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THE EBR-II MATERIALS-SURVEILLANCE PROGRAM:

I. Program and Results of SURV-1

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Reactor Technology

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THE EBR-II MATERIALS-SURVEILLANCE PROGRAM:
I. Program and Results of SURV-1

Compiled by Sherman Greenberg

EBR-II Project

September 1969

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THE EBR-II MATERIALS-SURVEILLANCE PROGRAM
I. Program and Results of SURV-1

Compiled by Sherman Greenberg

ABSTRACT

Eight subassemblies containing 15 alloys used in the primary system of EBR-II, and shield graphite canned in Type 304 stainless steel are being irradiated in the EBR-II core at 700°F for varying periods of time and neutron dosage. Most of the alloy specimens are exposed directly to reactor sodium, and the remainder are sealed in helium. The alloys include aluminum bronze; Stellite 6B; Inconel X-750; T-1 tool steel; beryllium copper; Types 304, 347, 416, and 420 stainless steel (the Type 304 in four variations); Type 17-4 PH stainless steel; and tantalum.

The first subassembly (SURV-1) has been removed, and examined after 671 days in the reactor and an estimated average fluence of 3×10^{19} n/cm^2 at energy levels >0.82 MeV. Of the materials exposed, only the beryllium copper and tantalum experienced appreciable weight loss, and thus appear to be unsuitable for use in contact with sodium under the conditions of the test. Types 304 and 347 stainless steel and the graphite assemblies experienced no significant changes in mechanical or physical properties. The other materials experienced appreciable changes in mechanical properties and/or metallurgical structure, but in the absence of thermal-effects control samples, the relative importance of the effects of neutron dosage and long exposures at reasonably elevated temperature cannot be established conclusively for these materials. Ample metallurgical evidence exists, however, to indicate that these materials could experience long-term thermal effects such as annealing, aging, and overaging.

I. INTRODUCTION

A long-term irradiation test program is in progress to monitor the behavior of the neutron-shield graphite and of the materials in service (primarily those in long-time service) in the primary-system sodium of EBR-II. The program involves the exposure and evaluation of ten subassemblies containing the materials of interest in suitable form. Samples of most of the materials that are exposed directly to primary-system sodium in EBR-II are

included in these surveillance subassemblies. These materials and their service temperatures are listed in Table I. (Other materials and conditions of exposure in EBR-II are given in the appendix of this report.)

TABLE I. Materials Exposed Directly to Primary Sodium in EBR-II

Material	Approximate Service Temperature, °F
Aluminum Bronze--Ampco Grade 18-30	700, 800-850, 900
Aluminum Bronze--Ampco Grade 18-23	700
Borated ($1\frac{1}{2}$%) Stainless Steel (Type 304)	700
Carbon Steel--ASTM 325-55T	700
Chromium-plated Stainless Steel Types 304, 410, and 420	700 and 900
Colmonoy 4 and 5	900
Inconel 600	700
Inconel X-750	700
Stainless Steel Types 303, 304, 316, 321, and 347	300 to ~1000
Stainless Steel Types 410 and 420	700 and 900
Steel--SAE 4340	700
Stellite Nos. 3 and 6B	700 and 900
Tantalum	700
Tool Steel Types 18-4-1 or -2 and Rex AA	700 and 900

Eight of the ten subassemblies in the surveillance program were placed in the reactor March 1, 1965. It is planned to remove them for evaluation of their contained material specimens at intervals extending over the operating life of the reactor. The remaining two subassemblies were placed in the storage basket in the primary tank to determine the long-time sodium and thermal (700°F) effects in the absence of appreciable radiation. The first subassembly, SURV-1, has been removed and examined.

This report evaluates the results of SURV-1 and describes the materials-surveillance program. Other evaluation reports will be issued as additional subassemblies are removed and examined.

II. EXPERIMENTAL METHOD

A. Description of Subassembly and Contents

1. General

Each subassembly (see Fig. 1) has three sections. The upper section contains graphite canned in Type 304 stainless steel, the center section

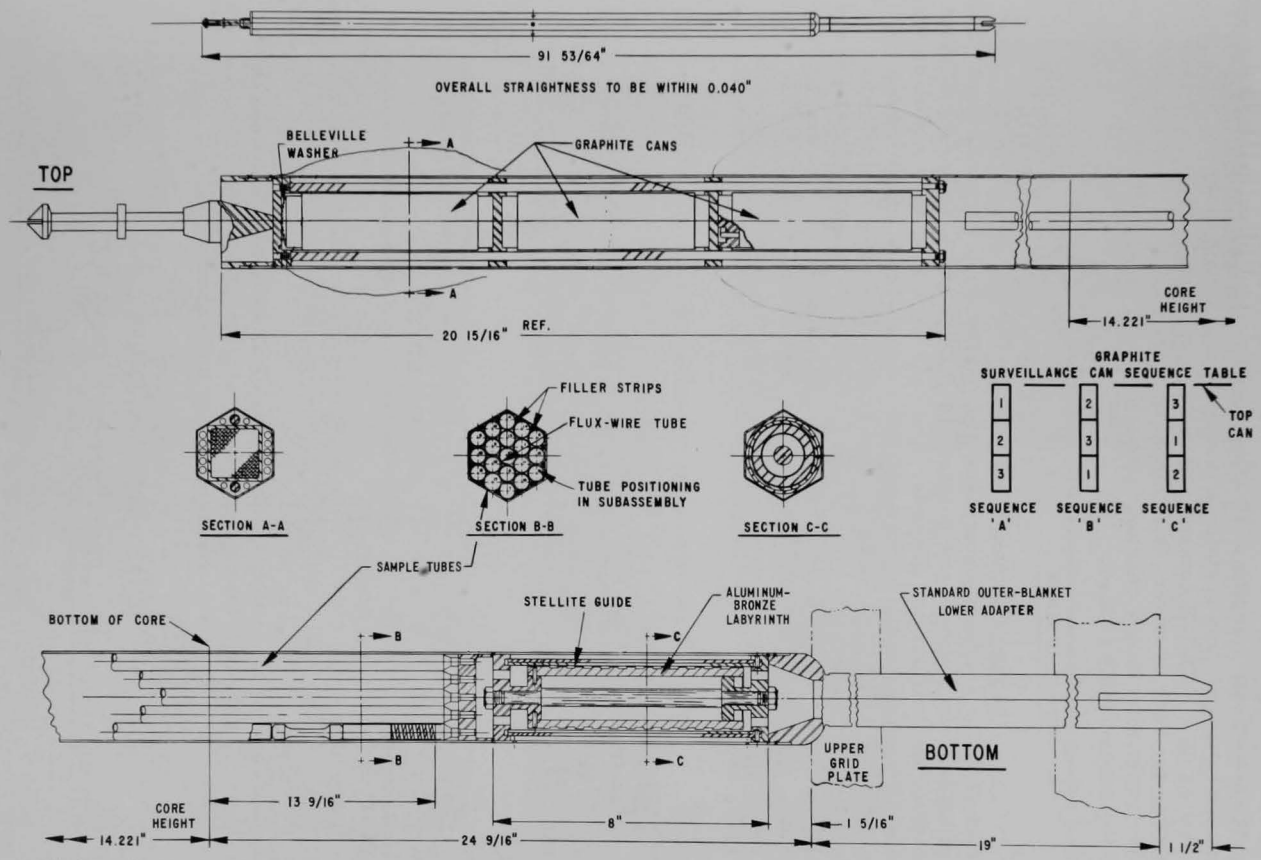

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Fig. 1. EBR-II Material-surveillance Subassembly

contains duplicate samples of 15 alloys used in the EBR-II primary system, and the bottom section contains a labyrinth and Stellite sleeve similar to that used on the EBR-II control-rod drive mechanisms where they pass through the reactor cover. All components are available for examination after removal from the reactor. Detailed descriptions of the three sections of the subassembly follow.

2. Graphite Cans

The upper section of each subassembly contains three cans of graphite under a helium atmosphere at a pressure of 150 to 200 Torr. One can holds plain graphite, one borated (3% B_4C) graphite, and one half plain graphite and half borated graphite. The cans were manufactured using the same procedures and graphite employed for the canned graphite neutron shield of EBR-II.* The wall thickness of the surveillance sample cans (0.050 in.) was chosen to give stresses approximately equal to those in the walls of the full-size shield cans. (The surveillance cans are necessarily smaller than the actual shielding cans in order to fit into the subassemblies.) The construction and appearance of the surveillance cans are shown in Figs. 2 and 3.

Figure 4 shows how the individual cans are assembled into a test section.

The bottom end plates of the section, which are made of 1/2-in.-thick Type 304 stainless steel (the same as in the actual shield cans), were fabricated to receive a piercing adapter to provide a valved,

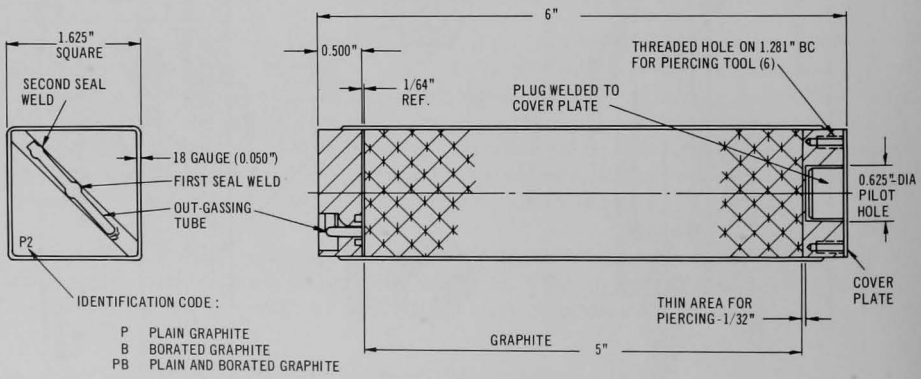

Fig. 2. Construction of Graphite Can

*Heineman, A. H., Postfabrication Evaluation of Integrity and Serviceability of Canned Graphite for the EBR-II Neutron Shield, ANL-7424 (March 1968).

