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WATER CHEMISTRY AND FUEL ELEMENT SCALE IN EBWR

by

C. R. Breden, I. Charak,
and R. H. Leyse

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C. R. Breden, I. Charak, and R. H. Leyse

Reactor Engineering Division

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ABSTRACT

The first two sections summarize investigations in EBWR concerned with some aspects of water chemistry. The results of many of these investigations have not been previously published in a form given wide distribution. Included are studies of water conditions, corrosion products (composition, activities, transportation, deposition and distribution), water dissociation, water activity, fission product release, and buildup of plant activity.

The last two sections of the report give the results of studies of the heat transfer characteristics of fuel element scale and effects of high-temperature heating on scale removal and fuel element growth. The maximum scale thickness measured was about 0.008 in. Heat transfer calculations based on scale thermal conductivity measurements indicate the possibility of maximum fuel temperatures as high as 1692°F at 100-Mw operation of the core. This temperature is in a range where fuel growth, with resulting fuel element distortion and damage, is expected. Observed trends that may alleviate damage are the tendency of scale to flake off in high heat transfer areas, and the restraining effect of cladding on growth of fuel. No satisfactory means has been found to descale the fuel plates.

I. WATER CHEMISTRY

A. Introduction

The Experimental Boiling Water Reactor (EBWR) is a direct-cycle, light water-moderated and cooled boiling water reactor originally designed for 20-Mw thermal and 5-Mw electrical operation. The characteristics of this reactor have been so good that increase in power to 62 Mwt was possible. Recently a decision was made to modify the heat-dissipation equipment to make possible its operation at powers up to 100 Mwt.

Inasmuch as a greater amount of fuel element scaling has been observed in this reactor than in any other boiling water or pressurized water reactor, it became of interest to examine this scaling phenomenon to see if it might have any foreseeable undesirable consequences in 100-Mw operation. This led to a series of studies and experiments which were described in internal memoranda and reports. The purpose of the present report is to summarize the results of these investigations, as well as those of earlier studies concerned with associated aspects of water chemistry. The source of much of this material has been the EBWR Test Reports.⁽¹⁾

B. The Reactor System

The EBWR has been adequately described elsewhere.⁽²⁻⁴⁾ However, since the problems described in this report are basically due to corrosion products, a brief résumé of characteristics of possible significance from the corrosion standpoint is desirable.

Demineralized water boils in the core of the reactor, in contact with the fuel elements, to form steam which passes through a dryer to the turbine and on to a condenser. A steam air ejector removes noncondensable gases from the condenser. Condensed steam collected in the hotwell passes through full-flow cotton string filters back into the reactor.

Water from the reactor vessel, at velocities up to 10 gpm, is continually bypassed through a purification system. In the purification system reactor water is cooled, filtered through 2-micron particle size cotton string filters, passed through a mixed bed (Nalcite HCR and Nalcite SBR) ion exchanger, through a cotton string after-filter, regenerative cooler, and is pumped back into the reactor. A flow diagram of the reactor is shown in Fig. 1.

C. Materials

The pressure vessel is made from SA-212B carbon steel, with a Type 304 stainless steel 0.11-inch liner. Materials in the reactor vessel in contact with the primary water are:

- Type 304 stainless steel - core support, shroud, and shock shield;
- Type 304 stainless steel with 2% boron - control rods;
- Type 304 stainless steel with 1% boron - thermal shield;
- Zircaloy-2 - fuel cladding, and control rod followers;
- Hafnium - control rods;
- Aluminum-1% nickel alloy, M-388* - dummy fuel elements.

The material from the pressure vessel through the steam dryer is Type 304 stainless steel.

*Developed by ANL Metallurgy Division; available commercially as Alcoa X-8001.

Piping from dryer to turbine is carbon steel lined with 0.005-inch Kanigen nickel.* The turbine is made of conventional alloy materials; everything inside except the rotor, however, is coated with Kanigen nickel.

The condenser is of carbon steel with (6063-T831) aluminum tubes clad inside with 72S aluminum. The upper layer of tubes are protected from water erosion by stainless steel shields over the upper half.

Feedwater pumps are made of 4-6 and 12-14 chrome steels.

The shells of heat exchangers of the air ejector, through which all of the feedwater flows, are made of steel, lined with 0.003 in. of Kanigen nickel. Tubes in these exchangers are made from No. 16 BWG aluminum.

The feedwater filters, cotton string type with 2-micron particle size capacity, are supported in Type 304 stainless steel-lined steel vessels. The remainder of the feedwater system to the reactor vessel is stainless steel.

All of the metallic material in the purification system in contact with reactor water is stainless steel.

*Chemically deposited nickel containing 7% phosphorus; supplied by The General American Transportation Corporation, Chicago, Illinois.

The fluid recovery system contains an aluminum sump tank and an aluminum tube cooler. Reactor water leaking past the seals on the feed-water pump and the control rods is collected in this tank, along with condensed vapor leaking past other seals, and is returned to the main condenser.

D. Water

1. Design Conditions

The purification system, with its mixed-bed ion exchanger, was designed to maintain high-purity water in the reactor. The design figure was set at a specific resistance of 1 megohm-cm (equivalent to 0.5-1.0 ppm total solids). Early calculation indicated that this quality could be maintained by a 4-gpm flow through the purification system. The system was designed, however, for a 10-gpm flow in case it should be necessary to remove boric acid.

The use of aluminum in dummy fuel elements, condenser, and air ejector coolers precluded the extended use of water of high pH.

2. Prestartup Conditions

After placing the aluminum-nickel M-388 dummy fuel elements in the reactor vessel, but before loading fuel, hydrostatic testing of the pressure vessel gasket was carried out by using tap water in the system, because sufficient quantities of pure water were not available. The tap water remained in the system for about two days. On reopening the vessel, extensive rusting of the vessel cladding was observed as a loose, reddish oxide film. Rather extensive corrosion of the aluminum-nickel dummy fuel elements was also observed. This corrosion was characterized by some blistering and was aggravated by galvanic attack near rivets holding stainless steel and aluminum-nickel in contact. A spectrographic analysis of a sample of the corrosion product showed:

<u>Element</u>	<u>% (Order of magnitude)</u>
Al	gg 1*
Fe	0.05
Mg	0.02
Ni	0.03
Si	0.1

*gg = Much greater than,
or predominant.

3. Early Operating Conditions⁽⁵⁾

On December 11, 1956, a total of 11,250 gm of boric acid was added to the reactor water to obtain physics data, and on December 14 the reactor vessel was drained, flushed, and refilled with demineralized water. The reactor water was circulated through the purification system to remove the remainder of the boric acid, but the amount remaining in the water was not determined.

On December 29, 1956, a total of 19.8 lb of boric acid was added to simulate full-poisoned conditions. The pH decreased to 5.2 and resistivity dropped to 90,000 ohm-cm. It had been planned originally to operate the reactor at 20 Mw for two days, but boric acid remained in the system until January 10, 1957. During this time the cleanup loop was not operated, and water conditions were very bad, i.e., there existed water with pH 4.4, a specific resistance of 38,000 ohm-cm (26 micromhos/cm), and a high "crud" content. No determination was made of insoluble solids.

