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INVESTIGATIONS OF MATERIALS COMPATIBILITY RELEVANT TO THE EBR-II SYSTEM--FY 1975

by

K. J. Longua, W. E. Ruther,
J. A. Shields, and A. F. Clark

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EBR-II Project

May 1976

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ABSTRACT

Due to the continuing low incidence of failure and replacement, no failed components were available during FY 1975 for destructive examination.

The annual inspection of the steam-generating plant is described. No abnormalities were noted, and there was no evidence of material degradation. The preliminary investigation of the $2\frac{1}{4}$ Cr-1 Mo ferritic pipe removed from the system indicated that the material had not suffered any significant degradation of mechanical properties.

Examination of a tantalum-clad photoneutron source showed the cladding had performed well in contact with 900°F (480°C) flowing sodium.

Stainless steel specimens (Type 304) removed from a secondary sodium system vent line did not reveal any abnormalities.

New materials for thermal-expansion-difference (TED) monitors were evaluated. Hastelloy-X material appears to be superior to Type 304 stainless steel for this application.

Experiments were conducted to determine possible sources of methane found in the EBR-II cover gas. The residual cleaning agent left on new fuel subassemblies produced methane upon immersion in 700°F (370°C) sodium. In another experiment, the on-line monitoring of the EBR-II cover gas for methane or other hydrocarbons was proved feasible.

A special gauge subassembly, designed to measure the distance between the core grid plate and the holddown finger, was tested in 700°F (370°C) sodium.

The visual examination of the annulus between the large rotating plug and the seal support trough revealed that a relatively hard and friable deposit completely filled the top 8-10 in. of the annulus.

I. INTRODUCTION

This report is the sixth in a yearly series[1-5] on investigations of components of the EBR-II reactor plant and on out-of-reactor experiments in which materials or techniques are tested before they are used in the EBR-II reactor system. Wide ranges of materials, circumstances, and evaluation techniques are involved. Therefore, each investigation is reported separately.

Reference 1 contains general information of the configuration of the EBR-II reactor plant, sodium chemistry, and sodium-removal techniques.

A series of reports on the material-surveillance subassemblies examined to date gives additional information on mechanical properties and compatibility of materials exposed in the blanket and fuel-storage regions of the EBR-II reactor plant.[6-8]

II. DATA AND RESULTS

A. Examination of Components

The opportunities for examination of components of EBR-II are limited to those removed for maintenance (or replacement) and to those that have been modified to provide inspection ports. The yearly inspections of the steam-generating system, initiated in this series last year, are being continued to provide up-to-date information on service experience and to provide complete information on the materials used in the EBR-II.

1. Examination of the Main Condenser

Both the cooling-tower water side and the steam/condensate side of the main condenser were visually examined and photographed. The present condition was compared with photographs obtained during the previous year's examination. No evidence of abnormal pitting or corrosion on the interior walls, supports, condenser tubing, or the distribution header for the desuperheater line was revealed. The distribution header and the steam side of the Admiralty metal tubes were lightly covered with red rust that was presumed to be Hematite.

Evidence of steam erosion was observed on the condenser supports next to the distribution header of the steam desuperheater. When this erosion is compared with photographs taken during previous years, no increase in the erosion depth was evident. The amount of erosion present is minor and is not detrimental to the supporting structure.

Inspection of the condenser top as viewed through the turbine inspection port indicated minor erosion on the tubing and supports. This erosion

is caused by wet steam exiting from the last stage of the turbine and impinging on the top area of the condenser. Small areas of erosion were also noted on the turbine casing. These eroded areas are shallow and are not considered detrimental.

The general appearance of the water boxes, which direct cooling-tower water to the condenser tubes, was similar to that during the previous year's examination. Deposits of hard, tightly adhering material were noted in the water boxes. This deposit is located on the bottom and sides of the water boxes and is thought to be a mixture of the hard-water elements and iron oxide. It does not interfere with condenser operation. The Admiralty metal tubes, which carry the cooling water between the water boxes, were clean without any evidence of fouling, corrosion, erosion, or dezincification. The Muntz metal tube sheet appeared normal.

2. Examination of the Steam Drum

The steam drum was first inspected in August 1962 and since then has been inspected periodically through 1975. The steam drum is subjected to visual inspection, photography, chemical analysis of deposits, and the use of corrosion coupons.

During these inspections, the inner walls of the drum, the moisture separators, internal fittings, and corrosion coupons are examined. The steam-drum walls and internal fittings were coated with a red-rust-type deposit. No evidence of gross corrosion or material degradation was evident. The only notable difference during this year's examination was a smaller accumulation of loose material in the drum. No change in the condition of the steam drum can be observed when photographs taken during the previous inspection are compared with those taken of the same areas during the most recent investigation.

Samples of corrosion products taken from each end of the steam drum were chemically analyzed to aid in evaluating the condition of the rest of the steam system. Results of the chemical analyses for this year, as well as the results from the previous year, are in Table I. The major change in composition was the amount of iron present. The adherent corrosion products may be higher in iron than are the loose corrosion products or nonadherent deposits. To try to resolve this difference, future inspection will include separate analysis of adherent and nonadherent corrosion products.

TABLE I. Analyses of Corrosion Products

		1975 Content				1975 Content	
Element	1974 Content	East End	West End	Element	1974 Content	East End	West End
Fe	8.7 wt %	32.7 wt %	36.9 wt %	Ca	0.2 wt %	0.12 wt %	0.03 wt %
Cr	0.6 wt %	0.16 wt %	0.16 wt %	Mn	0.3 wt %	0.3 wt %	0.3 wt %
Ni	7.2 wt %	14.6 wt %	9.9 wt %	Mo	<275 ppm	<27 ppm	<20 ppm
Cu	31.6 wt %	29.5 wt %	39.9 wt %	Sn	<1250 ppm	<785 ppm	<600 ppm

The two corrosion coupons, which are $2\frac{1}{4}$ Cr-1 Mo carbon steel, were removed, examined, and replaced. Comparison of these coupons with previous examinations reveals essentially no change. The mill marks, the stamped number, and the irregularities appear unchanged. This was expected and confirmed by the lack of significant weight change. The weight of each coupon is given in Table II. The slight weight gain occurring is the result of buildup of the adherent corrosion film. It is evident from these examinations that the water treatment and chemistry control presently used have successfully protected the steam system from serious corrosion.