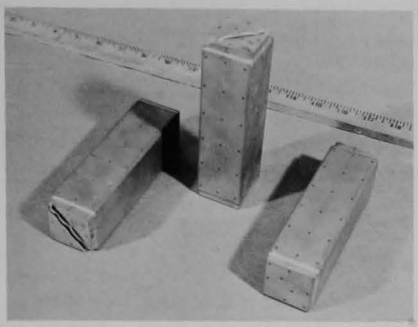

Fig. 3

Graphite Cans before Assembly into Test Section

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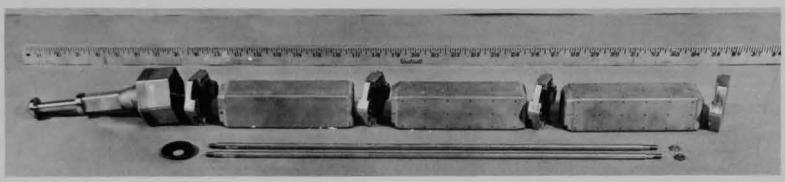

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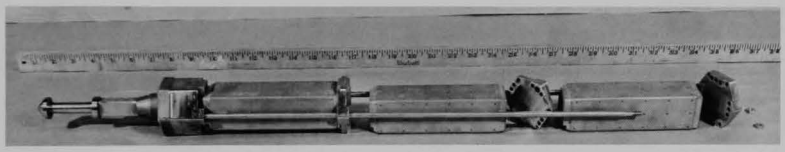

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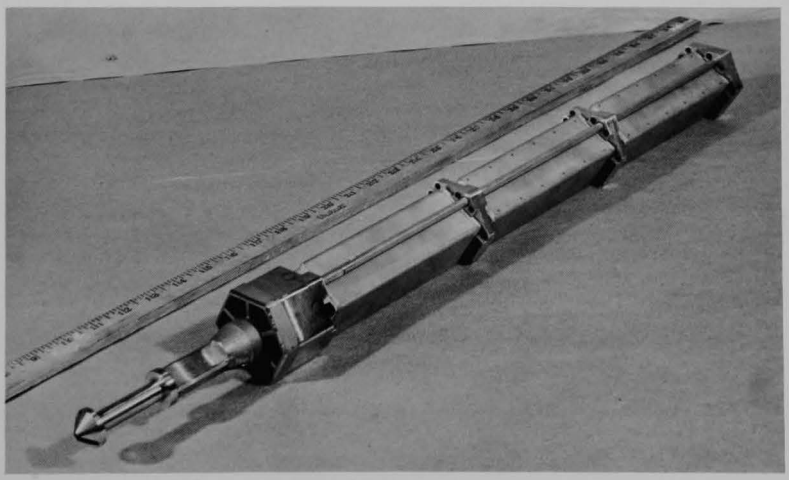

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Fig. 4. Steps in Assembling a Test Section of Graphite Cans

gastight tubing connection to the interior of the can for obtaining gas samples and making pressure measurements. All gas sampling can be done remotely in a hot cell.

Postexposure measurements and examinations may include pressure of the gas, composition of the gas, and physical properties of the graphite. The stainless steel can material also may be examined to determine the effect of internal and external environments.

3. Metallurgical Samples

The center section of each subassembly contains 19 closely packed tubes* fastened to a hexagonal grid. The central tube contains four flux-monitor wires (one each of iron, titanium, nickel, and copper).** The other 18 tubes contain duplicate samples of 15 materials in six configurations (not all materials are in all configurations), some of which are exposed to reactor sodium and some sealed in helium. The temperature of exposure is approximately 700°F. Each stack of samples, held in place by a spring, is distributed approximately equally above and below the centerline of the reactor core. The materials being tested are listed in Table II; compositions and mechanical properties of the materials are listed in Table III; and the metallurgical condition of the materials is summarized in Table IV. The sample types and their code designations are shown in Figs. 5 and 6.

T = tensile
X = Izod
W = Bend

TABLE II. Materials under Test in SURV Program

Material Code	Material	Sample Code[a]
A	Aluminum Bronze--Ampco Grade 18	T, V
B	Stellite 6B	V
C	Inconel X-750	T, V, X, Y, Z
D	Type 420 Stainless Steel	T, V
E	Tool Steel--T-1	T, V
F	Type 347 Stainless Steel	V
G	Type 416 Stainless Steel	V
H	Beryllium Copper--Berylco 25	V
I	Type 304 Stainless Steel with Boron	V
J	Type 17-4 PH Stainless Steel	T, V, X
K	Type 304 Stainless Steel	T, V
L	Type 304 Stainless Steel Welded with Type 308 Stainless Steel	T, W
M	Tantalum	T, V
P	EBR-II Cover Plate (Type 304 Stainless Steel)	T, V, X

[a]See Figs. 5 and 6 for key to code for sample types.

*The tubes are the same size as standard EBR-II outer-blanket elements, being of 0.493-in. OD and 0.457-in. ID.

**Each flux wire was sealed in helium in an individual tube which was, in turn, loaded into an open sample tube.

TABLE III. Composition of Test Samples[a]

Material Code	Material	Composition, %[b]										Other	Mechanical Properties[b]			
		C	Cr	Cu	Fe	Mn	Mo	Ni	P	S	Si		Hardness	Tensile, psi	Yield, psi	Elongation, %
A	Aluminum Bronze--Ampco Grade 18			bal.	3.6							10.7 Al		95,000	47,000	14
B	Stellite 6B	1.06	29.98		0.80	1.24	0.19	1.79			0.69	4.51 W, bal. Co	R_c 36	152,200	93,950	10
C	Inconel X-750	0.04	14.82	0.11	6.39	0.57		bal.		0.007	0.32	0.70 Al, 2.47 Ti, 0.84 Cb + Ta	R_c 39	185,000	135,000	28
		0.03	14.70	0.09	6.52	0.55		bal.		0.007	0.27	0.78 Al, 2.54 Ti, 0.94 Cb + Ta	R_c 39	187,000	135,500	27
		0.05	15.08	0.10	6.51	0.50		bal.		0.007	0.26	0.68 Al, 2.27 Ti, 0.79 Cb + Ta				
		0.03	15.56	0.05	6.42	0.58		bal.		0.007	0.36	0.61 Al, 2.17 Ti, 0.87 Cb + Ta				
D	Type 420 Stainless Steel	0.36	13.47		bal.	0.39	0.02	0.22	0.018	0.010	0.42		R_c 52			
		0.33	13.47	0.10	bal.	0.40	0.03	0.25	0.013	0.007	0.38					
E	Tool Steel--T-1	0.73	3.90		bal.	0.22					0.24	18.20 W, 1.11 V				
F	Type 347 Stainless Steel	0.07	17.93	0.15	bal.	1.48	0.13	9.70	0.019	0.009	0.69	0.90 Cb + Ta	BHN 248			
G	Type 416 Stainless Steel	0.10	12.91	0.08	bal.	0.39	0.04	0.19	0.013	0.347	0.69		R_c 38			
H	Beryllium Copper--Berylco 25			bal.								1.9 Be, 0.25 Co		110,000	90,000	15
I	Type 304 Stainless Steel with Boron	0.01	18.33	0.09	bal.	1.49	0.05	9.36	0.013	0.005	0.64	2.10 B				
J	Type 17-4 PH Stainless Steel	0.031	15.99	3.40	bal.	0.23		4.34	0.014	0.014	0.58	0.27 Cb, 0.02 Ta	R_c 43	193,000	192,000	20.2
		0.046	16.04	3.48	bal.	0.31		4.16	0.018	0.019	0.69	0.25 Cb, 0.01 Ta	R_c 44	203,000	200,000	14.0
		0.041	16.28	3.45	bal.	0.32		4.08	0.016	0.018	0.62	0.32 Cb, 0.02 Ta	R_c 43	204,000	203,000	14.2
K	Type 304 Stainless Steel	0.08	18.38	0.18	bal.	0.89	0.21	10.00	0.018	0.020	0.68		R_b 93			
L	Type 304 Stainless Steel Welded with Type 308 Stainless Steel	0.02	18.90	0.04	bal.	0.68	0.37	9.01	0.026	0.028	0.69					
		0.04	20.50		bal.	1.32		9.80			0.44					
M	Tantalum[c]						<20	<20				16 C; 50 O; 29 N; 3 H; 185 W; 340 Cb; <20 Zr, Ti, V, Co	R_b 35.5 to 38.5			
			<5	<1	7		21	<5			60	15 C; 14 O; 13 N; 60 Cb; <25 Al; <5 Ti; 170 W				
P	EBR-II Cover Plate (Type 304 Stainless Steel)[d]	0.056	18.57		bal.	1.17	0.16	9.46	0.015	0.009	0.57	0.013 Sn, 0.003 Pb	BHN 149	81,000	35,000	62.0

[a]Specification called for use of material still stocked from the manufacture of the original reactor parts or purchased from the same vendor with the same specification as for the original reactor part.

[b]From certified analysis received from vendor. Mechanical properties are not necessarily those of the samples, because some material was heat treated after preliminary or final machining.

[c]Impurities in ppm.

[d]When the original batch of Type 304 wrought specimens (Code K) was made, no original EBR-II stock was believed available. Subsequently, EBR-II stock was located, and specimens were prepared from it.

TABLE IV. Metallurgical Condition of Test Samples

Material Code	Material	Treatment and Condition Specification[a]
A	Aluminum Bronze--Ampco Grade 18	No heat treatment required.
B	Stellite 6B	Wrought condition.
C	Inconel X-750	Tensile, Impact, and Hardness Samples: Anneal starting at 800°F max; heat to 2100°F at 300°F/hr; hold at 2100°F for 4 hr; air cool and rough machine. Age harden by heating from 800°F max to 1625°F at 300°F/hr; hold at 1625°F for 24 hr; air cool. Reheat from 800°F max to 1300°F at 300°F/hr; hold at 1300°F for 20 hr; air cool and grind to finish dimensions. Final hardness should be 37 to 40 Rockwell C. Spring Samples: Heat treat at 1350 ± 25°F for 16 hr; cool in air. Heat press at 800°F for 1 hr with 100-lb load.
D	Type 420 Stainless Steel	Hardness 40 to 45 Rockwell C.
E	Tool Steel--T-1	Hardness 55 to 60 Rockwell C.
F	Type 347 Stainless Steel	Before machining, give full solution anneal (1850-2050°F) and water quench. After machining, stress relieve at 875-925°F for 2 hr.
G	Type 416 Stainless Steel	Harden. Temper at 900°F for 30-34 Rockwell C hardness.
H	Beryllium Copper--Berylco 25	Harden and temper to 41-45 Rockwell C.[b]
I	Type 304 Stainless Steel with Boron	None indicated.
J	Type 17-4 PH Stainless Steel	Harden to 36-41 Rockwell C. Stress relieve at 875-925°F for 2 hr.
K	Type 304 Stainless Steel	Stress relieve, after machining, by holding at 875-925°F for 2 hr.
L	Type 304 Stainless Steel Welded with Type 308 Stainless Steel	After machining, stress relieve at 875-925°F for 2 hr.
M	Tantalum	As received.
P	EBR-II Cover Plate (Type 304 Stainless Steel)	Annealed. Stress relieve at 875-925°F for 2 hr.