The boric acid was taken out by the mixed-bed ion exchanger a little at a time, starting January 8 and finishing January 11, when only 8.3 ppm remained of the original 748 ppm. Total operating time of the purification system was about 20 hr at 8.4 gpm. The efficiency of the ion-exchanger for the removal of the boric acid was about 98%.

The designed capacity of the water-cleanup system is much greater than needed for continuous 20-Mw operation, since the resistance of 38,000 ohm-cm was increased to one megohm-cm while operating at full power. No determination was made as to the minimum flow rate required to maintain water of 1 megohm resistance. On March 10, 1957, the filters and ion-exchange column were bypassed for 11 hr to increase the activity of the reactor water for DF (decontamination factor) data, and the specific conductivity increased from 0.9 to 1.2 micromhos.

The boric acid carryover in the steam was higher than had been anticipated. The DF for boric acid was only 18 as compared with the radioactivity DF of about 4.5×10^3 .

4. Normal Operating Conditions

In the absence of boric acid, the specific conductivity has been between 0.5 to 1.0 micromho/cm. The pH has varied from 6.9 to 7.2, a result which agrees very well with predicted conditions (pH = 7.0 and specific conductivity of 1.0 micromho).

a. Oxygen Content and Radiolytic Decomposition

Studies in EBWR indicated that normal steady-state oxygen levels at 600 psig and 20 Mw were as follows:

Feedwater	0-5 ppb
Reactor water	0.24 ppm
Steam	28 ppm

During the preliminary design stage it had been assumed, based on experience with pressurized water reactors, that there would be a negligible amount of radiolytic water dissociation (or water decomposition) and that as a result there would be very little oxygen in the water. This had been a large factor in justifying the use of conventional steam-plant materials. The BORAX experiments⁽⁶⁾ showed that boiling reactors have a relatively high oxygen content in the water due to their higher net rates of dissociation of water. The reason for this is, apparently, that the boiling process tends to strip gaseous decomposition products out of the reactor before they can recombine. This finding was the basis for the decision to line significant portions of the steam plant of EBWR with Kanigen nickel to protect it from the possible effect of the high oxygen content of the steam.

The first data on water decomposition in EBWR are given in Table 1.⁽⁵⁾ The results showed a rather wide spread from 45.7 cc to 66 cc/l and averaged 55.1 cc gas/l of condensed steam (uncorrected for H₂ and O₂ dissolved in the condensed steam). The average of the twelve analyses used gave 30.7% O₂, 66.4% H₂, and 2.9% inerts. This decomposition was about three-fourths that obtained from BORAX-III, namely, 77 cc gas/l analyzing 32% O₂, 60% H₂, and 8% inerts with water at pH 7.0. The pressure and temperature in EBWR were 600 psig and 488°F as compared to 300 psig and 420°F in BORAX.

The results obtained December 29, 1956, are of interest because they indicate that the presence of 0.748 gm boric acid/l of water (pH 5.2 and resistivity 90,000 ohm-cm) had no significant effect on water decomposition. The decomposition, 53 cc total gas/l of condensed steam at 20 Mwt, was very close to the 55.1-cc/l average.

The next group of experiments on water decomposition were concerned with effects of power level and pressure. The data were obtained by measuring the total gas liberated from a known mass of steam, and are summarized in Fig. 2 as liters of oxygen per minute at various power levels and pressures. BORAX data are included for comparison. It is seen that a plot of the decomposition rate as a function of reactor power indicates a linear relationship; further, at the same power level the higher pressure and temperature are associated with the lower net rate of

Table 1

EBWR WATER-DECOMPOSITION DATA

Date	Time	Power level, Mw	Reactor water quality		Gas yield, cc/liter ^a	Condensed-steam gas analysis			Air-ejector gas flow, cfm ^b	Air-ejector gas analysis			
			pH	Conductivity		O ₂ , %	H ₂ , %	Inert gas, %		O ₂ , %	H ₂ , %	Inert gas, %	
				Determined, ohms/cc									Recorded, μ mhos
12-23-56 ^c	0115	20	7.0	180,000	50	22.7	47.0	30.3 ^d		25.2	32.8	42.0	
12-23-56 ^e	2015	20	7.0	180,000	69	20.4	44.5	35.1 ^d		22.6	27.2(?)	50.2	
12-29-56 ^f	2210	20	5.2	90,000	53	30.2	55.9	13.9		29.4	56.1	16.1	
1-11-57 ^g	1630	20	5.0(?)	410,000	47.5	29.7	68.6	1.7					
1-30-57	1715	20	7.05	900,000	66	31.0	63.4	5.6		30.4	53.9	15.7	
2-11-57	1917	20	7.0	760,000	52.5	30.0	66.2	3.8		28.2	47.3	24.5	
2-11-57	2330	20	7.0	600,000	46	29.2	62.4	8.4		28.9	44.6	26.5	
2-12-57	1300	20	7.15	640,000	45.7	29.3	69.2	1.5					
2-12-57	2020	20	7.2		0.8	47.8	30.2	67.3	2.5	29.5	48.0	22.5	
2-13-57	1030	20	7.1		0.9	58.5	31.6	66.6	1.8				
2-13-57	1920	20	7.08		0.9	55	30.0	66.0	4.0	29.9	47.5	22.6	
2-14-57	1200	20	7.1		1.0	57.8	31.1	66.5	1.8	29.9	49.7	20.4	
2-14-57	2110	20	7.02		1.0	63.3	32.5	67.5	0.0	1.2	29.7	51.8	18.5
2-18-57	1115	20	7.3		0.8	59.8	31.7	67.0	<1.3	1.0	30.5	56.0	13.5
2-20-57	1000	20	6.9		0.9	58.0	31.7	66.2	2.0	0.92	30.9	56.1	13.0
Average of samples of 1-11-57 through 2-20-57						55.1	30.7	66.4	2.9		29.7	50.5	19.7

^aNot corrected for gases dissolved in the condensed steam.

^bNot corrected for gas density; values reported are for air at 80°F in cubic feet per minute.

^cThis sample was steam bypassed to the condenser.

^dValues are high owing to transfer of sample or actual nitrogen in system.

^eThis sample was steam going to the turbine.

^fBoric acid in the reactor water was 0.748 g/liter for this sample.

^gOnly 8.3 ppm boric acid was in the reactor water for this sample.

water decomposition. The rate of oxygen production at 600 psig is 0.4 l/min/Mwt and at 300 psig is 0.6 l/min/Mwt. Assuming that 8 Mev of the total energy per fission is absorbed in water dissociation, the G_{H_2} (moles of hydrogen produced per 100 ev) values are 0.15 and 0.21, respectively.

FIG. 2
REACTOR POWER VS. DECOMPOSITION RATE

In subsequent EBWR water-decomposition experiments the effect of gaseous and chemical additions to the reactor feedwater have been studied.