TABLE II. Weight of Corrosion Coupons

Date Examined	Coupon in Steam Phase, g	Coupon in Water Phase, g
5/8/72	533.9278	536.5701
4/25/73	533.9652	536.5713
5/6/74	533.9629	536.5791
4/15/75	533.9839	536.5831

3. Hardness Tests on Evaporator and Superheater Shells

To try to estimate the mechanical properties of the $2\frac{1}{4}$ Cr-1 Mo carbon steel used in construction of the EBR-II steam generators, hardness tests were performed on the superheater and evaporator shells using a Newage "Multi-Range" portable hardness tester, which was equipped with a Rockwell "C" scale indenter. Tests were made with the system depressurized, drained of water and sodium, and at ambient temperature [60-70°F (16-21°C)]. The surfaces were prepared by polishing the shell surfaces with a series of silicon carbide abrasive papers through No. 600 grit. Tests were performed at the bottom of each shell, at the approximate level of the centerline of the sodium inlet or outlet pipe.

Cahoon et al.[9] have presented a method to calculate the values of yield and tensile strength for materials. This method is based on hardness test results. Using an appropriate estimate for the strain-hardening coefficient,[10] the equations are

$$\sigma_y = 0.206H \tag{1}$$

and

$$\sigma_u = 0.339H, \tag{2}$$

where

H = diamond pyramid hardness,

σ_y = yield strength (ksi),

and

σ_u = ultimate strength (ksi).

These equations were developed to describe the relationship between yield strength, tensile strength, and hardness for nonferrous and ferrous materials alike. Other empirical relations between hardness and tensile strength have been developed for steels in general[11] and for $2\frac{1}{4}$ Cr-1 Mo steel in particular.[12] Reference 11 presents the data in tabular form. Reference 12 has developed a correlation of the form

$$\sigma_u = 0.482H. \tag{3}$$

This relationship has been found to hold for a variety of heat treatments, carbon contents, and product forms. In addition, the correlation coefficient between tensile strength and hardness has been found to be 0.98. This relation probably represents the "best estimate" of the tensile strength for the steam-generator shells.

Table III summarizes the calculations of room-temperature yield and tensile strengths obtained by using these relationships.[9] These values lie within and toward the upper limits of the scatter bands observed for virgin annealed $2\frac{1}{4}$ Cr-1 Mo carbon steel, indicating that 10-yr service has not resulted in gross changes in mechanical properties.

TABLE III. Yield and Ultimate Tensile Strengths of Evaporator and Superheater Shell Material Calculated from Hardness Measurements

Unit	Hardness, R_c	Hardness, DPH	Strain Hardening Coefficient, n	Calculated Yield Strength,[9] ksi	Calculated Ultimate Tensile Strength, ksi, from		
					Ref. 9	Ref. 11	Ref. 12
EV 701	20.3 ± 3.5	240 ± 15	0.21	49.3 ± 3.1	81.3 ± 5.1	110.0 ± 10.3	115.7 ± 7.2
SU 712	19.8 ± 4.1	238 ± 20	0.21	48.9 ± 4.1	80.6 ± 6.8	109.9 ± 11.8	114.7 ± 9.6

4. Material Investigations of $2\frac{1}{4}$ Cr-1 Mo Steel Steam Pipe

A 5-ft segment of $2\frac{1}{4}$ Cr-1 Mo carbon steel piping (ASTM A-335-55T, Grade P-22) was removed from a 6-in.-dia superheated-steam riser (No. S7-33-521) to determine its current mechanical properties. The pipe has been in service since construction of the reactor and operates at about 820°F (438°C) with an internal pressure of 1250 psi. The piping material has been subjected to impact testing and fractographic examinations to determine the effects of service on this material.

Charpy V-notch (CVN) impact specimens were machined in L-R (longitudinal direction with a radial crack plane) and C-R (circumferential direction with a radial crack plane) orientations. These specimens were tested in accordance with ASTM E-23, using instrumented impact techniques. In addition, specimens taken from the pipe were fractured in vacuo (10^{-10} Torr) at about -230°F (-146°C), and the fracture surfaces were examined by Auger Electron Spectroscopy (AES).

Figures 1 and 2 plot the impact data obtained on the material in the L-R and C-R directions, respectively. The Nil-Ductility Transition Temperature (NDTT), as defined by the knee of the lower-shelf region, is about -50°F (-46°C) in the L-R orientation and -90°F (-68°C) in the C-R orientation. The NDTT determined on virgin annealed material is about -100 to 0°F (-73 to -18°C).[13,14] This indicates that there has been little, if any, change in the NDTT with service. The two orientations exhibit differing upper-shelf

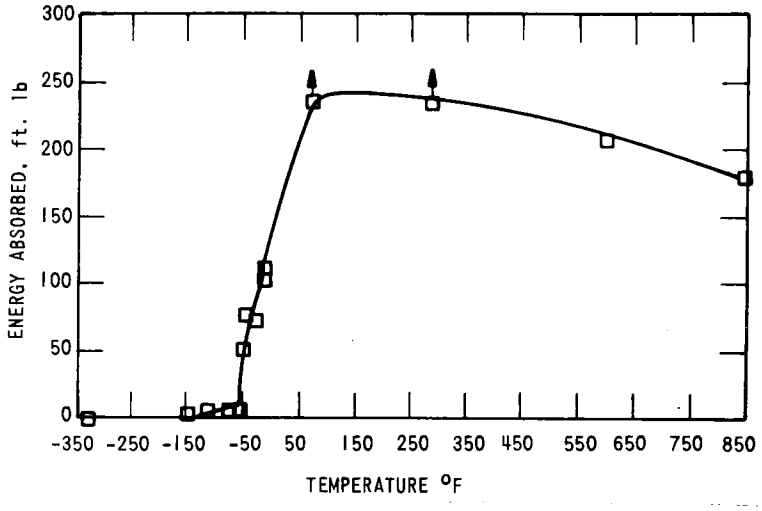

Fig. 1. Longitudinal (L-R) Specimen Charpy V-notch Results

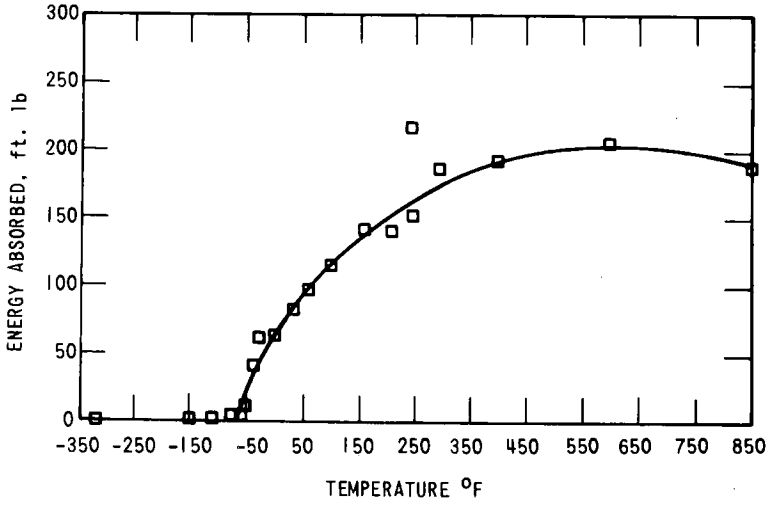

Fig. 2. Transverse (C-R) Specimen Charpy V-notch Results

and transition behavior; the C-R orientation has a much broader, less steep transition region, and the L-R orientation attains much higher energy values on the upper shelf. In fact, the L-R orientation was capable of stopping the hammer on the 240-ft-lb impact tester used in the investigation. The decrease in absorbed energy for the L-R orientation with increasing temperature is not a true effect. Instead, it is a result of increased ductility with increasing temperature, which allowed the specimens to deform and be drawn through the anvils of the tester without fracturing.