[a]Specifications as originally provided for EBR-II parts.
[b]More properly stated as solution treated and aged.

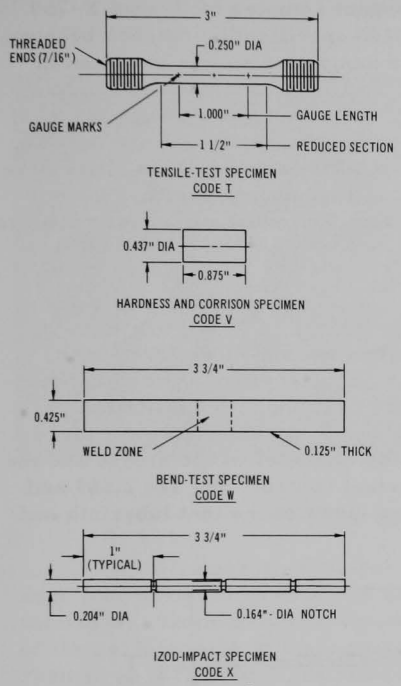
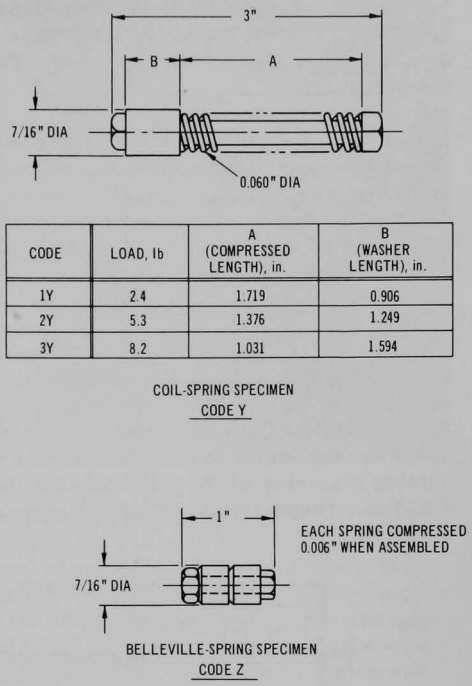


Fig. 5. Surveillance Test Specimens



CODE	LOAD, lb	A (COMPRESSED LENGTH), in.	B (WASHER LENGTH), in.
1Y	2.4	1.719	0.906
2Y	5.3	1.376	1.249
3Y	8.2	1.031	1.594

COIL-SPRING SPECIMEN
CODE Y

Fig. 6. Surveillance Spring-test Specimens

These samples can be used to determine changes in tensile strength, hardness, impact strength, weight, density, and metallurgical structure.

The test materials were selected on the basis that as many as possible of the materials exposed directly to sodium in the EBR-II primary tank were to be included.

Before being loaded into the surveillance capsules, the samples were cleaned ultrasonically in alkaline detergent solution, rinsed successively in hot tap water, alcohol, and acetone, and air dried. Subsequent handling was with gloves or tweezers.

Approximately 20% of the welded Type 304 stainless steel bend samples (Code W) had weld defects, as indicated by radiography, but there was not enough time before assembly to replace them, so they were used as received. The "defective" samples are known and are indicated at the appropriate point in the discussion of the results.

Many of the multinotch Izod-impact samples of Inconel X-750 and Type 17-4 PH stainless steel have out-of-specification notches because original defects had to be removed. These samples also are indicated in the discussion of the data.

The test load ranges for the coil-spring samples were based on the service conditions of the EBR-II springs. Three nominal preloads were specified: 2.4, 5.3, and 8.2 lb. The exact preload and spring rate were determined individually for each spring. Each Belleville-spring sample was preloaded to a compression of 0.006 in.

4. Labyrinth Section

The parts for the labyrinth section were manufactured in the same manner and from the same materials as the parts on the control-rod sleeves and drive rods in the reactor. They were made slightly smaller, however, to fit the subassembly. Figure 7 shows the dimensions of the parts for the tests; in comparison, the inside diameter of the sleeve and the outside diameter of the labyrinth for the actual control rods are 2.563 and 2.533 in., respectively. Figure 8 shows two views of the test labyrinth and guide tube.

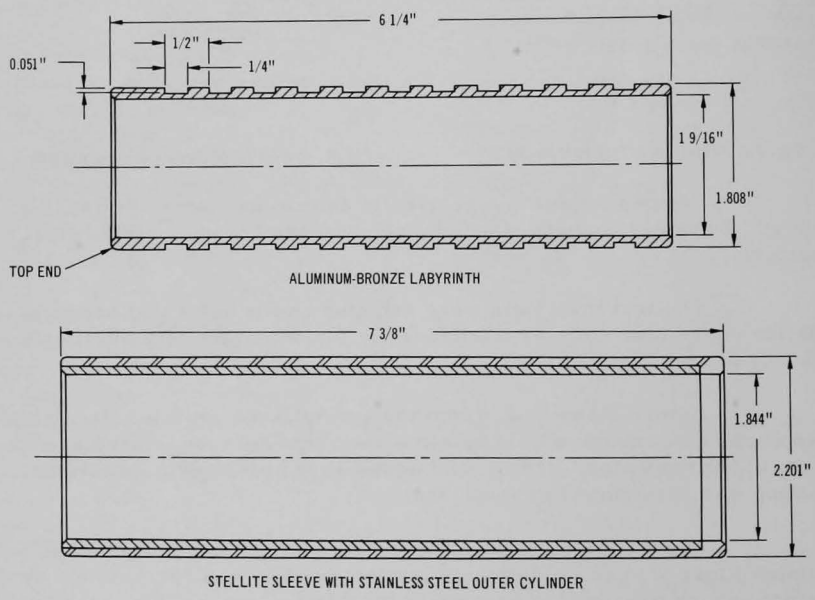
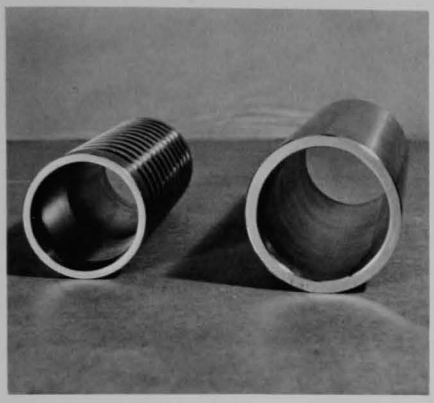


Fig. 7. Construction of Labyrinth Section



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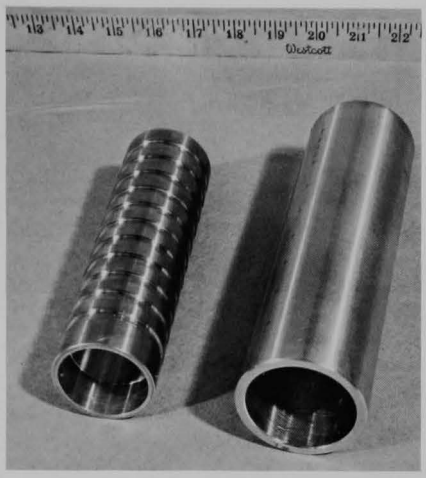

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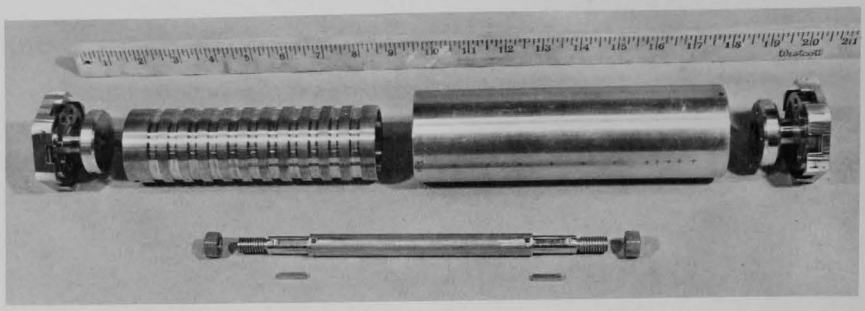
Fig. 8. Labyrinth and Guide Tube

Precise diametrical measurements were taken at 3 points on each land of the labyrinths and along 11 areas on the Stellite sleeve (opposite the lands). Examination of these parts after exposure will aid in evaluation of the performance of these materials and components, particularly with respect to corrosion. The component parts were assembled as shown in Fig. 9. The crosses on the outside of the sleeve indicate the locations of the diametrical measurements of the Stellite. The nuts were tightened to a torque of 180 in.-lb and tackwelded in place as the final step before fastening the labyrinth section into the subassembly.

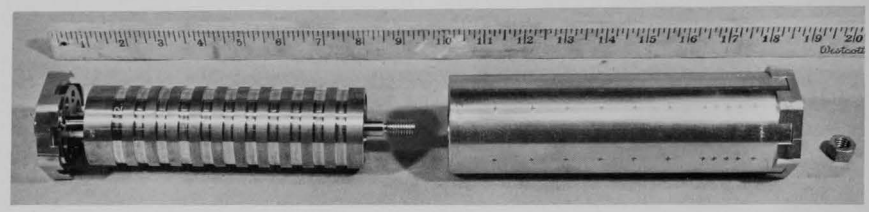
B. Assembly and Loading

The test sample tubes were sorted into their respective subassembly sets, placed into the grid according to the pattern indicated in Section B-B of Fig. 1, and welded to the grid. The tube bundle was placed into the hexagonal tube, the filler strips (identical to those used for the outer-blanket subassemblies) were added around the bundle, and the labyrinth section was spotwelded to the grid. A standard outer-blanket lower adapter was spotwelded to the labyrinth section, the hexagonal tube was pushed down over the shoulder of the lower adapter, and the tube was spotwelded to the lower adapter. The assembled graphite-can section (see Fig. 4) was placed into the upper end of the hexagonal tube, and the tube was spotwelded to the top end fixture.

