Average Density, gm/cc Before	After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
CAPSULE NO. ETR-2						
7.5% fissium; cast	17.490	16.810	--	801	1.36	13.3



COMMENTS: Over-all appearance good. Small (1/16 in. dia.) melt-through in clad about 1 in. below top of fuel. Melt-through attributed to 1 in. long void in underlying sodium bond.

Fuel Alloy and History	Average Density, gm/cc Before	After	Total Thermal Cycles	Maximum Central Metal Temp., °F	Burnup, at-%	Power, kw
CAPSULE NO. ETR-3						
7.5% fissium; cast	17.460	16.750	--	640	1.56	13.4



COMMENTS: Melt-through in clad near top of fuel. Lower portion of fuel in good condition.

A. Total Neutron Flux Dosage

1. CP-5

The total thermal and fast neutron flux values computed from bare and cadmium-covered gold foil traverses of the empty test hole are listed in Table V.

TABLE V

Calculated Thermal and Fast Neutron Flux in Vacant CP-5 Test Hole

	Reactor Power Level, kw	
	1,000	1,600
Thermal flux, n/(cm^2)(sec)		
$5\frac{1}{2}$ in. below core midplane (top of test fuel pin)	2.193×10^{13}	2.915×10^{13}
$14\frac{1}{2}$ in. below core midplane (bottom of test fuel pin)	1.643×10^{13}	2.415×10^{13}
Fast flux, n/(cm^2)(sec)		
$5\frac{1}{2}$ in. below core midplane (top of test fuel pin)	3.255×10^{12}	5.216×10^{12}
$14\frac{1}{2}$ in. below core midplane (bottom of test fuel pin)	1.388×10^{12}	3.905×10^{12}

Theoretical calculations(4) showed that the flux was depressed as a function of uranium enrichment, fuel geometry, and capsule material subsequently installed in the test hole. The averaged value of the radial flux was depressed to 89% for 5% enrichment, 80% for 10% enrichment, and 60% for 20% enrichment. The depression of the center line flux was computed to be 80% for 5% enrichment, 64% for 10% enrichment, and 36% for 20% enrichment. The flux depression incurred by the stainless steel capsule was estimated (from previous reactor experiments) at 15-25%.

2. MTR and ETR

Upon completion of the visual examination, the cobalt-nickel wire was removed from the pin and $\frac{1}{16}$-in. lengths were cut at $1\frac{1}{2}$-in. intervals. After carefully noting its position relative to the fuel pin, each length of wire was weighed and dissolved in a hydrochloric-nitric acid solution. The solution was diluted to volume in a standard flask and its activity compared on a gamma-ray scintillation counter against that of a calibrated NBS sample. The count rates were then used to compute the thermal flux exposure sustained by the fuel pin.

As shown in Table VI, the thermal flux determined from the cobalt-nickel wires averaged about 52% that measured in the empty hole. The decrease is attributed to flux absorption in the stainless steel capsule.

TABLE VI

Measured Thermal Neutron Flux in MTR and ETR Test Facilities

Capsule No.	Thermal Flux, $n/(cm^2)(sec)$	
	Empty Test Hole	Wire Foil
MTR-1	1.4×10^{14}	6.3×10^{13}
MTR-2	1.4×10^{14}	3.38×10^{13}
MTR-3	1.75×10^{14}	1.0×10^{14}
ETR-1	1.2×10^{14}	5.5×10^{13}
ETR-2	1.2×10^{14}	7.3×10^{13}
ETR-3	1.0×10^{14}	8.0×10^{13}

B. Fission Gas Release

Figure 7 shows the apparatus used to measure the quantity of fission gas released from the uranium into the gas expansion volume above the sodium bond level in the fuel pin. The volume of the entire system was calibrated prior to installation in the cave.

The sequence of each measurement was as follows:

(1) The upper end of the clad fuel pin was positioned between the chisel head and anvil in the sealed chamber. The chamber was connected to a vacuum pump and manometer.

(2) The system was evacuated and the manometer reading was recorded.

(3) The vacuum pump was isolated by closure of the stopcock.

(4) The cladding was pierced by striking the chisel head.

(5) The value of the manometer reading was recorded.