In conjunction with the impact investigations, specimens were prepared for in-vacuo fracture experiments using AES techniques. The primary purpose of this investigation was to determine the presence and extent of embrittling agents (phosphorus, antimony) on fracture surfaces of the material. In general, the fracture surfaces observed in this portion of the investigation were transgranular and ductile. Very little intergranular fracture area was observed, and phosphorus could be observed on fracture surfaces only occasionally. Figure 3 is a typical spectrum obtained from an area containing phosphorus. In addition to the peaks observed for the major alloying elements, a phosphorus peak is seen at 120 eV. Figure 4 is a concentration-versus-depth profile of the same region from which Fig. 3 was obtained. The surface composition of phosphorus in this area is about 3%, and the enriched layer is only about 15 Å deep. We cannot say unequivocally that the phosphorus observed in this area is the result of aging processes occurring during service. It may well be the result of thermal cycles in processes used to fabricate the pipe.

In summary, the preliminary investigations on material removed from the superheated steam riser S7-33-521 indicate that the $2\frac{1}{4}$ Cr-1 Mo carbon steel used in construction of the EBR-II steam system has not suffered gross degradation in mechanical properties. Detailed reports of the full series of investigations will be published in the future, when the test program is completed.

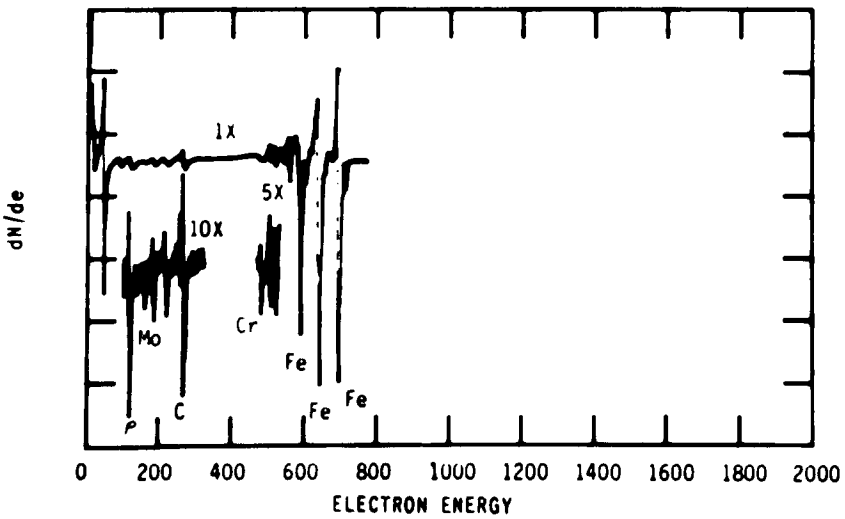

Fig. 3. Typical Auger Electron Spectrum from Phosphorus-containing Region (1X, 5X, and 10X refer to multiplication factors applied to detector signals)

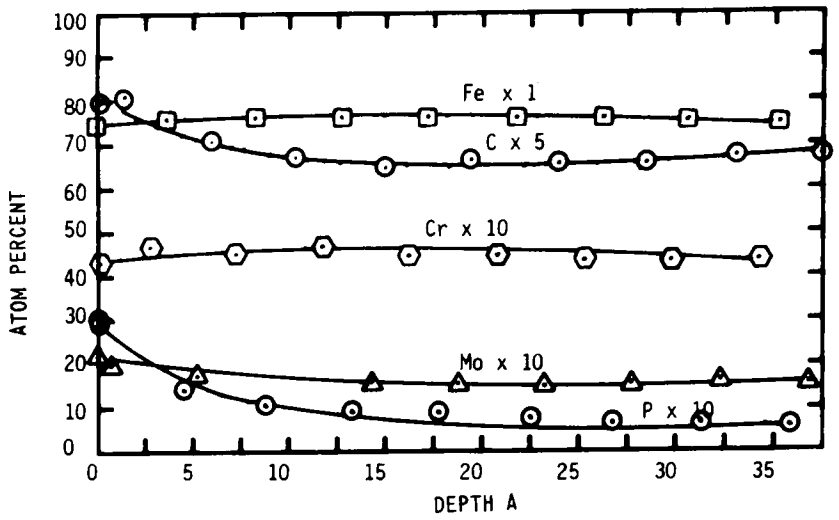

Fig. 4. Profile for Region in Which Spectrum of Fig. 3 Was Obtained (X 1, X 5, and X 10 refer to multiplication factors applied to detector signals)

5. Examination of Photoneutron Startup Source

The photoneutron sources are used to initiate neutron production during the startup phase of the reactor and to provide a reliable neutron-flux level for comparison as operation is started. The photoneutron startup source consists of a source thimble containing a beryllium annulus and a source rod containing an antimony metal cylinder; this antimony metal cylinder is clad in tantalum and is the last of its type. It was installed in the core on June 23, 1964, and was removed September 27, 1974. During its residence in the reactor, the source was exposed to 82,650 MWd and reached a fluence of 7.4×10^{22} n/cm^2 total. The sources presently in use are antimony metal clad in tantalum with a final cladding of stainless steel.

Visual examination of the tantalum cladding did not indicate any gross corrosion or erosion to be present on the outside surface. Postirradiation dimensions at the core midplane indicate a diameter of 0.9124 in. and a wall thickness of 0.0608 in. The only comparisons for these dimensions are the nominal drawing dimensions, as the actual fabrication dimensions are unavailable. The drawings are 0.914 ± 0.005 in. in diameter and 0.060 ± 0.005 in. thick.

Swelling of the tantalum due to irradiation is minimal as measured by the density change. Table IV shows the density of the cladding near the top of the core, core midplane, and core bottom. As noted in this table, the density appears to increase rather than decrease. One explanation for this anomaly may be the production of tungsten from the tantalum (nβ) reaction.