The completed assembly was inspected for straightness, and a final check was made of the major external dimensions. The subassemblies then were cleaned and wrapped for shipment to EBR-II.



112-4536



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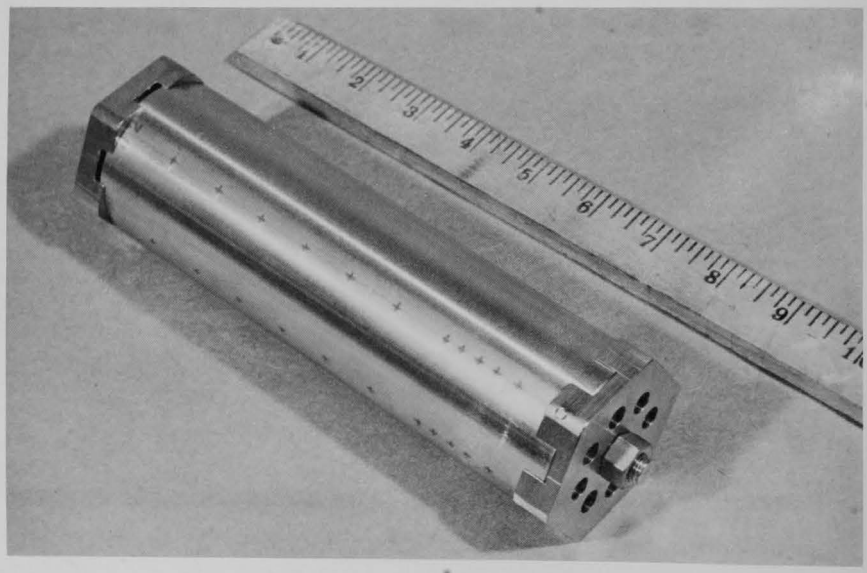

112-4533

Fig. 9. Steps in Assembling a Labyrinth Section

Normal flow of 700°F sodium through an outer-blanket subassembly in rows 10 to 16 is 1.67 gpm at a pressure drop of 4.0 psi. For those subassemblies, almost all of this pressure drop takes place in the lower adapter. In the surveillance subassemblies, the only section that can cause a significant additional decrease in flow is the labyrinth. Calculations indicated that for the same pressure drop of 4.0 psi, the overall flow through these subassemblies was decreased to 1.47 gpm.

The heat generated by these subassemblies is very small. At a radial distance of 56 cm from the center of the core (the location of the irradiation site), the γ heat for stainless steel and materials of similar density is 0.042 W/g and for graphite is 0.037 W/g. Additional heat of 0.026 W/g may be attributed to the $n \rightarrow \alpha$ reaction of the borated graphite. Based on the pessimistic assumption that all of the subassembly is at the centerline of the core, the total heat generation is slightly less than 1 kW. This would yield approximately an 18°F temperature rise in the sodium.

To keep the sample graphite cans in these subassemblies at a flux level equal to or greater than the flux level in the actual graphite cans within the reactor vessel, the subassemblies were placed in the twelfth row of the reactor.

To prevent these subassemblies from interfering with other experiments and to keep them away from the twelfth-row thermocouple in the reactor cover, they were placed in the following specific locations:

Subassembly No.	Reactor Position
SURV-1	12B1
SURV-2	12E1
SURV-3	12A7
SURV-4	12C7
SURV-5	12D7
SURV-8	12F7
SURV-9	12C4
SURV-10	12E8

These subassemblies were loaded into the reactor on March 1, 1965.

The other two subassemblies (SURV-6 and SURV-7) were placed in the storage basket in the primary tank to determine the long-time sodium and thermal (700°F) effects in the virtual absence of neutron radiation. These subassemblies were placed in positions 307 and 321 of the storage basket on February 27, 1965.

The loading arrangement of the metallurgical samples in SURV-1 is shown in Table V.

TABLE V. Loading Chart for SURV-1

Tube No.	Tube Type[a]	Bottom Spacer, in.	Position 1 (Bottom)			Position 2			Position 3			Position 4			Position 5			Position 6			Position 7			Position 8			Position 9			Position 10			Position 11			Position 12			Position 13			Position 14			Position 15		
			Sample Code	Material Code	Sample Number	Sample Code	Material Code	Sample Number	Sample Code	Material Code	Sample Number	Sample Code	Material Code	Sample Number	Sample Code	Material Code	Sample Number	Sample Code	Material Code	Sample Number	Sample Code	Material Code	Sample Number	Sample Code	Material Code	Sample Number	Sample Code	Material Code	Sample Number	Sample Code	Material Code	Sample Number	Sample Code	Material Code	Sample Number	Sample Code	Material Code	Sample Number	Sample Code	Material Code	Sample Number	Sample Code	Material Code	Sample Number	Sample Code	Material Code	Sample Number
1	I	4-55/64	IY	C	1	V	A	1	V	B	1	T	A	1	V	C	1	V	D	1	T	C	1	V	E	1	V	F	1	T	D	1	V	G	1	V	H	1	T	E	1	V	I	1	T	J	1
11	II	4-55/64	IY	C	11	V	A	5	V	B	5	T	A	5	V	C	5	V	D	5	T	C	5	V	E	5	V	F	5	T	D	5	V	G	5	V	H	5	T	E	5	V	I	5	T	J	5
2	I	4-55/64	V	J	1	V	K	1	2Y	C	4	T	K	1	V	M	1	V	A	2	T	L	1	V	B	2	V	C	2	T	M	1	V	D	2	V	E	2	T	P	1	V	F	2	T	A	2
12	II	4-55/64	V	J	5	V	K	5	2Y	C	14	T	K	5	V	M	5	V	A	6	T	L	5	V	B	6	V	C	6	T	M	5	V	D	6	V	E	6	T	P	5	V	F	6	T	A	6
3	I	4-55/64	V	G	2	V	H	2	T	C	2	3Y	C	7	V	I	2	V	J	2	T	D	2	V	K	2	V	M	2	T	E	2	V	A	3	V	B	3	T	J	2	V	C	3	T	K	2
13	II	4-55/64	V	G	6	V	H	6	T	C	6	3Y	C	17	V	I	6	V	J	6	T	D	6	V	K	6	V	M	6	T	E	6	V	A	7	V	B	7	T	J	6	V	C	7	T	K	6
4	I	4-55/64	V	D	3	V	E	3	T	L	2	V	F	3	V	G	3	1Y	C	2	T	M	2	V	H	3	V	I	3	T	P	2	V	J	3	V	K	3	T	A	3	V	M	3	T	C	3
14	II	4-55/64	V	D	7	V	E	7	T	L	6	V	F	7	V	G	7	1Y	C	12	T	M	6	V	H	7	V	I	7	T	P	6	V	J	7	V	K	7	T	A	7	V	M	7	T	C	7
5	I	4-55/64	V	A	4	V	B	4	T	D	3	V	C	4	V	D	4	T	E	3	2Y	C	5	V	E	4	V	F	4	T	J	3	V	G	4	V	H	4	T	K	3	V	I	4	T	L	3
15	II	4-55/64	V	A	8	V	B	8	T	D	7	V	C	8	V	D	8	T	E	7	2Y	C	15	V	E	8	V	F	8	T	J	7	V	G	8	V	H	8	T	K	7	V	I	8	T	L	7
6	I	5-15/64	T	M	3	X	C	1,2,3	T	P	3	X	J	1,2,3	T	A	4	3Y	C	8	T	C	4	T	D	4																					
16	II	5-15/64	T	M	7	X	C	7,8,9	T	P	7	X	J	7,8,9	T	A	8	3Y	C	18	T	C	8	T	D	8																					
7	I	8-43/64			Spacer				V	J	4	T	E	4	W	L	1	1Y	C	3	T	J	4	W	L	2	Z	C	1																		
17	II	None			Spacer				V	J	8	T	E	8	W	L	5	1Y	C	13	T	J	1	W	L	6	Z	C	4																		
8	I	8-43/64			Spacer				T	K	4	W	L	3	X	P	1,2,3	V	K	4	T	L	4	Z	C	2	2Y	C	6	X	C	4,5,6															
18	II	None			Spacer				T	K	8	W	L	7	X	P	7,8,9	V	K	8	T	L	8	Z	C	5	2Y	C	16	X	C	10,11,12															
9	I	8-43/64			Spacer				W	L	4	X	J	4,5,6	V	M	4	T	M	4	Z	C	3	X	P	4,5,6	T	P	4	3Y	C	9															
19	II	None			Spacer				W	L	8	X	J	10,11,12	V	M	8	T	M	8	Z	C	6	X	P	10,11,12	T	P	8	3Y	C	19															
10	I	None	Flux Wires--one each of iron, titanium, nickel, and copper, each 0.020 in. in diameter x 30-1/4 in. long																																												

[a]Type-I tubes have holes for sodium flow. Type-II tubes contain samples sealed in helium.

10 Capsules

III. DOSIMETRY AND EXPOSURE

SURV-1 was removed from the reactor on December 31, 1966. It had been in the reactor for 671 days at a temperature of approximately 700°F, and had received a total exposure of 11,541 MWd. The estimated average total fluence, as calculated from one-dimensional, 22-group diffusion theory, was 9×10^{20} n/cm^2.

Wire flux monitors, in reactor position 12B1, were used to determine fast fluence as a function of position in the subassembly. The upper ends of the wires were 10.58 in. above the centerline of the core.