In BORAX-III⁽⁶⁾ the addition of gaseous hydrogen to the reactor water was shown to have a marked effect in reducing net dissociation. The magnitude of the effect depended on the point of injection, i.e., when hydrogen was fed through the feedwater sparger, the reduction in net decomposition was greater than when it was injected through a drain line in the bottom of the reactor. This effect is believed to have been due to more complete solution of the hydrogen, better mixing, and possibly longer exposure time to flux effecting recombination, when the hydrogen was injected with the feedwater. Reduction in net decomposition by hydrogen addition has been observed also in BORAX-IV.⁽⁷⁾

In the EBWR experiments,^(8,9) hydrogen gas was metered into the reactor feedwater side stream from the purification system. After an equilibrating period of 20 min, samples of steam, condensate,

and feedwater were analyzed. Gas samples were analyzed for oxygen, hydrogen, and inerts, and the liquid samples for oxygen. The results are graphically shown in Fig. 3, which shows the steady-state oxygen concentrations in cc/l of condensed steam at 10 Mw and 20 Mw as a function of hydrogen additions. Data from BORAX-IV and ALPR are shown for purposes of comparison. Two conclusions reached from these experiments were:

- (1) The BORAX reactor requires much more added hydrogen to give a specific reduction in net decomposition than does EBWR or ALPR.
- (2) Although small amounts of hydrogen cause a sharp decrease in water decomposition, the effectiveness of additional amounts diminishes rapidly, and the data imply that it may be impractical to attempt reducing the oxygen content of the steam below $1/2$ -1 ppm by this method.

It should be kept in mind that, since rate of water decomposition is proportional to power, and since rate of steam generation is proportional to power, the concentration of decomposition gases in the steam is independent of power. Since in this plot the effect of power is minimized, the marked differences in slope between the BORAX-IV curve and the curves for EBWR and ALPR indicate significant differences between these reactors which

are not directly associated with power. The nature of these differences is unknown, but, in view of the effect of changing the point of hydrogen injection found in BORAX-III, it is believed that differences in flow through the core are involved.

The effect of added hydrogen in reducing radiolytic decomposition and lowering the oxygen content of the steam, and the reported breakdown of hydrazine (N_2H_4) under radiation to evolve hydrogen suggested the use of hydrazine as a more convenient source of hydrogen for use with a boiling reactor than bottled gas. As a 35% water solution, it could be metered and injected easily by a positive displacement feed pump.

The addition of about 3 ppm of hydrazine resulted in a sharp rise in conductivity of the water in the reactor and hotwell, and in activities measured at several points in the system.

The rate of water decomposition decreased slightly and then increased to a value considerably above the initial rate. The pH of the hotwell increased slightly while that of the reactor decreased to a value of about 5.2. No hydrazine was detected in the reactor, and nitrate concentration increased from less than 0.1 to 0.8 ppm.

The addition of 0.5 ppm ammonia produced similar results. The addition of nitrogen gas produced no noticeable changes in decomposition rate, reactor water conditions, or plant activities. It was concluded that hydrazine, ammonia, and nitrogen offered no promise in reducing water decomposition.

b. Solids Content

The first investigation of solids in the reactor water system was made on samples taken May 28, 1957. This was concerned with insoluble solids collected by passing known water volumes through Millipore filters. The pore size of the filters is not recorded. Similar surveys were made on June 18, June 21, and November 12, 1957. Only on the June 21 survey was a record made of the pore size of the filter, namely, 10 microns. The results of the four surveys have been recalculated to the same basis and are given in Table 2. Values for B, Ba, Cr, Mg, Zn, and Zr were given in most instances, but the amounts were so low that they have been omitted from the table.

Some conclusions that appear to be indicated by the results obtained are:

- (1) Very little chromium was found in the insoluble particulate matter, as compared to aluminum, iron, and nickel. This suggests that most of the chromium corrosion product must be present in a soluble form.
- (2) Values obtained for aluminum, iron, and nickel on June 21 samples were consistently lower than those on other sampling dates. Since this was the only group where the pore size of the filter was recorded, there is a possibility that a filter of smaller pore size was used in the other surveys.
- (3) Neglecting the June 26 results, the amount of iron in the water was about the same in May and June as in November. During this same interval the amounts of aluminum and nickel increased.
- (4) Significant amounts of crud pass through the prefilter and to a lesser extent the ion exchanger.

Table 2

SPECTROGRAPHIC ANALYSES OF INSOLUBLES CAUGHT ON MILLIPORE FILTERS
MICROGRAMS PER LITER (ppb) - FACTOR OF 2 ACCURACY

Element	Month and Day 1957	Reactor Water			Feedwater		Steam	
		S-5 ^a	After Prefilter S-6	After Ion Exchanger S-7	Before Filter S-3	After Filter S-4	Before Dryer S-1	After Dryer S-2
Al	5/28	19	8	3.2	3.6	-	3	1.8
	6/18	20	16	7.1	1.2	0.3	0.2	1.6
	6/21	6.5	-	-	-	-	0.7	1.5
	11/12	55	-	-	T ^b	0.5	-	0.5
Ca	5/28	0.8	0.4	0.9	1.2	-	2.5	2.3
	6/18	0.6	0.3	0.4	0.4	0.1	0.1	0.5
	6/21	0.2	-	-	-	-	0.3	0.6
	11/12	4.0	-	-	1.8	0.2	-	5.0
Cu	5/28	7.7	1.1	1.9	1.8	-	4.0	1.2
	6/18	1.2	1.1	0.8	0.4	0.1	T	0.8
	6/21	-	-	-	-	-	0.5	0.5
	11/12	7.5	-	-	5.0	T	-	0.5
Fe	5/28	77	42	16	30	-	12	14
	6/18	60	55	20	61	0.9	1.5	9.8
	6/21	1.6	-	-	-	-	2.9	6.0
	11/12	88	-	-	80	20	-	20
Mn	5/28	2.3	0.8	0.3	0.1	-	0.2	0.5
	6/18	0.3	0.3	0.1	0.1	T	T	0.6
	6/21	T	-	-	-	-	T	0.1
	11/12	2.5	-	-	0.7	0.2	-	0.5
Na	5/28	6.2	2.2	1.3	9.0	-	10	18
	6/18	0.1	0.6	0.2	0.1	0.1	T	0.2
	6/21	0.2	-	-	-	-	0.1	0.6
	11/12	-	-	-	-	-	-	-
Ni	5/28	25	14	6.5	1.2	-	1	0.1
	6/18	25	35	10	0.1	0.1	1.6	0.2
	2/21	1	-	-	-	-	0.4	0.5
	11/12	88	-	-	25	3.0	-	2.5
Pb	5/28	1.2	0.4	0.2	<0.3	-	0.4	2.3
	6/18	0.4	0.1	0.3	0.2	0.1	T	9.8
	6/21	0.2	-	-	-	-	0.1	4.5
	11/12	-	-	-	-	-	-	-

^aSampling point on Fig. 1^bTrace (<0.1 μ gm/liter)

- (5) A significant decontamination factor exists between reactor water and steam for particulate corrosion products.
- (6) The steam dryer alone does not seem to exert much of a decontaminating effect on suspended particulate matter.
- (7) The amount of suspended aluminum in the feedwater is less than that in the reactor water. None is removed by the feedwater filter, indicating that if aluminum comes from the condenser it is in a dissolved form.
- (8) The amount of insoluble nickel in the feedwater is less than that in the reactor water. Some is removed by the feedwater filter.
- (9) The amount of suspended iron in the feedwater is nearly as great as in the reactor water. Significant amounts are removed by the feedwater filter.