The tantalum cladding performed well in contact with 900°F (480°C) flowing sodium. However, in sodium removal and subsequent handling, its

intensely radioactive corrosion product becomes airborne and causes difficulty with alarms and filters. For this reason, the current sources do not expose tantalum to the coolant.

TABLE IV. Density of Irradiated Tantalum

Location	Density, g/cm^3
Top of core	16.70
Core midplane	16.72
Bottom of core	16.59
Unirradiated (literature value)[7]	16.69

6. Examination of Sample from EBR-II Secondary Sodium System

A section of Type 304 stainless steel pipe (2-in. nominal diameter) was removed from the secondary-sodium-system vent line during the major shutdown in the winter of 1968-1969. It was replaced with a removable section containing three Type 304 specimens measuring 8 by 1 by 0.062 in. Until the recent major shutdown, these specimens were exposed to sodium flow at 3-5 gpm. The sodium temperature was about 830°F (443°C) when the reactor was operating and dropped to about 580°F (304°C) during shutdown.

The sample holder and the specimens were cleaned of sodium by immersion in ethyl alcohol and then in deionized water. Both solvents were filtered, and no measurable residue was recovered.

The stainless steel specimens had areas of very thin, dark staining that could be removed easily with a soft rubber eraser. One specimen was sacrificed for metallographic examination and carbon analyses.

An etched transverse section revealed slight grain-boundary enrichment in carbon to a depth of about one grain. This is reflected in the carbon analysis shown in Table V.

TABLE V. Carbon Content vs Depth of Analyzed Layer of Type 304 Stainless Steel

Layer Depth, mils	Carbon Content, wt %	Layer Depth, mils	Carbon Content, mils
0-3	0.083	6-9	0.060
	0.108		0.062
3-6	0.052	9-12	0.065
	0.0468		0.061

Similar slight surface enrichment in carbon is frequently noted for stainless steel surfaces in the as-received condition. However, the piece of pipe originally removed in 1969 showed a similar slight increase[1] in carbon content of its surface layer exposed to sodium. In any event, the slight increase in carbon content of this thin layer would have no measurable effect on the mechanical properties of the secondary piping.

B. Out-of-reactor Testing in Support of EBR-II

Investigating the effects of sodium exposures and high temperatures on materials to be used in EBR-II is often advantageous. The results of these out-of-reactor exposures permit a realistic evaluation of the risk, if any, that subsequent in-reactor experiments impose on the reactor. Four such investigations were completed in the past year.

1. New Materials for TED Passive Temperature Monitors

Thermal-expansion-difference (TED) monitors are based on the thermal expansion of sodium being greater than that of the material of a surrounding, completely filled, capsule. The capsule is first filled with sodium and seal-welded in a vacuum. It is then heated to a reference temperature and cooled, and its volume is accurately determined by the immersion technique. When it is heated above the reference temperature and cooled, the volume change due to plastic deformation of the capsule material is a measure of the temperature rise.

Several hundred TED monitors have been manufactured with Types 304L or 321 stainless steel as capsule material because of the availability of the thin-walled tubing required. However, these materials swell in high-fluence reactor environments. This irradiation-induced swelling is in addition to plastic deformation from sodium expansion. Corrections can be made, but uncertainty in the calculated temperature rises rapidly at fluxes above about 1×10^{22} n/cm^2.

Other alloys show significantly lower void-swelling tendencies than stainless steels when bombarded with 5-MeV nickel ions. For example, in the study reported,[15] swelling of both a ferritic alloy and of Hastelloy-X was less than that of Type 304 stainless steel by more than two orders of magnitude. If the same order of improvement would be obtained in a neutron flux, using these materials for the capsule wall would significantly improve the TED monitor.

Hastelloy-X alloy was investigated first. Initially, the small pieces of tubing (0.174-in. OD and 0.009-in. wall) used in the monitor were manufactured from rod stock. TED monitors made with this tubing burst after the first or second increase in temperature. Metallography indicated that the grain size of the tubing was so large that the tubing wall was only two to five grains thick.

Some Hastelloy-X fine-grained tubing was obtained in a different size (0.220-in. OD and 0.016-in. wall). Monitors made with this tubing did not deform as expected at the lower temperature [≤450°C (840°F)], as required for a TED monitor. The tubing also showed considerable creep when held at temperatures as high as 700°C (1290°F). The difficulty with these monitors was that the capsule wall was too strong, being nearly twice as thick and made of an alloy roughly twice as strong as Type 304 at high temperatures.

Finally, TED monitors were manufactured from the same tubing source of Hastelloy-X, but the tubing was mandrel-turned to a wall thickness of 0.005 in. These monitors performed well in their calibration, as shown in Fig. 5.

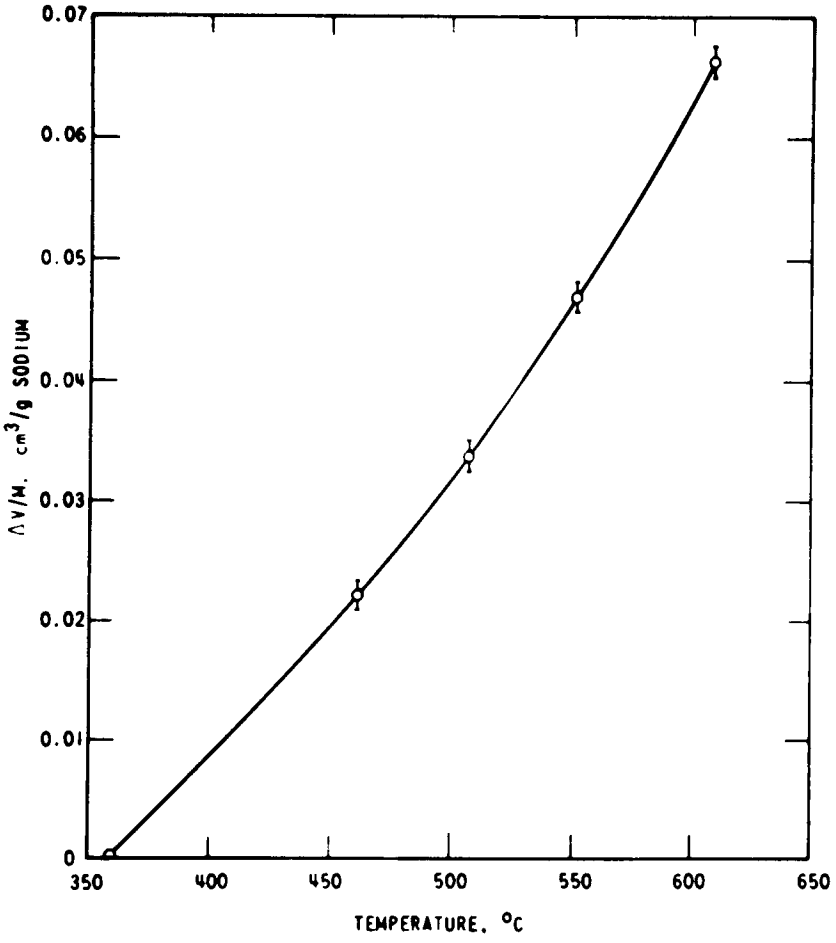

Fig. 5. Calibration Curve for Thin-walled Hastelloy-X TED Monitors (average and range for three monitors)

An attempt was made to fabricate monitors from Type 430 ferritic steel. This type was chosen because of its ductility, its reasonable resistance to water corrosion, and its resistance to irradiation-induced swelling. The tubing required (0.174-in. OD and 0.009-in. wall) was made from a fine-grained rod.