The neutron reaction for each of the wires is:

Copper: $^{63}Cu(n,\alpha)^{60}Co$

Iron: $^{54}Fe(n,p)^{54}Mn$

Nickel: $^{58}Ni(n,p)^{58}Co$

Titanium: $^{46}Ti(n,p)^{46}Sc$

Each wire was cut into 1-in.-long sections. Each section was weighed, and then analyzed by gamma spectrometry. The activity in counts/min, summed under the photopeak and corrected for sample weight, was converted to dis/min per gram of wire and corrected for decay to December 31, 1966. The data are summarized in Figs. 10 through 13. The curves

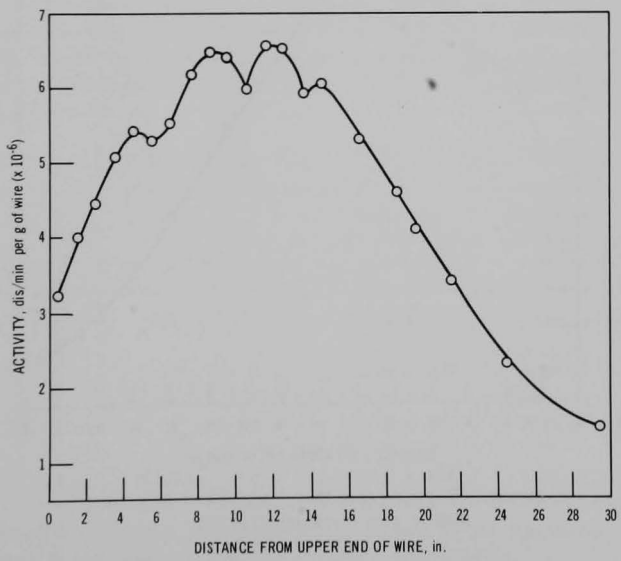

Fig. 10. ^{60}Co Activity of Copper Wire That Monitored SURV-1 Flux (measured 7/6/67)

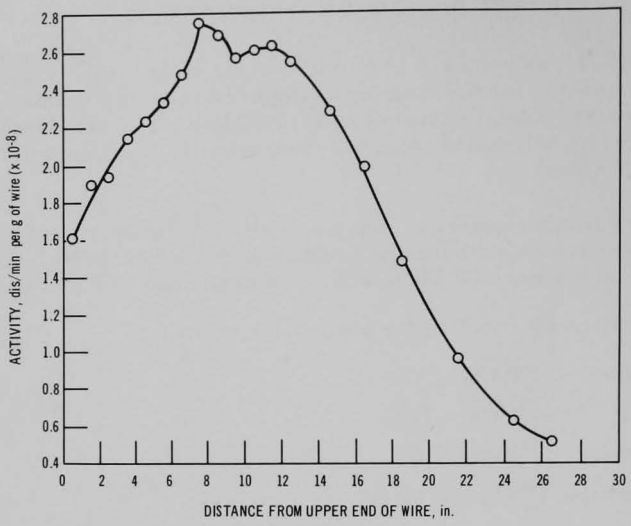

Fig. 11. ^{54}Mn Activity of Iron Wire That Monitored SURV-1 Flux (measured 7/6/67)

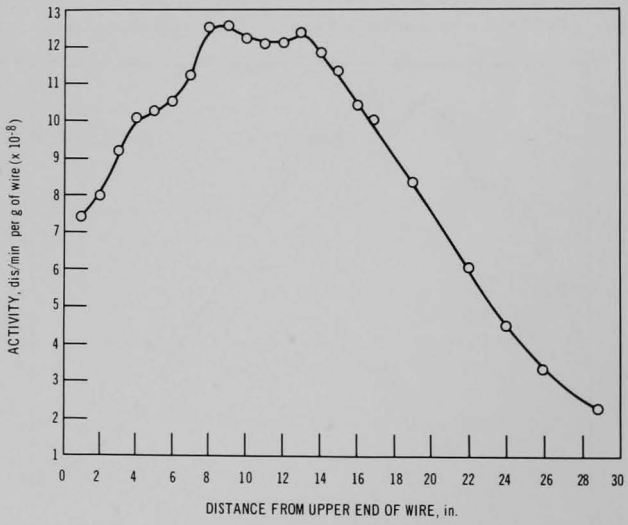

Fig. 12. ^{58}Co Activity of Nickel Wire That Monitored SURV-1 Flux (measured 7/14/67)

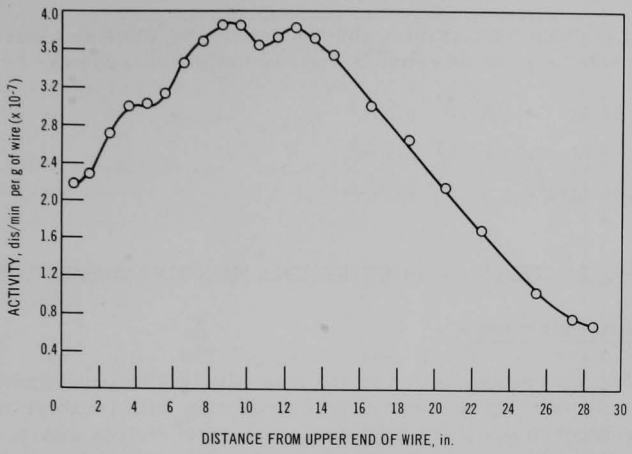

Fig. 13. ^{46}Sc Activity of Titanium Wire That Monitored SURV-1 Flux (measured 7/6/67)

in those figures are useful in assessing the relative exposure of the samples to fast neutrons as a function of position in the subassembly.

The equilibrium activation rate has been calculated from the data for wire activity for a position ~2 in. above the centerline of the core. The calculations are summarized in Table VI. To correct for the intermittent operation of the reactor, the quantity $(1 - e^{-\lambda t_1})(e^{-\lambda t_2})$ was evaluated for each reactor run, where t_1 = irradiation time and t_2 = decay time from end of irradiation to counting time. The values of $(1 - e^{-\lambda t_1})(e^{-\lambda t_2})$ for all reactor runs were summed.

TABLE VI. SURV-1 Neutron Exposure Calculated from Flux Wires

Reaction	$\Sigma\ (1 - e^{-\lambda t_1})(e^{-\lambda t_2})$	Activity, dis/sec per gram of wire	Equilibrium Activation Rate[a]	E_i, MeV
$^{54}Fe(n,p)^{54}Mn$	3.26×10^{-1}	6.95×10^{6}	3.66×10^{8}	~3
$^{58}Ni(n,p)^{58}Co$	5.05×10^{-1}	1.40×10^{8}	4.08×10^{8}	2.9
$^{46}Ti(n,p)^{46}Sc$	4.93×10^{-1}	3.02×10^{6}	7.72×10^{7}	-
$^{63}Cu(n,\alpha)^{60}Co$	8.44×10^{-2}	1.09×10^{5}	1.87×10^{6}	-

[a]Calculated by dividing the activity (column 3) by the summed term in column 2 and correcting for isotopic abundance.

The average energy distribution for the SURV-1 exposure, based on the diffusion-theory calculation, is approximately as follows:

>0.82 MeV--3×10^{19} n/cm^2,

>1.35 MeV--8×10^{18} n/cm^2,

>3.68 MeV--1×10^{18} n/cm^2.

IV. RESULTS OF POSTIRRADIATION EXAMINATIONS

A. General Observations

SURV-1 was dismantled in the Fuel Cycle Facility. Sodium trapped within the subassembly was removed by reaction with isoamyl alcohol and then with ethanol to preclude high-temperature chemical attack of the metallic surfaces by sodium hydroxide and undesired high-temperature metallurgical changes. (The standard procedure for removing sodium from EBR-II fueled subassemblies involves the use of moist argon.)

Observation during the dismantling operations revealed that deposits of corrosion product and/or precipitated metal had accumulated to varying degrees on the stainless steel used for the various parts of the subassembly. For example:

(1) The upper extension shaft used for removing the subassembly was covered with a dark-brown, fairly adherent coating. Several large blobs of white material (probably sodium-alcohol reaction products) also were attached to the gripper extension.

(2) The outer hex can was almost completely clean; small dark areas were visible at the bottom and the top.

(3) The outer stainless steel cylinder of the Stellite sleeve was covered with a very dark-brown coating.

(4) The lower adapter was differentially filmed over with dark corrosion products. In the two grid-plate bearing regions, a thin film very lightly masked the finely ground surfaces. Between these surfaces, the shaft was more heavily coated.

In general, the subassembly had a normal appearance when compared with standard EBR-II fueled subassemblies.

The following observations were made as the subassembly contents were being removed from the hexagonal tube:

(1) The stainless steel cans encapsulating the three samples of graphite were heavily coated with a dark-gray, strongly adherent product. Tie rods, hex guides, nuts, and Belleville-spring washers of the canned graphite assembly were similarly coated.

The effects of the force of the Belleville-spring washer supporting the graphite cans were not determinable beyond establishing that the three cans were still held together tightly.

(2) All 19 tubes containing the EBR-II materials of construction were bright and shiny.

(3) All 12 filler strips in the 19-tube array were coated with a rust-colored corrosion product.

(4) A disc of newly machined yellow brass was introduced into the work area to better judge the appearance of the aluminum-bronze labyrinth under the yellow sodium lights illuminating the caves. The exposed labyrinth was unmistakably coated with a thin film of dark corrosion product through which was seen the characteristic yellow color of the bronze. All the machined edges were sharp, and there was no visible pitting. A preliminary micrometer measurement of the diameter indicated little or no change of dimension.

(5) Variations in notch ductility were observed while disassembling the 19-tube cluster. As few as four complete bends through an angle of 30/45° to as many as 12 bends were required to break the connecting notched pin between tube and hexagonal grid plate; about two-thirds of the pins broke after eight bends.

Tube No. 10, containing four wire flux monitors (Cu, Ti, Fe, and Ni), was removed from the Fuel Cycle Facility and gamma scanned; a maximum activity of 4 R at the middle was found. A similar scan of one of the canned graphites showed 25 R.

The results of measurements and examinations performed with SURV-1 samples and components are given in the following sections. It will be noted that not all measurements and examinations contemplated in the original design of the experiment have been made. Omissions are due either to lack of suitable equipment or to a realization that the data obtained would not be of importance (e.g., if a sample showed no change in microstructure as a result of exposure in sodium, there was no point of examining a similar sample exposed in helium).

B. Weight-change Data

The hardness samples (Type V) that had been exposed to sodium were used for weight-change measurements as well. The weight-change data

obtained are summarized in Table VII. Small weight losses were observed for all samples except beryllium copper and tantalum, which exhibited relatively large weight losses.