In June, 1957, water samples were taken from various parts of the system for investigation of total solids as contrasted to suspended or particulate solids. Due, it is believed, to the small size of water samples, and other problems in technique, the results were inconclusive.

In 1959 a special device was developed for determining the amount of total solids (suspended and dissolved) in reactor water. The basic design allows water from any place in the primary system to be pumped into a weighed crucible in an autoclave which is operating at 100°F above the saturation temperature. The steam is condensed, expanded to atmospheric pressure, and the total water flow measured. By using various safeties in the system, the experiment may be operated continuously with only occasional maintenance. A schematic outline of the device is given in Fig. 4.

After installation of the equipment, one run was made on reactor water before final shutdown for 100-Mw conversion. Evaporation of 107 liters of reactor water resulted in the collection of 7.7 mg, or a concentration of 0.07 ppm. Examination of the activity present by means of a 200-channel gamma-ray analyzer indicated Co^{58} and Co^{60} present in a ratio of about 10 to 1.

A wet chemical analysis for aluminum, iron, and nickel gave the results summarized in Table 3 calculated on basis of assumed composition.

FIG. 4
SYSTEM FOR MEASUREMENT OF SUSPENDED AND DISSOLVED SOLIDS

Table 3

WET CHEMICAL ANALYSIS OF REACTOR WATER SOLIDS

Element	Assumed Compound	Amount, mg		%	
		Element	Compound	Element	Compound
Al	$\text{Al}_2\text{O}_3 \cdot \text{H}_2\text{O}$	1.59	3.53	59	70
Fe	Fe_2O_3	0.68	0.97	25	19
Ni	NiO	0.43	0.54	16	11
		2.70	5.04	100	100

c. Corrosiveness

Although operating conditions in EBWR have varied widely (Figs. 5, 6 and 7), the corrosion behaviors of miscellaneous structural materials are of interest. To provide information of this type, testing of specimens has been initiated in both reactor water⁽¹¹⁾ and steam.⁽¹²⁾

(1) Reactor Water

One group of samples has been removed from the reactor, but examination has not been completed, so no results are available at this time. A list of the samples, and a description of the experiment are given in Reference 11.

FIG. 6
EBWR OPERATIONAL RECORD - 1958

FIG. 7

EBWR OPERATIONAL RECORD - 1959

(2) Steam

In November, 1957, samples of six types of structural materials used in the EBWR turbine were placed in a sample holder and then in a tee in the 6-inch steam line between the reactor and the steam dryer. Representative samples were withdrawn three times during the next nineteen months. The corrosion resistances of all six materials were good until the last month of operation prior to shutdown for 100-Mw conversion. During this last month, pitting and accelerated corrosion of the low alloy samples occurred. This accelerated attack may have resulted from two known operational factors: the addition of oxygen to the reactor water for a short time, and submersion of the steam line samples in water for five weeks before reactor startup for the last month of operation.

Alloy designation and description of the materials are given in Table 4. After each of the three periods, given in Table 5, representative specimens were removed for examination.

The appearance of the samples after the first period was satisfactory; the corrosion rates over the first two periods were good (see Fig. 8). However, during the final month, the low alloy samples exhibited accelerated corrosion and pitting attack. The new samples placed in the holder at the start of the last month were even worse than those which had been in the system since the start of the test, indicating that the oxide on the old samples afforded some protection against attack.

Observations on the corrosion behavior of Kanigen nickel are described in Section II-F.

d. Activities

The sources of radioactive nuclides in a water reactor fall into three main groups: water, corrosion products, and fission products.

(1) Water

The primary activity originating from water itself is N^{16} , which is produced by a fast n,p reaction of O^{16} . It decays with emission of a β^- and gamma rays of 6.13 and 7.13-Mev energy, with a half-life of 7.35 sec. This activity is responsible for most of the radiation hazards about the external parts of the system during operation. Activity levels at various components of the EBWR system attributed to N^{16} at different reactor powers are listed in Table 6.⁽⁴⁾ Due to its short half-life, this activity disappears very quickly after reactor shutdown.

Table 4

STEAM CORROSION SPECIMENS

No.	Designation	Material	Turbine Part
1	ASTM A-113 stainless steel clad	Structural steel	Diaphragm web
2	AISI 4140	Steel	Wheels
3	Type 403 stainless steel	13% chromium-stainless steel	Buckets
4		1% chromium, 1/2% molybdenum steel	Shaft
5		Low carbon steel	Diaphragm band
6		13% chromium-stainless steel, non-hardening	Nozzle element

Table 5

STEAM CORROSION SAMPLE RESULTS

Period	Inclusive Dates	Reactor Operation Time				Alloy and Sample No.	Metal Loss (mg/cm ²)
		For the Period		Total			
		hr	Mw-hr	hr	Mw-hr		
1	11/2/57-4/6/58	2,249	41,861	2,249	41,861	1C	2.05
						2A	1.14
						3A	0.22
						4D	1.24
						5A	2.76
						6A	0.31
2	4/11/58-11/8/58	3,522	67,272	5,771	109,133	1A	2.98
						2C	1.47
						3C	0.19
						4A	1.33
						5C	3.70
						6C	0.18
3	4/29/59-7/3/59	798	16,178	6,569	125,311	1D	4.19
						2D	3.58
						3D	0.46
						4C	16.90
						5D	3.23
						6D	0.25
		798	16,178	798	16,178	1E*	11.68
						2E	11.97
						3E	0.23
						4H	9.70
						5G	15.58

*These five samples placed in holder at start of third period.

FIG. 8
SUMMARY OF EBWR STEAM LINE CORROSION RATES
(1 mil $\approx$ 20 mg/cm²)

Table 6

EQUIPMENT SURFACE ACTIVITIES ATTRIBUTED PRIMARILY TO N¹⁶
(in mr/hr)

Location	Reactor Power			
	10 Mwt	20 Mwt	40 Mwt	61.7 Mwt
Air Ejector after Cooler	90	400	4000	7000
Condenser Hot Well	10	40	240	580
Steam Dryer	80	150	600	720
Turbine Exhaust Casing	2	4	14	20
Feedwater Filters	8	10	50	100
Plant Air Exhaust	6	12	40	120

Experimental work on the distribution of N¹⁶ activity has been carried out.⁽¹³⁾ Very briefly, the results may be summarized as follows:

BORAX-IV studies showed that only a small fraction of the N¹⁶ produced was released into the steam and that certain chemical additives affected its distribution.

At EBWR, measurements were made of the absolute concentration in the water and steam within the reactor vessel. The concentration of N^{16} at different locations was obtained from measured gamma-ray activities of water or steam samples continuously removed from the reactor vessel. Information about the amount of anion, cation, and neutral species in which N^{16} is associated was obtained by passing the sample through ion-exchangers before exposure to the counter.

Since it is produced in the core by a nuclear reaction, N^{16} appears as a recoil atom with various degrees of ionization. As a result, it combines with oxygen and hydrogen in the coolant to form ions or compounds of the following types:

Anion Species	NO_2^- , NO_3^-
Cation Species	NH_4^+
Neutral Species	NO , NO_2

After its production, N^{16} combines mainly in an anion form which remains in the water; only cation and neutral species were detected in the steam. Due to limitations of sensitivity of the method, no anion activity could be detected in the steam, and no neutral activity could be detected in the water.