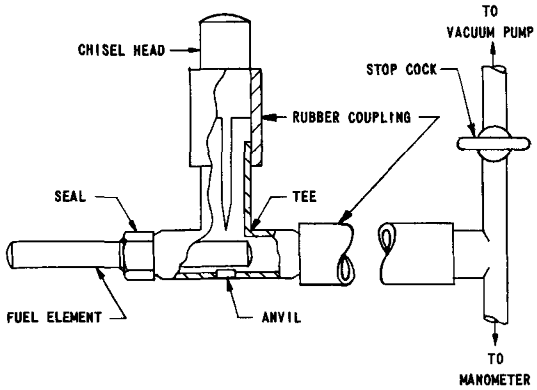


FIG. 7
GAS-SAMPLING APPARATUS FOR IRRADIATED FUEL ELEMENTS

The expansion volume of each pin was originally sealed with approximately one atmosphere of gas pressure. Therefore, any significant gas pressure buildup would be reflected by an appreciable change in the manometer reading upon release of the gas into the measuring system. Since the presence of even small quantities of gas would result in relatively large pressure increases (~2-3 atm) in the fuel pin, only relatively large changes in manometer readings were considered significant. However, none of the measurements evidenced any significant buildup of fission gas.

C. Dimensional and Density Changes

The fuel was removed for dimensional checks and density measurements by cutting three longitudinal slits (spaced 120 degrees apart) in the cladding, and then peeling the clad back from top to bottom. In several instances, the extreme brittleness of the fuel resulted in gross fracture of the uranium during the decladding operation.

The surface of the uranium was washed with alcohol to remove any adherent sodium, and then examined and photographed through a magnifying periscope.

The entire length of the fuel and cladding was traversed at 1-in. intervals to determine (1) whether the fuel had increased in general or in particular areas, and (2) whether the growth was sufficient to expand the cladding. Where diametral growth was observed, densities of 2-in. portions of the fuel were measured in the normal manner - by weighing the sample in air, and submerged in carbon tetrachloride.

Measurements of the can diameters showed that, in general, the cladding did not expand. The exceptions were all caused by excessive fuel temperatures which, in most cases, were promoted by a poor sodium bond between the fuel and the cladding. In one instance of extreme high temperature, the fuel and can grew to an OD of 0.302 in., the inside diameter of the outer capsule.

Measurements of the fuel indicated a diametral growth in most cases. However, the growth was not uniform throughout the length of the pin. Certain of the pins were tapered from one end (higher temperature) to the other (lower temperature), while others evidenced bulges in the central portions. The bulges were promoted by poor sodium bond and consequent higher fuel temperatures.

Precise measurements of the diametral changes in the fuel were precluded by surface conditions. Radiation damage usually starts by a slight roughening of the surface, followed by wrinkling of the metal itself if the metal is hot enough to permit plastic flow. Therefore, density measurements were used to assist in the determination of dimensional changes.

Growth also depends upon the amount of fuel burnup. Pins irradiated to 0.5% burnup at ~800°F showed no change in diameter, whereas other pins irradiated to 1% burnup at the same temperature evidenced a diametral increase of 0.004 in.

Measurement of longitudinal changes was equally difficult, owing to the majority of fractures of fuel pins incurred during the decladding operations. However, although there were a number of cases where the diameter of the fuel increased 10% (abutting the cladding), there was only one case of longitudinal expansion ($4\frac{1}{2}$%) which resulted in contact between the top of the fuel and the restrainer in the fuel pin. This extreme longitudinal growth was attributed to the fact that the diametral expansion was constrained by the cladding.

D. Burnup Analyses

The burnup analyses were performed by Argonne Chemical Engineering Division personnel using the method of determining the Cs^{137}-U^{235} ratio.(5) The fuel samples for analysis included specimens taken from several locations along the length of each pin, and a cross section of the fuel pin.

The analysis showed that the radial flux depression and axial flux gradient resulted in a burnup ratio of approximately 2 to 1 from the top to the bottom of the fuel pin. Consequently, the burnups reported in Table IV are based on the region of highest flux and are an average across the cross section of the fuel pin.