TABLE VII. Weight Changes of Hardness Samples[a]

Material	Average Initial Weight, g	Weight Loss, mg[b]		
		Average	Range	Comments
Aluminum Bronze--Ampco Grade 18	16.06	1.7	0.5-2.6	
Stellite 6B	18.09	1.4	1.1-1.7	
Inconel X-750	17.74	1.6	0.2-3.3	
Type 420 Stainless Steel	16.50	1.8	1.2-2.6	Hardened
Tool Steel--T-1	18.68	4.2	0.6-8.9	
Type 347 Stainless Steel	16.94	1.1	0.0-1.7	3 samples
Type 416 Stainless Steel	16.29	2.4	1.6-2.9	Hardened
Beryllium Copper--Berylco 25	17.97	151.2	126.2-177.0	3 samples[c]
Type 304 Stainless Steel with Boron	16.78	3.1	2.0-4.2	
Type 17-4 PH Stainless Steel	16.69	1.1	0.7-1.5	Hardened
Type 304 Stainless Steel	16.85	1.3	0.0-2.1	3 samples
Tantalum	36.20	13.1	3.8-22.4	2 samples

[a]Except as noted, the results are based on four samples of each material.
[b]Exposed surface area was 7.8 cm^2.
[c]A fourth sample lost only 3.4 mg and is not included in the average.

C. Metallographic Examination

The hardness samples exposed to sodium and the unexposed control samples were examined by optical metallography. The results of the examination are summarized in Table VIII. Only the beryllium-copper alloy showed surface attack.

TABLE VIII. Effect of EBR-II Exposure on Microstructure

Material	Observations
Aluminum Bronze--Ampco Grade 18	Dispersion of γ_2-phase during exposure; probably due to peritectoid reaction at exposure temperature.
Stellite 6B	Martensitic transformation.
Inconel X-750	No change. Exposed sample was less sensitive to etchant.
Type 420 Stainless Steel	Grain-boundary precipitation and precipitate agglomerization.
Tool Steel--T-1	No change.
Type 347 Stainless Steel	No change. Exposed sample was more sensitive to etchant.
Type 416 Stainless Steel	No change.
Beryllium Copper--Berylco 25	Irregular surface dissolution. Considerable fine precipitation apparently due to overaging, since time and temperature far exceeded standard aging conditions.
Type 304 Stainless Steel with Boron	No change.
Type 17-4 PH Stainless Steel	No change.
Type 304 Stainless Steel	No change except for possible transformation at surface.
Tantalum	No change.

D. Hardness-change Data

The average Vickers Hardness Number (average of at least five indentations on each sample) of samples exposed in sodium and of the control samples are given in Table IX.* All the hardness tests, including those with unirradiated samples, were made with the same Tukon microhardness tester. Reproducibility (to ±3%) was verified by hardness measurements on a calibrated standard test block before and after testing each sample.

Though the samples were exposed to the primary sodium during irradiation, most exhibited no surface hardening or softening. The individual hardness numbers varied in a random fashion across sections of several of the samples. The variation probably was due to dispersed precipitates in the alloys.

The most striking result of the hardness measurements was the reduction in hardness of the beryllium-copper (Berylco-25) samples. This is a precipitation-hardening alloy that previously has been reported to increase in hardness during irradiation. Possibly the sodium environment is responsible for the softening; surface corrosion was observed on all four Berylco-25 samples. The microstructures of these samples also changed; there was severe overaging as evidenced by agglomerization.

The samples of T-1 tool steel also became softer during the irradiation. This probably was due to decarburization in sodium at the ~700°F operating temperature of the subassembly.

The Inconel X-750, the borated Type 304 stainless steel, and the Type 304 stainless steel did not harden appreciably during the irradiation.

The remaining samples (aluminum-bronze, Stellite, Type 420 stainless steel, Type 347 stainless steel, Type 416 stainless steel, Type 17-4 PH stainless steel, and tantalum) hardened appreciably during the irradiation. Generally, the hardness of these samples increased with neutron exposure of the samples. It is not possible, however, to determine the correlation between the two because of uncertainties in both hardness data and neutron-exposure values.

E. Strength and Ductility

Stress-strain curves to approximately 0.5% elongation were obtained at room temperature for the unirradiated and irradiated material. The results are summarized in Table X.**

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*The hardness of the samples exposed in the helium environment was not measured.

**The experimental program was stopped before the tool steel could be tested.

TABLE IX. Effect of EBR-II Exposure in Sodium on Hardness

Material and Sample No.	Exposure: Approximate Percent of Maximum Integrated Flux	Exposure: Approximate Total Integrated Flux,[a] $n/cm^2 \times 10^{19}$	Average Vickers Hardness Number (500-g load)
Aluminum Bronze--Ampco Grade 18			
Control	0	0	191
1	44	0.6	224
2	78	1.1	213
3	99	1.4	223
4	26	0.4	202
Stellite 6B			
Control	0	0	371
1	50	0.7	425
2	99	1.4	485
3	90	1.3	453
4	31	0.4	428
Inconel X-750			
Control	0	0	372
1	73	1.0	375
2	99	1.4	353
3	75	1.1	376
4	55	0.8	399
Type 420 Stainless Steel			
Control	0	0	358
1	78	1.1	372
2	99	1.4	403
3	26	0.4	397
4	61	0.9	377
Tool Steel--T-1			
Control	0	0	710
1	99	1.4	672
2	90	1.3	646
3	31	0.4	681
4	99	1.4	696
Type 347 Stainless Steel			
Control	0	0	148
3	55	0.8	162
4	99	1.4	166
Type 416 Stainless Steel			
Control	0	0	275
1	99	1.4	314
2	26	0.4	306
3	63	0.9	294
4	99	1.4	277
Beryllium Copper--Berylco 25			
Control	0	0	320
1	90	1.3	161
2	31	0.4	165
3	99	1.4	176
4	90	1.3	176
Type 304 Stainless Steel with Boron			
Control	0	0	237
1	75	1.1	270
2	73	1.0	268
3	99	1.4	231
4	75	1.1	244
Type 17-4 PH Stainless Steel			
Control	0	0	298
2	78	1.1	363
Type 304 Stainless Steel			
Control	0	0	141
1	31	0.4	143
2	99	1.4	143
3	90	1.3	143
Tantalum			
Control	0	0	121
1	73	1.0	161
2	99	1.4	209
3	75	1.1	210

[a]At energy levels >0.82 MeV. Based only on axial variation in flux; no correction was made for attenuation of flux across diameter of subassembly.

Room Temp

TABLE X. Strength and Ductility of SURV-1 Materials

Exposure		Strength, psi x 10^{-3} [a]				Ductility, % [a]			
		Tensile		Yield Point		Elongation		Reduction of Area	
Fluence[b]	Environment[c]	Average	Range	Average	Range	Average	Range	Average	Range
				Aluminum-Bronze--Ampco Grade 18					
0	Control	103.3	102-106	48.7	47-51	29.7	28-31	41.3	40-44
8 x 10^{18}	Na	98.7	96-101	58.0	55-60	11.3	10-13	11.3	10-13
9 x 10^{18}	He	97.5	95-100	57.2	54-61	9.7	9-10	11.2	9-19
				Inconel X-750					
0	Control	170.0	168-171	96.0	91-99	24.0	21-26	23.7	21-26
8 x 10^{18}	Na	159.3	157-160	130.3	127-133	17.5	15-20	18.5	14-21
8 x 10^{18}	He	157.2	152-161	129.0	124-132	18.2	16-24	18.5	13-27
				Type 420 Stainless Steel					
0	Control	224.1	207.0-228.4	170.1	164.0-180.3	5.7	5.0-6.0	35.7	30-44
9 x 10^{18}	Na	209.8	200-227	173.8	157-183	12.2	9-16	40.5	34-44
9 x 10^{18}	He	201.2	198-202.8	171.3	169-173	9.0	6-12	40.5	33-43
				Type 17-4 PH Stainless Steel					
0	Control	174	172-176	168.3	166-171	17.3	17-18	53.3	52-54
1 x 10^{19}	Na	214.7	211-221	204.5	201-208	12.2	12-13	41.5	38-44
1 x 10^{19}	He	212.2	209-216	204.0	194.5-209.5	9.7	3-13	29.7	8-40
				Type 304 Stainless Steel					
0	Control	111	101-123	85.0	75-98	40.3	34-46	58.3	49-69
8 x 10^{18}	Na	117	104-128	92.0	77.9-104	39.0	35-45	63	57-68
8 x 10^{18}	He	108	104-118	79.0	78-80	42.7	36-46	70.7	67-77
				Type 304 Stainless Steel Welded with Type 308L Stainless Steel					
0	Control	87.7	87-88	47.7	47-48	37.7	33-42	59.0	57-61
9 x 10^{18}	Na	90.1	89-91	51.9	50-53.3	26.3	25-29	54.0	45-60
9 x 10^{18}	He	88.9	88-89.9	51.5	47-60	29.2	23-34	51.0	46-59
				Tantalum					
0	Control	51.3	47-56	36.3	32-41	56.7	54-63	95	90-99
1 x 10^{19}	Na	97.1	89.7-103	96.1	89-101.5	20.7	16-24	86.2	80-89
1 x 10^{19}	He	100.1	89.6-106	96.5	92-104	16.2	15-18	69.7	69-70
				EBR-II Cover Plate (Type 304 Stainless Steel)					
0	Control	80.7	80-82	33.7	33-34	74.0	72-77	79.0	75-82
1 x 10^{19}	Na	79.2	78-80.4	35.0	33.0-38.3	70.0	69-71	80.5	79-82
1 x 10^{19}	He	79.8	79-80	34.3	33-40	70.3	68-72	75.0	71-80

[a] Average and range for three or more samples.
[b] Exposure averaged from vertical disposition of samples in reactor. Includes only that portion of fluence with energy >0.82 MeV.
[c] Na and He refer to the exposure environment in EBR-II.

The strength and ductility of annealed and moderately work-hardened Type 304 stainless steel were not affected in the irradiation cycle; ductility of Type 308 stainless steel weld metal was reduced slightly. Hardened Inconel X-750 and Type 17-4 PH stainless steel overaged, with accompanying losses in ductility; Inconel X-750 was weakened, but the Type 17-4 PH steel (a precipitation-hardened alloy) was strengthened. The tensile strength of tantalum was almost doubled, but the ductility of the metal was reduced considerably. The aluminum bronze apparently overaged similarly to Inconel X-750. Both tantalum and aluminum bronze exhibited an approximately 50% increase in the modulus of elasticity. Analyses for carbon, oxygen, and nitrogen in tantalum exposed to sodium (made in an attempt to explain the change of properties in terms of increase in concentration of those elements) showed no appreciable increase in bulk concentration of these elements as a result of reactor exposure. The softening of hardened Type 420 stainless steel is believed to be an effect of long-time annealing.