Under normal operating conditions, N^{16} is present in the water as anion and cation in the ratio of about 7 to 1 (Fig. 9). In the steam, the ratio of neutral to cation species decreases with increasing reactor power (Fig. 10).

FIG. 9
PERCENTAGE OF CATION AND ANION SPECIES IN
REACTOR WATER 52 in. ABOVE FUEL.

FIG. 10
PERCENTAGE OF NEUTRAL AND CATION
SPECIES OF N^{16} IN STEAM 92 in.
ABOVE FUEL.

FIG. 11
EFFECT OF H_2 ADDITION ON
 N^{16} ACTIVITY IN STEAM
112 in. ABOVE FUEL.

The effect of adding various chemicals to the reactor feedwater was studied. Nitrogen, oxygen, and argon gas had no effect. Hydrogen, hydrazine, and ammonia decreased the ratio of concentrations of anion to cation species in the water. In each case the activities in steam and plant increased above normal value. The increase in steam activity with hydrogen addition at different power levels is shown in Fig. 11.

At 20 Mw, 600 psig, and 489°F reactor operation, the addition of 40 cc H_2/l of feedwater indicated a shift to nearly equal proportions of anion and cation species in the water. At the same time, the steam activity, as well as the steam and condensate system radiation monitor indications,

increased by a factor of four. No change was observed in the total water activity since the amount of N^{16} in the steam was still a small fraction of the amount produced.

Experimental determinations of N^{16} concentrations were made at specific points in the reactor water above the core and in the steam in the upper portion of the reactor vessel. The results

FIG. 12
 N^{16} CONCENTRATION IN REACTOR
WATER 52 in. ABOVE FUEL.

indicate that in the water (Fig. 12) the concentration was proportional to reactor power, while in the steam the concentration appeared to be proportional to the square of the power (Fig. 13).

The ratio of N^{16} in a gram of steam to N^{16} in a gram of water has been found to vary; it is increased by addition of certain chemicals and by an increase in power. Results at BORAX-III⁽⁶⁾ indicated that this ratio was increased by a drop in pH.

It was concluded that the decontamination factor for N^{16} in passing from water to steam is a function of the concentration of hydrogen gas and also a function of the void fraction in the reactor water. Because of the combined effect of the high recirculation ratio and the smaller-than-unity ratio of steam

to water N^{16} concentration, the amount of N^{16} which leaves the reactor vessel with steam at 20 Mw is less than 1% of the total produced. The addition of hydrogen gas to the reactor feedwater can result in the N^{16} in the steam being equal to or greater than that in the water.

FIG. 13
 N^{16} CONCENTRATION IN STEAM 92 in. ABOVE FUEL
AT VARIOUS OPERATING PRESSURES.

Another activity formed in EBWR that originates from water itself, but is of less importance than N^{16} , is F^{18} . Its occurrence, characteristics, and mode of formation are listed in Table 7.

Table 7

CORROSION PRODUCT ACTIVITIES AT EBWR

Isotope	Half-life	γ rays (Mev)	Probable Formation	Concentration ^a $\times 10^{-2} \mu c/cm^3$
Reactor Water:				
Mn^{56}	2.6 hr	0.850 1.82 2.15	Fe^{56} (n,p) Mn^{55} (n, γ)	4.5
Na^{24}	14 hr	1.36 2.75	Al^{27} (n, α) Na^{23} (n, γ)	1.5
Co^{58}	72 d	0.81	Ni^{58} (n,p)	0.088
Co^{60}	5.2 y	1.17 1.33	Co^{59} (n, γ) Ni^{60} (n,p)	trace
$F^{18}{}^b$	1.9 hr	0.51 ^c	O^{18} (p,n)	1.3
Condenser Tubing Deposit:				
Co^{58}	72 d	0.81	Ni^{58} (n,p)	8.8
Reactor Steam: ^d				
F^{18}	1.9 hr	0.51 ^c	O^{18} (p,n)	0.3
Co^{58}	72 d	0.81	Ni^{58} (n,p)	trace
Anion Resin:				
Cr^{51}	26 d	0.32	Cr^{50} (n, γ)	-

^aAll concentrations determined for 20 Mw over periods of 24 hr or longer.

^bComes from O^{18} in water; not a corrosion product.

^cAnnihilation quanta.

^dBefore steam dryer. Concentration per cm^3 condensed steam.

(2) Fission and Corrosion Product Activities

Soon after the original startup of the reactor,⁽⁵⁾ background higher than normal was noticed by several low-counting laboratories on the site. Samples of gas were taken from the air ejector discharge and reactor vent system and counted. Cs^{138} was identified immediately and the fission gases, Xe^{138} , Xe^{135} , Kr^{89} , Kr^{90} , and Kr^{91} , were identified later. Examination of reactor water samples on January 10, 1957, for Cs^{138} , Sr^{89} , Ba^{139} , and Mo^{99} gave positive results for all. From the analytical results on four samples it appeared that the production had reached equilibrium conditions. On the basis of the data obtained, it was concluded that the amount of fuel (uranium) in the reactor water was 59 mg.

The source of the uranium was concluded to be surface contamination of fuel elements which resulted from rupture of about three plates during autoclave testing previous to the loading of EBWR. Although the autoclaves were decontaminated, subsequent plates could have been contaminated, and it is not certain how well the plates which had been in the autoclave at the time of the failures were decontaminated.

Another possible source is the reported presence of about 1 ppm uranium in the Zircaloy-2 used for fuel cladding.

The results of a survey of the activities in and around EBWR, made early in 1957,⁽¹⁴⁾ are given in Tables 7 and 8. The relative importance of the activated corrosion products should be noted.

Table 8

FISSION PRODUCT ACTIVITIES AT EBWR

Isotope	Half-life	γ rays (Mev)	Decay Chain	Concentration ($\mu\text{c}/\text{cm}^3$)
Air-ejector gas: Cs^{138}	33 m	0.46 0.98 1.44	$\text{Xe}^{138} \rightarrow \text{Cs}^{138}$	7.4×10^{-3}
Xe^{135}	9.2 h	0.250	$\text{Xe}^{135\text{m}} \rightarrow \text{Xe}^{135}$	3.2×10^{-4}
$\text{Kr}^{85\text{m}}$	4.5 h	0.150	U fission	1.2×10^{-4}
Air-ejector filter: Ba^{139}	85 m	0.163	$\text{Xe}^{139} \rightarrow \text{Cs}^{139} \rightarrow \text{Ba}^{139}$	Trace
Ion-exchange filter: Ba^{140}	13 d	1.60	$\text{Xe}^{140} \rightarrow \text{Cs}^{140} \rightarrow \text{Ba}^{140} \rightarrow \text{La}^{140}$	Trace
Room air: Cs^{138}	33 m			1.2×10^{-7}

In a sample of water taken July 11, 1958,⁽¹⁵⁾ about a year later, over 90% of the gamma-emitting nuclides present at the time the sample was taken could be accounted for as indicated in Table 9.