E. Fission Product Contamination of Sodium Bond

Samples of the sodium bond removed from the fuel pin were reacted to place the sodium and associated fission products into water solution. The solution was acidified (with nitric acid) to dissolve any uranium particles. The solution was then analyzed for the presence of Ba^{140}, Sr^{89}, Ru^{106}, Ce^{144}, and uranium.

Table VII lists the isotopic contamination of specimen bond sodium from Capsule No. CP-5-1, which had been irradiated to a calculated burnup of 0.4%.

TABLE VII

Fission Product Contamination of Sodium Bond in Capsule No. CP-5-1

Isotope	Concentration, μgm/gm Sodium
Ba^{140}	173
Sr^{89}	335
Ru^{106}	1.97
Ce^{144}	110

The uranium concentration was undetectable. Similar results were obtained in subsequent analyses. Consequently, it was concluded that bond contamination occurred by a combination of recoil and diffusion of fission products, and that only a very small fraction of the total products is released to the sodium.

An analysis for similar fission product contamination was also performed on the capsule sodium bond removed from Capsule No. CP-5-5. Prior to installation in the capsule, a 0.60-in. hole had been drilled through the cladding at the top of the fuel pin. The objective was to determine the amount of fission products that would be entrained by the reactor coolant in the event of a clad failure in EBR-II. The results indicated that the quantity of fission products released to the capsule sodium bond was less than 1% of the quantity released to the sodium bond in the fuel pin, and that the latter contained less than 0.1% of the total fission products formed in the uranium.(6)

V. DISCUSSION AND CONCLUSIONS

The data in Table IV illustrate several observations made during the course of the irradiation program. The first observation is the stability of the alloys under combined thermal cycling and irradiation conditions. No significant warpage occurred despite the fact that each fuel pin irradiated in the CP-5 reactor accumulated at least 200 to 300 thermal cycles from operating temperature to reactor ambient temperature.

The second observation is the change in the physical state of the irradiated fuel. The entire surface of each fuel pin was pimpled in appearance, which, in general, was more pronounced in the high-temperature areas. The 7.5% fissium alloys evidenced a lesser disposition toward surface coarsening than the low fissium and the zirconium alloys. All of the fuel alloys were embrittled to the extent that it was very difficult to handle the specimens remotely without breakage.

With respect to fuel burnup, it is recognized that some extrapolation is necessary to adapt the data accumulated in a thermal flux environment to that in a fast reactor core. However, this factor can be considered conservative since a fast flux would result in a more uniform volume burnup. Owing to the radial flux depression incurred by the capsule structure, the majority of the fuel burnup occurred at the periphery of the fuel where the temperatures were the lowest. However, since diffusion of the fission products would occur at the temperatures encountered, the net effect would be similar to that of irradiation in a fast flux.

The density of all fuel specimens decreased upon irradiation. However, a review of the data in Table IV showed that, despite the importance of fuel burnup, the dominant factor controlling the volume expansion of the fuel was the operating temperature. This was supported by observations made during a series of furnace heating tests at 1300°F, 1450°F, 1600°F, and 1750°F. These tests were performed on pieces of fissium alloy that had been previously irradiated to a burnup of 1.24% at a maximum temperature of 794°F (MTR-3). As shown in the composite, Figure 8, the heating tests resulted in diametral expansion and, ultimately, severe swelling and cracking of the fuel.

In all of the irradiated samples, the largest volume expansion also occurred in the radial direction. Further, there was only one instance of longitudinal expansion sufficient to effect contact between the top of the fuel and the restrainer. The 5% fissium alloy showed a lesser tendency toward diametral expansion than the 2% zirconium alloy at a given burnup and temperature. With respect to stability, the 7.5% fissium alloy appeared to be slightly better than the 5% alloy.

The absence of irradiation data at 2% burnup precluded a more decisive comparison. However, an overall evaluation of the data accumulated leads to the postulation that the 5% fissium alloy will perform well up to 2% burnup with a central metal temperature approximately 1200°F. The major changes in the fuel will be a decrease in density, with considerable roughening of the fuel surface. Fission gas release is not expected to be troublesome. Accordingly, the 5% cast fissium alloy was selected for the first core loading of EBR-II.