TED monitors made from this steel performed well for the first two temperature increases, but burst as they were being heated from

417 to 450°C (783 to 840°F). This temperature is in the temper-embrittlement range, but normally the brittleness would be noted only when the alloy was cooled below about 100°C (212°F). A metallographic examination showed a uniform fine-grained tubing-wall thickness; thus, the reason for the failure remains obscure.

2. Experiments on Source of Methane in Cover Gas

Out-of-reactor immersion experiments in sodium were conducted as an aid in determining the source of the methane in the cover gas of EBR-II. An initial concern was that one or more of the graphite-containing cans in EBR-II had sprung a leak, and that methane was formed by reaction between sodium and an impurity in the graphite.

The experiments were performed in a small sodium loop with a sodium volume of 4 liters and an argon-cover-gas volume of about 1.2 liters. A cold trap at 230°F (110°C) maintained a low level of oxygen and hydrogen in the system. Samples of the argon were taken before and after immersion of various items in the sodium and analyzed for methane. The results are shown in Table VI.

TABLE VI. Methane Produced by Immersion in Sodium

Experiment No.	Material	Sodium Temp, °F (°C)	Time, hr	CH_4 Conc,[a] ppm
1	Archive graphite in "fresh" stainless steel basket	700 (370)	0	5
		700 (370)	1	90; 117
		700 (370)	25	77; 74
		700 (370)	49	46; 39
		700 (370)	193	26; 26
		900 (480)	1	20; 24
2	Archive graphite in "exposed" stainless steel basket	700 (370)	0	1; 2
		700 (370)	1	20
		900 (480)	1	32
3	Fuel-element tubing wiped with acetone	700 (370)	0	1; 2
		700 (370)	1	11; 17
4	Fuel-element tubing oxidized and wiped with ethanol	700 (370)	0	2; 3
		700 (370)	1	25; 38

[a]Two values listed on the same line are for duplicate samples taken within a few minutes of each other.

Archive samples of EBR-II graphite were used in the first two experiments. The graphite was machined into solid cylinders about 3/4 in. in diameter and $3\frac{1}{2}$ in. long, avoiding all contact with organic materials. Since the extent of the sodium-graphite reaction was unknown, a fine-mesh stainless steel basket was used to contain each sample.

In the first experiment, the graphite sample was immersed for 193 hr in the sodium at 700°F (370°C), and the system argon was periodically sampled. About 100 ppm of methane were released in the first hour

(see Table VI). The evolved methane then reacted slowly with the sodium to gradually reduce the methane concentration in the argon to about 26 ppm after 193 hr. The sodium then was heated over a 10-min period to 900°F (480°C) and held at that temperature for 1 hr. The methane concentration in the argon further decreased to about 20 ppm through accelerated reaction with the higher-temperature sodium. No additional methane was generated by the increase in sodium temperature.

The second experiment was performed after the system argon was completely replaced with fresh gas. A second sample of graphite confined in the same ("exposed") basket was immersed in sodium at 700°F (370°C); the system argon contained 20 ppm methane after 1 hr. The sodium temperature was rapidly increased to 900°F, and after 1 hr at that temperature, the methane concentration was 32 ppm.

Comparison of the results of the two experiments suggested that a major portion of the evolved methane was produced by the initial contact of the stainless steel basket with the sodium. Therefore two additional immersion tests were performed with fuel-element tubing to determine if the normal addition of new subassemblies to the EBR-II primary tank could produce significant methane.

One specimen of fuel-element tubing with wire wrap attached was cleaned with technical-grade acetone. A second specimen was heated for 1 hr in air at 930°F (500°C) to simulate the fuel-bonding cycle, and then was wiped with ethanol according to the appropriate procedure. Each specimen was immersed separately in 700°F (370°C) sodium. Table VI shows that, although both surfaces produced methane upon immersion, a greater amount was produced from the ethanol residue on the oxidized surface of the second sample.

Thus, the insertion of fresh subassemblies, cleaned according to standard procedure, into the EBR-II tank likely explains the methane noted in the cover gas. Furthermore, since there was no rapid or extensive evolution of methane (in experiment 2, particularly) during immersion of archive graphite in 700°F (370°C) sodium, the presence of methane in the cover gas of EBR-II does not indicate leaks in the graphite cans.

3. Monitoring the Hydrocarbon Level of EBR-II Cover Gas

As indicated in Sec. 2 above, on-line monitoring of the EBR-II cover gas for methane would provide timely information on the amount of organic residue being added with the fresh subassemblies and would be an effective early warning of possible in-leakage of pump oil.

A Model 400 hydrocarbon analyzer was obtained and assembled into a simulated on-line system as shown in Fig. 6. The output of the hydrocarbon analyzer depends on the flow rate of the sample gas. A mass flowmeter accurately measures this flow, which is adjusted to the desired value with a metering valve. The specifications of the gases used are given in Table VII.