Table 9

ACTIVITIES IN REACTOR WATER

July 11, 1958

Nuclide	Half-life	Approx. % of Total
Na ²⁴	15 hr	75
Mn ⁵⁶	2.6 hr	20
Co ⁵⁸	72 d	2
Cu ⁶⁴	12.6 hr	3

A comparison of the specific activities of these nuclides with those found in a different survey is shown in Table 10.

Table 10

REACTOR WATER ACTIVITIES COMPARED

Nuclide	Activity, d/(min)(ml)	
	July 11, 1958	April-May, 1957
Na ²⁴	1.6×10^5	3.3×10^4
Mn ⁵⁶	6×10^4	1×10^5
Co ⁵⁸	4×10^3	2×10^3

It thus appears that during the year the Na²⁴ content of the reactor water increased by a factor of 5 and the Co⁵⁸ increased by a factor of 2.

It should be noted that the actual concentrations of these impurities were negligibly small and would have been hard to detect by ordinary chemical means.

(3) Corrosion Product Activity Used as Tracer

The corrosion product activity in the water, particularly Na²⁴, has been made use of in investigation of moisture and radioactive carryover in the steam, and the decontamination factor.⁽¹⁶⁾ Additional sodium was added in tests in EBWR,⁽⁴⁾ the results of which are summarized in Table 11.

Table 11

STEAM QUALITY AND CARRYOVER

Thermal Power (Mwt)	Wt % Moisture, Steam		Decontamination Factor,* Steam		
	Before Dryer	After Dryer	Before Dryer	After Dryer	Across Dryer
20	0.062	0.032	1600	3110	1.94
30	0.133	0.06	754	1660	2.2
40	23.2**	0.6	4.3	167	38.8

* Sodium Activity in Reactor Water
Sodium Activity in Steam at Sample Point

**The high moisture content at 40 Mwt is attributed to slugs of water leaving the reactor. This occurred because of high reactor water level during the test.

(4) Fission Product Release from Defective Fuel Elements

Investigations were made in EBWR with deliberately defected experimental fuel elements. The elements were composed of $\text{UO}_2\text{-ThO}_2$ pellets in stainless steel tubes. The main objectives of the investigations were to determine the release rates of fission gases from this fuel, and to learn something about the operational problems that might result in event of fuel cladding failure in a boiling reactor.

The work has been described in detail⁽¹⁵⁾ and so will be summarized only very briefly here.

The fuel was 90 wt % ThO_2 -10 wt % U^{235}O_2 of high density (9.0 and 9.5 gm/cc). The deliberate defect in each test was a single drilled hole through the tube wall. In Test No. 1, a 0.010-inch diameter hole was centrally located between the top and bottom of the element; in Test No. 2 a 0.020-inch diameter hole was located at the top of the element.

In both tests the maximum thermal neutron flux on the fuel was about $8 \times 10^{12} \text{ n}/(\text{cm}^2)(\text{sec}^1)$ and the maximum surface heat rates were about 160,000 Btu/(hr¹)(ft²).

The results indicated that release of the fission gases was controlled by the rate of diffusion through the oxide.

An interesting observation was that when the control rod nearest the defected element was moved so as to increase the neutron flux in that element, a burst of activity was shown by a peak in the counting rate of the gamma-ray monitor at the air ejectors.

Radiation-monitoring instruments in the steam lines or in the air ejector gases, other than that noted above, did not show any significant increase in counting rate during either test. This was due to the fission product activities being entirely overshadowed by the N^{16} activity and the activity resulting from activation of structural material corrosion products. The only way in which consistent differences between defect and non-defect tests could be detected was by analysis of air ejector gases.

E. Conclusions

The use of conventional materials, except for Kanigen nickel plate, in the steam plant of EBWR has resulted in no significant corrosion problems in spite of (or perhaps because of) the high oxygen content of the steam resulting from radiolytic water decomposition. It is questionable if the Kanigen plate has been of value.

High water quality has been maintained, as indicated by resistivity and pH values, except when boric acid was in the system. No deleterious effects on water decomposition or corrosion have been noted as a result of the presence of boric acid.

Although the addition of hydrogen to the reactor feedwater markedly reduced the net water decomposition, the process is not economically attractive for EBWR because of loss of gas through the air ejector. If the presence of oxygen in the steam proves to be beneficial in controlling corrosion in the steam plant,⁽¹⁷⁾ its reduction by added hydrogen or other means such as catalytic recombination may not be desirable. This subject requires further investigation.

Limited corrosion studies with sample coupons in the steam line suggest that plant operation procedures, such as opening of system to air during shutdown and filling of system with air-saturated water, may influence plant corrosion.

The major activity arising from activation of the water is N^{16} . Certain chemicals, such as hydrogen gas, ammonia, and hydrazine added to the water have a slight influence on its partition between water and steam.

No fuel element cladding failures have occurred in EBWR up to the time it was shut down for 100-Mw conversion. Such fission products as have been found were attributed to tramp fuel contamination on the surface

of the elements, the reported presence of about 1.5 ppm uranium present as an impurity in Zircaloy-2, or to deliberately defected oxide fuel elements. Major activities from the latter were fission product gases and their daughters, whose release to the water appeared to be controlled by rate of diffusion through the oxide.

The dissolved and suspended solids in the reactor were very low in quantity, in the parts per billion range, and were composed largely of iron, nickel, and aluminum, presumably as oxides. The major short-lived corrosion product activities were Mn^{56} and Na^{24} . The major long-lived corrosion product activities were Co^{58} and Co^{60} .

II. NATURE AND DISTRIBUTION OF DEPOSITS

A. Purification System

During the boric acid run, December 29, 1956 to January 11, 1957, the amount of insoluble "crud" in the reactor was high and most of the activity associated with it was deposited on the No. 1 prefilter. During this run the after-filter became radioactive also (2.0 r/hr at 4 in.). It is believed that this may have been due to colloidal particles which had passed through the prefilter and either had time, or in some way were caused, to agglomerate by going through the ion-exchange column and were then picked up on the after-filter. Cartridges from the pre-filter were a dark red-brown color from deposited material, while those from the after-filter were a light brown color.⁽⁵⁾

A radiochemical analysis was made on a sample taken in February, 1957, from the prefilter. Results showed:

1. Major activities were Fe^{59} and Co^{60} .
2. Minor activities in decreasing amounts were Ni^{59} , Cr^{51} , Zr^{95} , Ce^{144} , and Zn^{65} .
3. Trace activities, identification uncertain, were Sr^{89} , Ba^{140} , Mo^{99} , Cs^{138} , and possibly Nb^{95} .

In another examination of the prefilter, particularly for shorter-lived activity, an activity half-life of about 3 hr was found. This corresponds to the 2.6-hr half-life of Mn^{56} and checks with the fact that a primary activity in the reactor water was found to be Mn^{56} .

After cleaning up the boric acid and associated activities on January 11, 1957, the activity of the ion-exchange column was checked at three ports in this shielded vessel. The activities were 1.9 r/hr at the top, 200 mr/hr at the center, and 100 mr/hr at the bottom port.

Later, another ion-exchange bed, which was removed after a 14-day decay period, exhibited the following dose rates after use:

Top of bed	1 r/hr at 6 in., hard
1 ft down	500 mr/hr at 2 in.
2 ft down	100 mr/hr at 2 in.