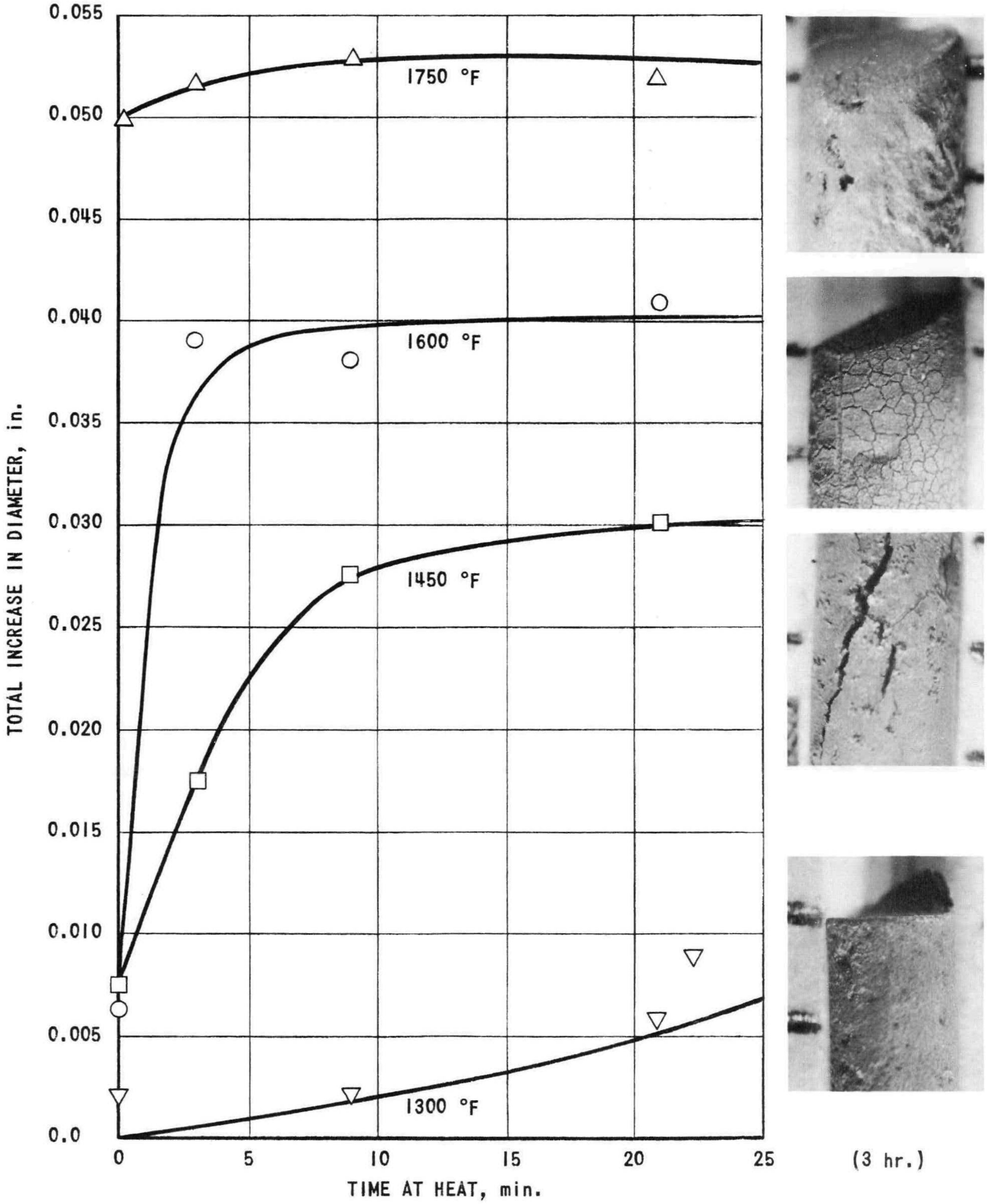


FIG. 8

DIAMETRAL EXPANSION OF FURNACE HEATED 5% FISSIUM ALLOY PREVIOUSLY IRRADIATED TO 1.24% BURNUP AT 794 °F (MTR-3).

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