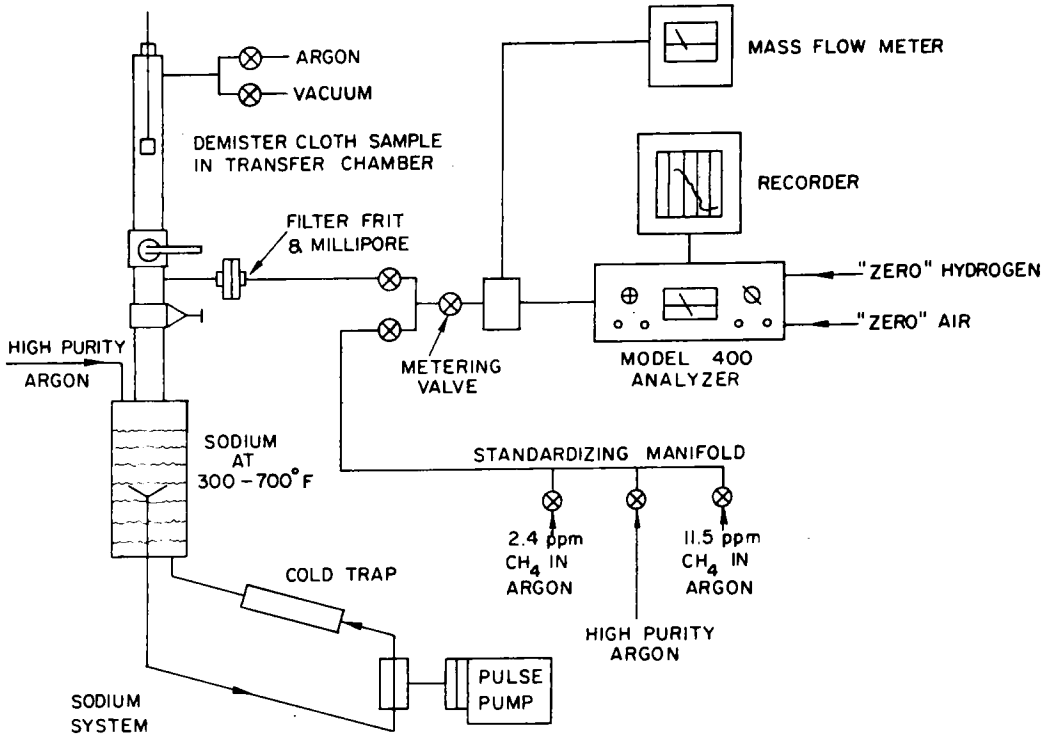

Fig. 6. Schematic Diagram of Test of Simulated On-line Analytical System

TABLE VII. Gases Used with Hydrocarbon Analyzer

Air	Ultrazero grade, less than 0.1 ppm hydrocarbon content, 220 ft^3, Certified Standard
Hydrogen	Zero grade, less than 0.5 ppm hydrocarbon content, 216 ft^3, Certified Standard
Mixed gas	Methane in argon, 2 ppm CH_4, 32 ft^3, Certified Standard
Mixed gas	Methane in argon, 10 ppm CH_4, 32 ft^3, Certified Standard
Argon	Ultrahigh purity, 290 ft^3

The sodium system used was a small (4-liter) pumped system with cold trap and access port. The volume of the cover gas was about 1.2 liters. The temperature of the sodium was varied from 300 to 700°F (150 to 370°C).

It was quickly determined that sodium aerosol interfered with the operation of the analyzer. An E-grade stainless steel frit filter was inadequate to remove the aerosol. A Millipore Type HA filter disk (calculated pore size, 0.45 μm) completely eliminated the aerosol, and this resulted in a smooth and relatively stable hydrocarbon analyzer output with the sodium temperature at 700°F (370°C). A hydrocarbon level of 3 ppm was indicated. Over the next weeks of testing, a variation of 2-4 ppm was noted for the cover gas.

An experiment was performed to determine the behavior of the system when organic residue was introduced into the sodium. A small roll

of York Demister (stainless steel) cloth used for the new EBR-II secondary cold trap was lowered into the 700°F (370°C) sodium with the hydrocarbon analyzer operating on the cover gas. The trace of the response of the analyzer is shown in Fig. 7. The hydrocarbon level, initially 4 ppm, peaked at 110 ppm in 1 hr and returned to 3 ppm overnight.

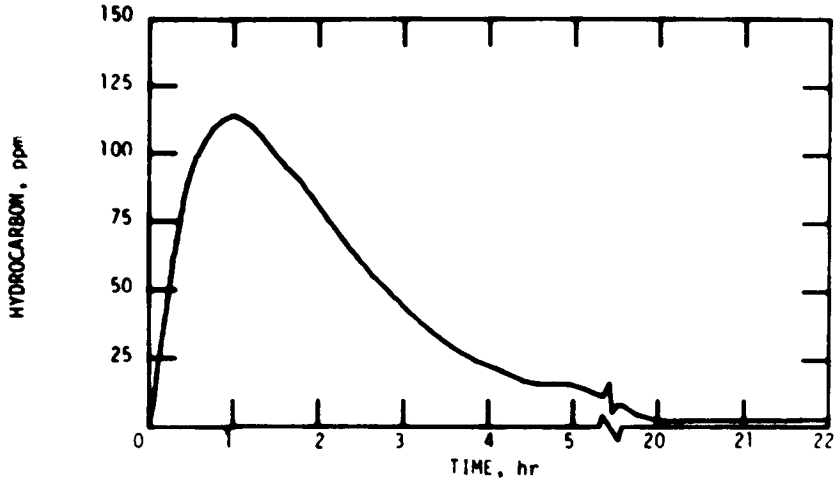

Fig. 7. Response to the Addition of Demister Cloth to Sodium

Two additional simple experiments were performed to determine calibration linearity and sensitivity of the analyzer to sample flow rate. The results are shown in Figs. 8 and 9. Calibration linearity, based on the very limited number of available gases, appears to be satisfactory. The requirement for precise control of flow of the sample stream is evident in Fig. 9.

The major difficulty experienced with the hydrocarbon analyzer was zero drift, which was primarily due to changes in the ambient temperature of the internal electronics. This problem will be alleviated in the reactor installation by placing the electronics in an air-conditioned room.