The primary radionuclides present on resins and filters were Co^{58} , Ba^{140} , La^{140} , Fe^{59} , and Co^{60} . Chromium-51 was found only on the anion resin, presumably as chromate ion. A high percentage of the particles of deposited material were ≤ 0.1 micron in diameter when examined by electron microscope.

During the $2\frac{1}{2}$ -year operating life of the reactor prior to shut-down for 100 Mw conversion, ion-exchanger No. 1 was charged three times, No. 2 once. The last replacement for No. 1 occurred June 3, 1958, after one year's use in treating 1,064,600 gal of water. After 33 days decay the activity levels measured with a survey meter are given in Table 12.

Table 12

ACTIVITY LEVELS OF ION-EXCHANGE BED AFTER 33 DAYS DECAY

Position	Activity Level (mr/hr at 2 in.)
Directly above resin	22,000
Side of resin basket at top	17,000
One-fourth distance from top	7,000
One-half distance from top	500
One-fourth distance from bottom	170
Bottom	100

B. Feedwater Filters

In February, 1957, No. 2 feedwater filter was changed for a second time, about 230 hr of operation. The activity level of the unloaded assembly of cartridges measured about 1.5 mr/hr. The maximum activity observed on the surface of these unshielded vessels was 15 mr/hr at 2 in. at 20 Mw.(5)

A sample of about 1 sq in. from the surface of one of the cotton string cartridges was spectrographically analyzed. After subtracting blanks on the filter itself, the results indicated the deposit to be 67% iron, 30% nickel, and 3% balance of (Mn, Cr, Mo) based on an accuracy within a factor of 2.

Filter cartridges were replaced four times in 1957 when pressure drop across the filter reached 30 psig. Subsequent experience has indicated that the frequency of replacing the cotton fiber cartridges depends on the number of times the vacuum is broken in the main condenser. Release of vacuum increases O₂ content of the primary system by addition of air and accelerates corrosion. During startup, corrosion products are picked up and tend to clog the filters. On a rough comparative basis, it has been observed that one shutdown is equivalent to about 600 hr of continuous operation at 20 Mw. This indicates reduced maintenance will result from continued, rather than intermittent, power operation.

C. Steam Dryer

During the June-July shutdown in 1957, after 1370 Mw-days of heat power production, the steam dryer was opened for inspection.⁽¹⁸⁾ The radiation level in the dryer was between 5 and 6 mr/hr, and was primarily gamma activity. The walls of the vessel were, in general, free of deposit. Small areas had a brownish appearance, but no deposit sample could be obtained.

The emergency cooling coil in the lower half of the vessel did have some loosely adherent deposit, dark red-brown in color. This deposit, in general, was located on the upper surface of the tubes and a sample from the second row of tubes from the top was obtained for spectrographic and radiochemical analysis.

The results of the spectrographic analysis, accuracy within a factor of 2 are given in Table 13. The relative aluminum, chromium, iron, and nickel contents should be noted for comparison with other water deposit analyses.

Table 13

SPECTROGRAPHIC ANALYSIS OF DEPOSIT FROM STEAM DRYER

Element	%
Al	3
Cr	1.5
Cu	0.1
Fe	Very large
Mn	0.6
Na	0.2
Ni	1
Pb	0.1

The results of a radiochemical analysis (beta activity) are given in Table 14. Later examinations showed that the activity level increased to about 20 mr/hr inside the dryer (emergency cooling coil) and remained at about that level.

Table 14

RADIOCHEMICAL ANALYSIS OF DEPOSIT
FROM STEAM DRYER

Element (Isotope)	Activity, % Total
Co ⁵⁸ , Co ⁶⁰	58
Fe ⁵⁹	5
Ni ⁶³	0.5
Cr ⁵¹	0.5
Ce ¹⁴⁴ and Pr	6
U	29 ± 50%, ppm
Gross β activity:	1.9 x 10 ⁵ d/m/100 mg

D. Startup Heater

In June, 1958, during a preliminary hydrostatic test of the reactor vessel and its components, the startup heater leaked between gasketed flanges at a pressure of 600 psig.⁽¹⁹⁾ On removing the head, radiation levels as high as 2 r/hr were observed.

A thick, hard, caked deposit of corrosion products was found between the flanges above the lower part of the gasket. It appeared likely that with the large number of thermal cycles encountered by the startup heater, it expanded and contracted sufficiently to pack the accumulation and to finally separate the flanges to cause the leak.

Spectrochemical analysis gave the more important constituents as:

Al	0.3%	Fe	20%
Ba	4%	Mn	0.1%
Cr	0.8%	Ni	0.2%

The relative percentages of aluminum, chromium, iron, and nickel are of particular interest in comparing with other water deposit analyses.

E. Condenser

During the June-July, 1957, shutdown, the condenser was examined.⁽¹⁸⁾ Since the examination was a few weeks after reactor shutdown, the activity observed was due to longer-lived radioisotopes. The hotwell

area, lower section, showed 1-2 mr/hr, while the upper portion, above the condenser tube banks, showed only 0.1-0.2 mr/hr. This difference was expected due to the more active washing action of the steam and condensate in the upper areas.

A radiation survey with a Juno meter showed that the major activity was gamma. Preliminary gamma-ray analyses of two samples of deposit showed the only measurable activity present 56 days after shutdown was Co^{58} . Quantitative figures obtained were 0.088 and 0.054×10^{-6} curie/gm of deposit ($\pm 20\%$). At the time of shutdown this would have been 0.15 and 0.09×10^{-6} curie/gm, respectively.

Condenser tube deposits in the lowest section of the tube bank (just above the hotwell) varied in color from dark reddish-brown to black. The layer was very thin and adherent, and appeared to be a discoloration film rather than a deposit. Scraping with a knife gave no sample.

An appreciable amount of deposit was present on the stainless steel shields on the top row of condenser tubes. Designating the two center sections of the condenser directly below the steam inlet as B and C, and the two end sections as A and E, the deposits (upper row of tubes in each case) were light brown in color and dull in appearance (powdery type) in sections A and D. They were not adherent and could be removed by wiping with a cloth. The deposits in sections B and C were different both in color and appearance. Most of the surface had a burnished copper or brass appearance, and behaved more like a scale. It could be scraped off, but not wiped off with a cloth. When the undersides of the shields were examined, the surface in contact with the aluminum showed occasional white spots, assumed to be aluminum oxide. No pit-of the aluminum was observed.

Deposit samples were taken from ten of the shields by scraping. Appearance, weights of samples and areas are given in Table 15. It will be noted that there is no good correlation between the amount of deposit and location or color. Except for one high and one low value, the range of densities was quite narrow and averaged 0.89 mg/in.^2 .

Spectrographic analyses of individual and composite samples of the deposits are given in Table 16.

X-ray diffraction examination showed the iron to be present largely as gamma $\text{Fe}_2\text{O}_3 \cdot \text{H}_2\text{O}$. The presence of iron is attributed to erosion-corrosion of the iron support members by the high-velocity steam. The high oxygen content in the steam accounts for the ferric rather than ferrous oxide which would normally be observed under low oxygen conditions in a condenser of this type.