Inspection of Table X reveals no conclusive evidence that the sodium or helium environment had any effect on strength or ductility.

Comparison of Tables IX and X reveals no consistent relationship between hardness and tensile strength. For example, the hardening of Type 17-4 PH stainless steel reflected its strengthening, but aluminum bronze hardened and lost strength. The annealing of the previously hardened Type 420 stainless steel was undetected by hardness measurements.

F. Impact Strength

Impact tests were made with a Baldwin impact tester that had a maximum impact-energy capacity of 16 ft-lb delivered at 11.3 ft/sec, and which had been modified for remote operation. The results of impact tests of unirradiated and irradiated Inconel X-750 and Type 17-4 PH stainless steel are compared in Figs. 14 and 15, respectively. Although sample fluence ranged from 0.6×10^{19} to 1.1×10^{19} (>0.82 MeV), scatter of the data masked any effect of neutron dosage as well as any difference between the effects of exposure in sodium and helium. Data for all irradiated samples, whether exposed to sodium or helium, are plotted on a single line for each material.

The significant reduction of impact strength of Inconel X-750 is probably attributable to radiation hardening, but may be a result of aging at 700°F. Resolution of the contribution of the different factors in the environment is impossible in the absence of thermal-effects control samples. There was no discernible effect of sodium on impact strength.

Impairment of the impact strength of the hardened Type 17-4 PH stainless steel (H900 condition) was more marked than for hardened Inconel X-750. The exposed stainless steel was hardened considerably (from 298 to 363 Vickers), an effect believed due to aging during the

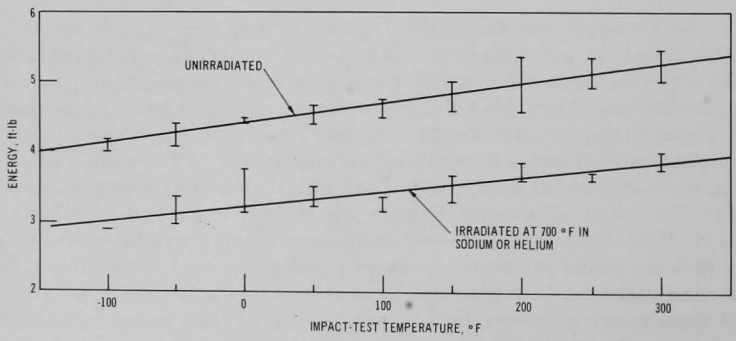

Fig. 14. Izod Impact Strength of Inconel X-750

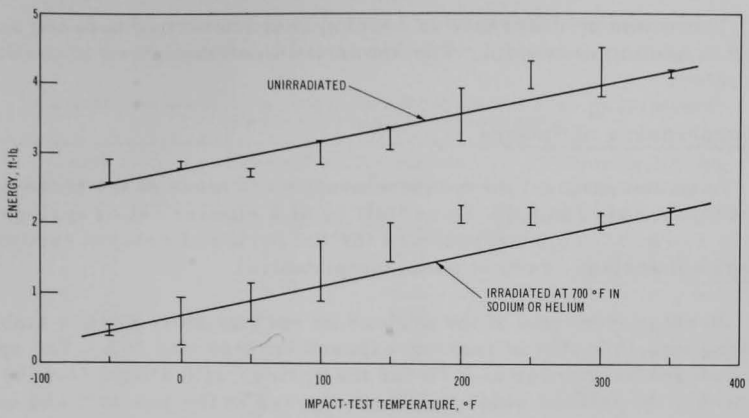

Fig. 15. Izod Impact Strength of Type 17-4 PH Stainless Steel

long-time exposure at 700°F. As with Inconel X-750, there can be no further resolution of the observed effects in the absence of thermal-effects control specimens. There was no discernible effect of sodium on impact strength.

The notch ductility of the EBR-II cover-plate material (Type 304 stainless steel) was preserved at temperatures as low as -250°F. Although bent at the notch, no specimen was fractured by the impact machine.

G. Bend Test of Welded Type 304 Stainless Steel Samples

The beam-bend test specimens were supported on two round pins $2\frac{1}{2}$ in. apart as the bending force was applied perpendicularly at the mid-point to the 3.75 x 0.425-in. face. Deflections were measured using a compressometer with a maximum travel of one inch. Maximum load was reached at about 1/2-in. deflection, after which the load tapered off with additional deflection.

The maximum load sustained by samples exposed in EBR-II varied between 265 and 288 lb for the eight samples tested. Load at 1-in. deflection varied between 232 and 252 lb. Corresponding loads for control (unexposed) samples were 264-282 and 200-222 lb, respectively. There was no consistent relationship between relative loads at 1/2-in. deflection and at 1-in. deflection, probably because of friction effects of the three load points.

Visual examination of the bend specimens showed no large cracks or other abnormalities. The slight graininess of the surface metal in bend areas was equivalent to that noted in unirradiated Type 304 stainless steel welded control samples that had been bent under similar conditions. (Non-uniform local deformation results in delineation of grains.)

There was no difference in bending characteristics between samples exposed in sodium or helium. The known weld defects in two of the samples had no effect.

H. Measurements of Springs

Load and spring-rate measurements were made of the Inconel X-750 coil springs exposed in EBR-II, as well as of a similar set of springs which had been stored at room temperature for the period of reactor exposure (compressed springs creep at room temperature).

Average reduction of the preload on springs stored under ambient conditions was 16%; that of reactor-exposed springs was 26%. The spring rate increased an average of 3.7% for the springs with a light (2.4 lb) and medium (5.3 lb) preload which had been exposed in the reactor, and only 0.3% for the corresponding springs stored under ambient conditions. Scatter in the data for the springs with high (8.2 lb) preload, both before and after exposure, makes a meaningful comparison for these springs impossible.

There was no discernible difference in behavior between springs exposed in sodium and those sealed in helium.

The Belleville springs have not been examined.

I. Tube-burst Tests

Tube-burst tests using demineralized water were conducted with 4-in. lengths of the Type 304 stainless steel blanket-rod tubes used as containers for the metallurgical samples. The tubes tested were from two different environments and in two different stress states: (1) unstressed tubes with flowing sodium outside and essentially static sodium inside, and (2) nominally stressed tubes with flowing sodium outside and helium inside. The ultimate tensile strengths and ductilities of the irradiated tubes at room temperature were compared with the corresponding values for unirradiated blanket-rod tubes from two separate sources (i.e., tubes used as containers for shipping the SURV-1 specimens from Idaho to Illinois, and tubing withdrawn directly from shop stock). Table XI summarizes the results of the tube-burst tests.

Although the effects of exposure on the ultimate strength are clouded by the scatter of the results for unirradiated specimens, some of which apparently were cold-worked slightly, the following conclusions are apparent:

(1) Neither the strength nor the ductility of the tubes was significantly affected by the 1.2×10^{19}-n/cm^2 dosage; the length of the longitudinal split and the bulge diameter at splitting (both ~1.5 times the diameter) were essentially the same for all tube sections tested (29 unirradiated and 37 irradiated).

(2) There were no observable effects of differences in environment, radiation dose, or thermal gradient along the tube axis.

TABLE XI. Results of Room-temperature Burst Tests of Type 304 Stainless Steel Blanket-rod Tubing (of 0.493-in. OD)

Tube Identification No.	Fluence,[a] 10^{19} n/cm^2	Bursting Pressure, psi	Average Bursting Stress, psi
	Tubes Used as Shipping Containers[b]		
S-1-13	0	6630	88,700 (0.018-in. wall)
S-1-16	0	6630	85,800 (0.0185-in. wall)
S-1-3	0	6360	84,800 (0.018-in. wall)
S-1-12	0	6000	84,700 (0.017-in. wall)
S-1-17	0	5730	76,400 (0.018-in. wall)
S-1-8	0	5700	75,900 (0.018-in. wall)
S-1-1	0	5600	74,600 (0.018-in. wall)
S-1-19	0	5500	73,300 (0.018-in. wall)
	Blanket-rod Tubing from Shop Stock (Trentweld A269)		
Heat 620768	0	6520	84,500 (0.0185-in. wall)
	Irradiated Blanket Tubes from SURV-1 Subassembly		
0 (Na)[c]	0.6-1.2	6400 (4)[d]	85,000-88,000[e]
8 (Na)	0.6-1.2	6475 (5)	
11 (He)	0.6-1.2	6570 (6)	
12 (He)	0.6-1.2	6500 (6)	
13 (He)	0.6-1.2	6500 (6)	
15 (He)	0.6-1.2	6450 (6)	
16 (He)	0.6-1.2	6600 (4)	

[a]At energy levels >0.82 MeV.
[b]Average of three tests per tube.
[c]Internal tube environment indicated in parentheses.
[d]Average burst pressure for number of tube samples indicated in parentheses.
[e]Nominal bursting stress for 0.018-in.-wall tubing as calculated from the formula $P = ts/(R_0 - 4t)$ given in ASME Unfired Pressure Vessels, Section VIII, Paragraph UA-1, where P is bursting pressure, t the original wall thickness of the tube, s the bursting stress, and R_0 the original outside radius of the tube. (The wall thickness of the irradiated tubing was not measured. A nominal thickness of 0.018 in. was assumed.)

J. Examination of Lower Adapter, Labyrinth, and Guide Tube

The lower hard-chromium-plated adapter was sectioned to determine the effects of sodium in the crevices of the two grid-plate bearing areas. The plating was still present to a depth of 0.065 mm, and the outer surface was smooth on a microscale. Imperfections in the plating were similar to those noted on a control that was not exposed. An attempt was made to distinguish the plated areas from the stainless steel by superficial hardness testing. This was unsuccessful both for the control and the exposed specimen, but microhardness traverses of the control clearly distinguished between the plating and the steel. Equipment for a microhardness traverse of the exposed sample was not available.*

*This rudimentary investigation of the performance of chromium plating was made in support of a proposal to use a chromium-plated nickel reflector in EBR-II. It was not part of the original program.

Measurements of the diameters of the labyrinth and guide tube showed increases of less than 0.002 in. in all cases. All changes were in the direction of increased material dimensions; nevertheless, the labyrinth and guide tube were very easily disassembled.