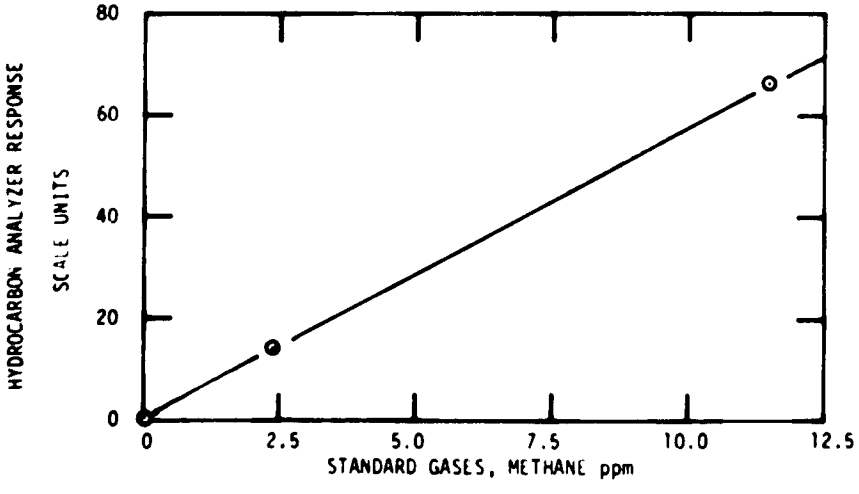

Fig. 8. Calibration Linearity, Mass Flowmeter at 8.00 cc/min

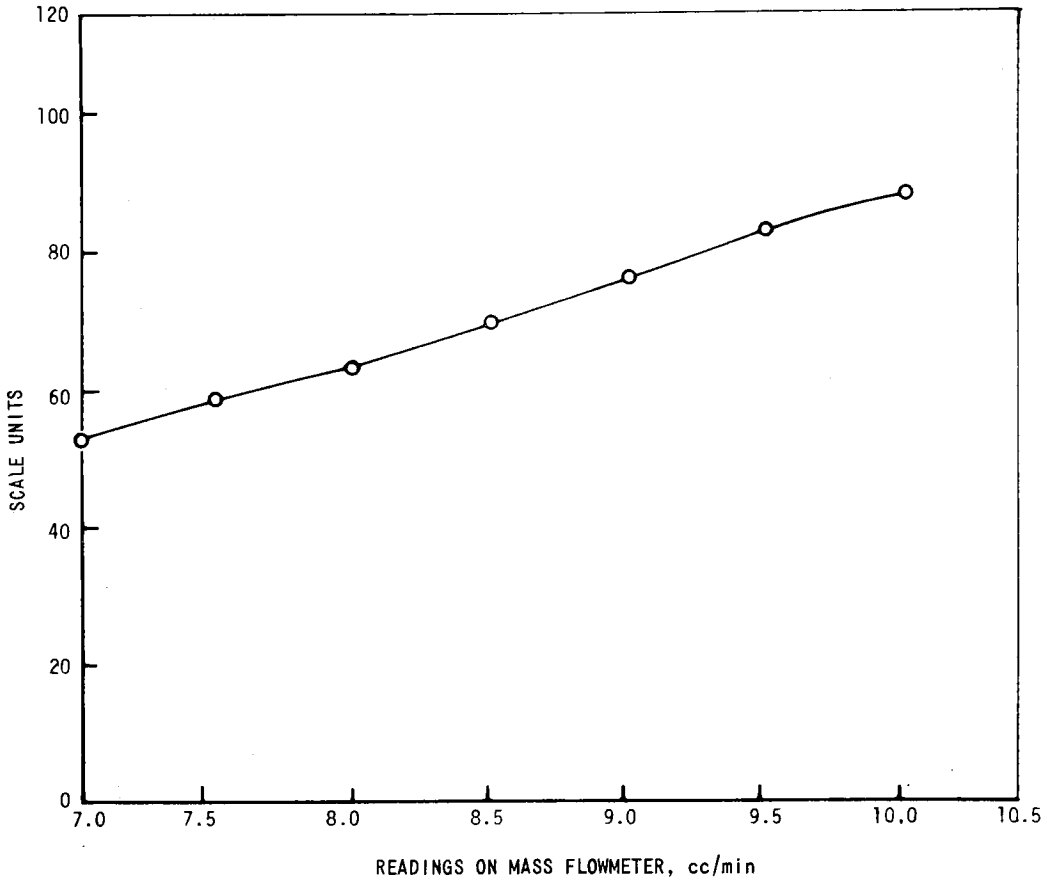

Fig. 9. Hydrocarbon Analyzer Response as Function of Sample Flow Rate

4. Gauge-subassembly Testing in High-temperature Sodium

A special measuring gauge, called a gauge subassembly (GSA) has been designed and fabricated. It consists of a scratching stylus and a metallic recording disk on a ratchet. They are arranged so that depressing the upper adapter of the subassembly causes a scratch on the disk equal to the distance of depression. Raising the upper adapter causes the disk to ratchet into a new position for recording the next depression. The GSA will be used to measure the actual length between the reactor support grid (which supports the subassemblies) and the core holddown fingers located in the core cover. To ensure its satisfactory performance in EBR-II, the GSA was tested in high-temperature sodium under conditions identical to its proposed in-reactor use.

The test equipment consisted of a length of 3-in.-dia stainless steel pipe with a flanged joint at the top. Provisions were made on the flange for a shielded thermocouple, an electrical probe for sodium level detection, and a tube for entry of argon. An O-ring slip seal in the center of the flange permitted actuation of the GSA without loss of cover gas. A rack-and-pinion-driver mechanism was bolted to the flange joint to provide the force and to measure displacement. The bottom of the 3-in. pipe was connected to a sodium reservoir tank by means of tubing and valve. Both the cover gas in the 3-in. pipe and the reservoir could be varied in pressure from a vacuum to 5 psig to aid in transferring sodium between the 3-in. pipe and the reservoir.

The GSA was assembled with a metallurgically polished, stainless steel disk and placed in the 3-in. pipe and aligned by means of a perforated ring slipped on the GSA. The rack-and-pinion mechanism and the flange were bolted in place, and the gas and electrical connections were completed.

The test enclosure was evacuated and purged twice with argon before a pressure of 5 psig was established. The test enclosure was then brought to the desired 700°F (370°C) by external heaters and controllers. Sodium was transferred to a level about 2 in. over the top of the upper adapter of the GSA. Temperature in the test enclosure was then stabilized at 700°F (370°C).

The O-ring slip seal in the flange was loosened slightly to minimize binding of the vertical shaft. A series of accurately measured (dial indicator) displacements was performed by smoothly rotating the handle on the pinion gear. The speed of rotation was chosen to provide displacement rates approximating the 18 in./min used to lower the EBR-II reactor cover. In some instances, the displacements were held at their maximum for about 30 sec; in others, the rotation of the pinion was reversed as soon as the desired dial indicator reading was obtained.

After being removed from the test equipment, the GSA was transferred in air and immersed in an ethyl alcohol bath to remove the adherent sodium. It was then disassembled, and the individual parts were cleaned in water.

The overall lengths of the scratches on the disk were measured on a microscope equipped with a micrometer stage. The width of the scratch was also measured, and the dimension of the displacement was taken as the overall length minus the width, because this value represented the center-to-center distance of travel for the stylus. In general, the agreement between actual displacements and those calculated from the scratch length was quite good. During 60 experimental displacements in sodium, only one measurement was in error more than 15 mils. The GSA also performed well in August 1975 during its first reactor use.

C. Investigation of Problems Related to the EBR-II Rotating Seal

Fuel handling in EBR-II is made possible by two rotatable plugs in the top of the primary tank (see Fig. 10). The gas seal of each plug is maintained by immersing a circular stainless steel blade, attached to the plug, in a trough of liquid eutectic (bismuth-42% tin) alloy that is attached to the primary-tank cover on the large plug. The heat required to keep the alloy liquid is supplied by tubular heaters inserted in holes drilled in the circular blade.