K. Examination of Graphite and Cans

The graphite samples were irradiated in a position which would yield an exposure approximately half that at the core centerline. After exposure, the cans were measured on a granite surface plate to check width, length, twist, and straightness. Length measurements were accurate to ±0.001 in. Width, twist, and straightness measurements were accurate to ±0.002 in. Within these limitations, there was no distortion of the cans.

The three graphite capsules were decanned after the gaseous contents were sampled. No in-leakage of sodium to the graphitic materials was found, although sodium leaked into the annulus between the cover plate and the end cap of the can containing plain graphite. (The end-cap/cover-plate arrangement was at one end of the can for convenience in piercing and sampling; performance of this structure has no relevance to soundness of canning procedures used for EBR-II graphite.) Problems associated with the piercing and gas-sampling procedure cast doubt on the reliability of the analyses; they are not reported here. There was no evidence of B_2H_6 in any of the capsules.

1. Graphite

The densities of the irradiated graphite materials, as calculated from physical measurements and weights of machined cylinders, are listed in Table XII. The density of the unirradiated graphite was obtained in the same manner, using machined blocks. Two other methods of density measurement--the displacement of water and the pycnometer using fine lead shot--were found unsatisfactory, the first because of the porosity of the graphite and the second because of uncertainties in the density of the packed lead shot.

TABLE XII. Effect of Exposure on Calculated Densities of Graphite Irradiated to 6×10^{18} nvt ($E > 0.82$ MeV)

		Density, g/cm^3	
Can No.	Graphite Type	Before Exposure	After Exposure
P-1	Plain	1.5419-1.6376[a]	1.6366
B-1	Borated	1.5601	1.5545
PB-1[b]	Plain	1.5419-1.6376[a]	1.6429
PB-1[b]	Borated	1.5601	1.5478

[a]Range for six lots used in EBR-II.
[b]Can PB-1 contained the mixed graphites (one the plain and borated types).

Dimensional measurements revealed no significant changes. None of the four samples of graphite had cracked or spalled during the irradiation.

2. Can Material

A section of the Type 304 stainless steel can was examined to determine the extent of any carbon transfer. A hardened case, 0.03-0.04 mm deep, was observed on the interior surface. There was no tendency for the graphite to stick to the stainless steel during the separation. Tensile samples also were prepared, but no measurements have been made. However, since the carburized zone was such a small fraction of the total thickness of 1.3 mm, it is improbable that the mechanical properties were significantly modified.

V. DISCUSSION AND CONCLUSIONS

The primary purpose of the EBR-II Materials Surveillance Program is to ensure the adequacy of materials of construction of EBR-II primary-system components, particularly those components intended to have a long service life. In addition, the program is designed to allow accumulation of data of general pertinence to the LMFBR Program.

SURV-1 confirmed the soundness of choices of EBR-II materials with only two exceptions. As would be expected from the data, beryllium-copper control-rod bushing rings had failed* in service and were replaced with aluminum bronze. The aluminum-bronze rings, as could be expected from the SURV-1 data, have performed satisfactorily. Failure of the beryllium-copper rings did not result in serious problems.

As indicated in Table VII, the behavior of tantalum was erratic. Relatively thick tantalum is used in EBR-II as cladding for the neutron source. Corrosion of the cladding has resulted in a nuisance activity problem, but the cladding has not failed. The tantalum-clad sources are being replaced with stainless steel-clad sources.

It is necessary to keep in mind two limitations of the Materials Surveillance Program in the absence of additional supporting work, particularly with respect to the LMFBR Program. The first limitation is one of temperature. Many, if not all, of the processes of interest (corrosion and changes in structure and properties) are highly temperature sensitive. Thus, data obtained at 700°F, if not supplemented by additional data at higher temperatures, are of little value to the LMFBR Program. Of more concern is the role of the Surveillance Program as a "watchdog" for EBR-II. Some of

*Failure of these rings was observed before removal of SURV-1.

the EBR-II components are in service at temperatures appreciably higher than 700°F, as shown in Table I. It should be noted, however, that these components are not designed for long service life.

The second limitation is related to resolution of temperature and radiation effects. Two subassemblies of the program are stored within the hot reactor sodium, but in a location where they will not be subject to appreciable neutron flux. However, it is anticipated that these subassemblies will not be removed for examination for several years. Until then, the properties of the samples from these subassemblies cannot be compared with the properties of the samples from other program subassemblies which have received cumulative exposure to both relatively high temperatures and neutron fluences. This limitation is important only in terms of general materials knowledge. It does not compromise judgment as to reliability of EBR-II components.

The Materials Surveillance Program is (or will be) supplemented to a certain extent by other programs* now active or in the advanced planning stage. These programs include: exposure of corrosion samples to flowing sodium in the core at temperatures to ~950°F; examination of fuel-element cladding; examination of control-rod thimbles; inclusion of corrosion samples in high-temperature subassemblies; and inclusion of corrosion samples in the planned radioactive sodium chemistry loop. Although these programs do not include as wide a range of materials as the Materials Surveillance Program, Type 304 stainless steel is included in all of them.

*These programs also are carried out in the EBR-II by the EBR-II Project.

APPENDIX

Other Materials and Conditions of Exposure in EBR-II

The following tables list conditions of exposure for materials in EBR-II that are not exposed directly to primary-system sodium. The conditions of exposure for materials in direct contact with primary-system sodium in EBR-II are listed in Table I of the report.

TABLE A-I. Materials Exposed to Argon Containing Sodium Vapor[a]--Primary System

Material	Approximate Service Temperature, °F
Metals Exposed Directly	
Aluminum alloys 1100 and 6061	120-300
Ampco aluminum-bronze alloys 18-13, 18-23, and 22	500-700
Brass	150-200
Chrome plating on Type 304 stainless steel	250
Colmonoy alloys 4 and 5 on Type 304 stainless steel	500
Inconel 600	200-300
Inconel X-750	700
Music wire	200
Nichrome	700
Oxalloy-28[b]	400-500
Silver solder	120-300
Soft solder	100
Type 17-4 PH stainless steel	400-700
Stainless steel Types 301, 303, 304, 316, 321, and 347	120-700
Stainless steel Types 410 and 420	200-400
Stainless steel Type 430	300-400
Steel--carbon, galvanized, hot-rolled, mild, "stressproof," "accuracy" stock, keystock, AISI-C1045, and AISI-C1141 (also includes items identified only as "steel")	120-700
Stellite 6	700
Nonmetallics Exposed Directly	
Asbestos	120-300
Buna-N	300
Graphitar	300
Graphited asbestos	300
Silicone rubber	120-400
Sodium aluminum silicate	400
Teflon	120-400
Thermon,[c] Grade T-63	300-400
Exposed Only If Type 304 Stainless Steel Jacket Fails	
Aluminum oxide	250
Asbestos	300
Chromel-Alumel	350-500
Copper	150-550
Fiberglass and glass braid	300-500
Heavy concrete	250
Lead	200-300
Magnesium oxide	700
Silver solder	500
Steel--mild, ASTM-A235-55T	200-700
Depleted uranium	300-400

[a]Only those materials exposed at temperatures above the melting point of sodium (208°F) are listed. The concentration of sodium in argon is negligible below 208°F.

[b]Copper conductor sheathed in Type 430 stainless steel.

[c]Thermon is a heat-transfer cement or putty; after curing, it is largely a graphitic material with a small amount of binder, probably organic.

TABLE A-II. Materials That Would Be Exposed to Primary-system Sodium If Their Primary Containment Failed[a]

Material	Component
Aluminum alloys 1100, 3003, 6061, and 6063	Instrument thimbles
Alumina	Instrument thimbles; heat-exchanger thermocouples
Antimony	Source
Antimony oxide	Source
Beryllium	Source
Boral	Instrument thimbles
Borated graphite	Neutron shielding
Carbon steel	Reactor-vessel-cover lift mechanism and a flowmeter
Chromel-Alumel	Heat-exchanger thermocouple
Copper	Pump bus bar
Formica	Pump bus bar
Glass braid	Level control
Graphite	Neutron shielding
Lavite	Pump
Lead	Instrument thimbles
Magnesia	Flowmeter
Mica	Pump
Mild steel	Shield plug
Nylon	Instrument thimbles
Oxalloy-28[b]	Level indicator
NaK	Pressure transducers and shutdown cooling plug
1020 steel	Magnet
Uranium	Mark-III gripper of fuel-unloading machine (FUM)
Zirconium hydride	Instrument thimble

[a]All materials except copper and formica at ~700°F; copper and thermocouples at ~850°F; formica at 400-500°F.
[b]Copper conductor sheathed in Type 430 stainless steel.

TABLE A-III. Materials Exposed Directly to Secondary-system Sodium

Material	Approximate Service Temperature, °F
Inconel 600	770
Type 304 stainless steel	300-860
Type 316 stainless steel	590
Type 347 stainless steel	560-840
Type 501 stainless steel	860
Ferritic steels: $2\frac{1}{4}$ Cr-1 Mo (ASTM Grades A-182, F22; A-213, T22; A-234, WP22; A-335, P22; A-387, D)	580-860
Stellite	590

TABLE A-IV. Materials Exposed to Argon Containing Sodium Vapor[a]--Secondary System

Material	Approximate Service Temperature, °F
Carbon steel	360-500
Carbon steel, galvanized	360
Type 304 stainless steel	360-590
Type 316 stainless steel	590-700
Type 347 stainless steel	590
Stellite	590

[a]Only those materials exposed at temperatures above the melting point of sodium (208°F) are listed. The concentration of sodium in argon is negligible below 208°F.

TABLE A-V. Materials That Would Be Exposed to Secondary-system Sodium If Their Primary Containment Failed

Material	Primary Containment	Approximate Service Temperature, °F
Carbon steel	Inconel rupture discs	770
Chromel-Alumel	Type 304 stainless steel and $2\frac{1}{4}$ Cr-1 Mo ferritic steel thermocouple sheaths	300-860
NaK	Type 304 stainless steel pressure transmitters	590-860

ACKNOWLEDGMENTS

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L. D. Borders coordinated the shop work.

D. M. Cheney and E. A. Perry participated in the dismantling of SURV-1 and the examinations performed in Idaho.

W. D. Jackson participated in the examinations performed in Illinois.

N. Peters devised the instrumentation for gas-sampling of graphite cans, assembled the apparatus for and conducted the tube-burst tests, and conducted the tensile tests.

B