1. Seal Cleaning

Periodic seal cleaning was conducted on the air side of both the large and small plugs in FY 1975. The technique used is to scrape, skim, and

dip through the new side access holes, and remove oxidation-product dross of the bismuth-tin alloy. This technique reduces the quantity of material removed compared to that removed by the brush-cleaning technique, because only a minimum of alloy itself is removed. As a result, only 295 lb of alloy was added to the plugs during FY 1975 as compared to 2000 lb in 1969.

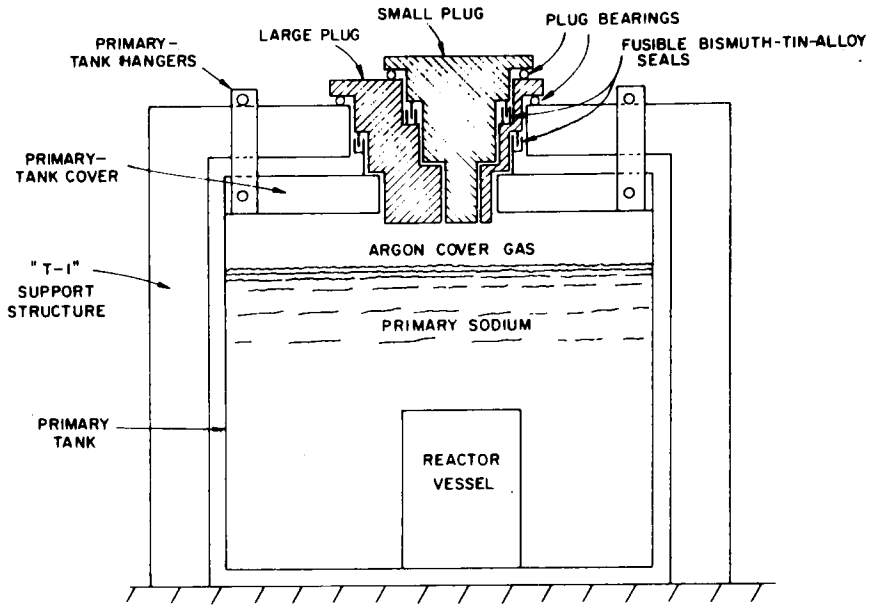

Fig. 10. EBR-II Rotating Plug

2. Investigation of Cover-gas Side of Seal

In the spring of 1974, a 3-in.-dia hole was drilled from the top of the large plug to provide access to the annulus between the large plug and the seal support (see Fig. 11). Visual observation indicated that a relatively hard and friable deposit completely filled the top 8-10 in. of the annulus. The presence of this material probably results in a continuing problem of "sticking" of the large plug during attempts to achieve rotation.

Samples of the material were withdrawn, and an analysis was performed. The results of analysis from four samples is shown in Table VIII. The melting point of this deposit was determined to be greater then than 430°C (800°F), so attempts to melt it out are not feasible in its location. A mechanical device is being designed to remove this deposit.

III. DISCUSSION

The material used in the EBR-II reactor plant indicated very little degradation after 11 yr of service. The corrosion and erosion rate in the steam-generating system is negligible. The interaction of sodium coolant with the remaining materials is immeasurably slow at the temperatures and oxygen concentrations present in the EBR-II systems.

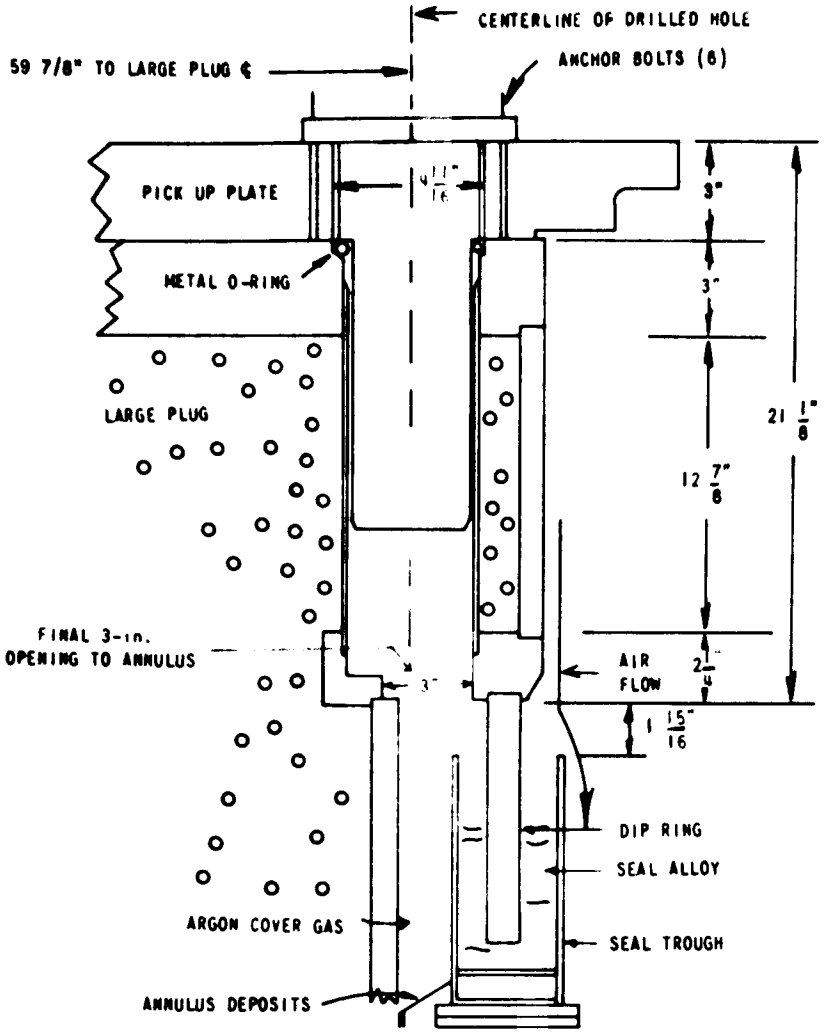

Fig. 11. Location of 3-in. Access Hole in Large Rotating Plug

TABLE VIII. Analysis of Deposit Found in Seal

Element	Content	Element	Content
Sodium	40%	Iron	300-1450 ppm
Bismuth	34-40%	Nickel	72- 246 ppm
Tin	14-22%	Chromium	113- 400 ppm

The nil-ductility transition temperature of the $2\frac{1}{4}$ Cr-1 Mo carbon steel used in the evaporators and superheaters remains very low; this indicates a lack of embrittlement due to service or change in material properties.

No significant downtime of the EBR-II reactor occurred because of material problems in FY 1975, although minor difficulties were experienced with the rotating seal.

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