Table 15

DEPOSITS FROM STEAM SIDE OF CONDENSER

(Scraped off stainless steel shields)

Location in Condenser (a)				Color of Deposit	Amount of Deposit		
Sample No.	Section	Bank (b)	Row (c)		Area Sampled (in. ²)	Amount (gm)	Concentration (mg/in. ²)
1	A	1	2	Burnished brass	21	0.0154	0.74
2	A	2	5	Dull light red-brown	11.3	0.0093	0.82
3	A	3	6	Dull light red-brown	22	0.0199	0.91
4	A	4	4	Half dull brown, half burnished brass	10	0.0044	0.44
5	B	1	5	Burnished brass	20	0.0230	1.15
6	B	2	8	Burnished brass	11.3	0.0083	0.73
7	B	4	4	Dull light red-brown	16.5	0.0127	0.77
8	C	2	6	Dull light red-brown	22	0.0166	0.76
9	D	1	2	Dull light red-brown	18	0.0182	0.01
10	D	3	6	Burnished brass	12	0.0192	1.6
							Avg. 0.89

(a) All shields from top row of tubes.

(b) Counted from north side of condenser.

(c) Counted from south side of each bank.

Table 16

SPECTROGRAPHIC ANALYSES OF DEPOSITS FROM STEAM SIDE OF CONDENSER

Element	Condenser Tube Shield Samples			
	Dull Light Red-brown Color		Burnished Brass Appearance	
	Single Shield %	Composite of Four Shields %	Single Shield %	Composite of Four Shields %
Fe	VS(a)	VS	VS	VS
Al	1.5	3	2	2
Ni	1.5	3	3	4
Cr	0.02	0.15	0.02	0.2
Cu	0.1	0.3	0.1	0.2
Pb	1.5	1.5	2	3
Na	0.08	0.7	0.08	0.7
K	0.05	0.15	0.03	0.2
Ca	0.08	0.4	0.02	0.3
B	<0.1	0.2	<0.1	0.2

(a) VS - very strong, major constituent

F. Turbine

During the June-July, 1957, shutdown, the exhaust end of the turbine casing and the last row of turbine blades were inspected.⁽¹⁸⁾ The interior surface of the turbine casing, which is Kanigen nickel plated, was free of deposits and the plating appeared to be in good condition. The last row of blading was clean and the trailing edges of the blades showed no sign of erosion.

In December, 1957, a more thorough examination was made of the turbine.⁽²⁰⁾ Activity level readings taken ten days after shutdown gave the following results in mr/hr:

No. 1 HP steam seal	2.0
First row blades	0.2
Remainder of blading	0.2 or lower
Admission valve stems	3.0
Nozzle inlets	5.0
Inlet side of trip throttle valve	2.0

The turbine in general appeared to be in excellent condition. No detectable wear or erosion could be noted on the blading or diaphragms. The blading was clean with only a small amount of deposit or discoloration noted. There was a small amount of greenish deposit on the No. 1 HP steam seal retainers and some also on top of the blade shrouds where the blade ends were peened.

There was some rust on the rotor at both the HP and LP seals in the air pressure and air vacuum regions. When this rust was removed, only very slight pitting was present. The nickel plating, covering in general everything but the rotor, was still in good condition.

For activity analyses,⁽²¹⁾ samples were taken from the HP steam end gland, from first, fourth, ninth, and thirteenth rows of turbine blades, from the condenser, and from the steam pipe just before the steam throttle valve. The samples were removed by using a bearing scraper and actually removing metal plus deposited activity. The gross gamma activity from the samples was so low that they had to be analyzed in an iron room where whole-body gamma spectra are made on humans. This work was done by C. E. Miller of the Radiological Physics Division. The spectrum obtained from all samples was essentially that of Co⁵⁸.

The percentages of the long-lived activities found on the turbine at the time of shutdown was estimated to be as follows:

Co ⁵⁸	92%
Ba ¹⁴⁰ -La ¹⁴⁰	8%
Co ⁶⁰ -Cr ⁵¹ -Fe ⁵⁹	Trace.

A rough calculation of the concentration of the Co⁵⁸ activity at the various points sampled is given in Table 17.

Table 17

Co⁵⁸ CONCENTRATION FOUND IN TURBINE

	Microcurie/in. ²
Steam end gland	108 x 10 ⁻⁴
First row of blading	5.23 x 10 ⁻⁴
Fifth row of blading	46 x 10 ⁻⁴
Ninth row of blading	28.8 x 10 ⁻⁴
Condenser tubes	0.206 x 10 ⁻⁴

The observed activities can be explained as follows: the Co⁵⁸, Co⁶⁰, Cr⁵¹, and Fe⁵⁹ all exist in the reactor water. There is a known decontamination factor of about 5×10^3 between the reactor water and the steam for solid material. It has been observed that Co⁵⁸ is the major long-lived gamma-emitting isotope in reactor water, and that there are smaller amounts of Cr⁵¹, Co⁶⁰, and Fe⁵⁹. Hence, it is not impossible that these activities were deposited in the turbine by carryover with the steam. The Ba¹⁴⁰-La¹⁴⁰ are brought to the steam system by Xe¹⁴⁰ through the following chain:

The short-lived Xe¹⁴⁰, being a gas, is stripped from the reactor water by the boiling and, on its way to through the steam system, decays to Cs¹⁴⁰, which can then deposit on a surface and decay to Ba¹⁴⁰. This mechanism only works for gases short-lived enough to decay in the steam system before being removed by the air ejectors on the main condenser.

In May, 1958, mechanical failure of a turbine blade⁽²²⁾ necessitated opening the casing and removing the diaphragms (Kanigen plated) and rotors (not Kanigen plated). A survey meter examination about five days after reactor shutdown showed the maximum detectable radioactivity level to be 2 mr/hr at 2 in.

Detailed microscopic examination of the surface of the (rotor) blades showed no evidence of stress corrosion, general corrosion, intergranular attack or other types of surface attack, and the general surface condition was found to be similar to other blades returned from service under standard steam conditions. There is no evidence that boiling water reactor steam has had any adverse effect on the blade material.

The surfaces of the blades were covered with a thin, brown-colored deposit. A green deposit was also observed on the surface of the blades, just under the shroud.

Radioactivity of the blades was very low. The beta plus gamma count did not exceed twice that of the background, as measured with a Geiger counter survey meter.

The brown deposit on the blades was a soft, nonmagnetic, amorphous solid consisting of major amounts of iron and nickel, and very small amounts of chromium, copper, lead, and zinc. The green deposit was also soft and nonmagnetic. X-ray diffraction patterns of the green deposit were similar, but not identical, with those of the nickel hydroxides. Small amounts of iron and copper were also present in the green deposit.

Examination of the diaphragms (Fig. 14) at this time showed that the Kanigen nickel plate had been completely removed in the high-velocity areas due to erosion and/or corrosion.

FIG. 14
SECTION OF TURBINE DIAPHRAGM SHOWING TYPICAL LOSS OF
NICKEL PLATE FROM NOZZLES

The base metal exposed showed no visible evidence of attack. A lack of bonding was evidenced in some spots in the low-pressure stages of the diaphragms by a flaking tendency, with no signs of attack of base metal underneath.

In September, 1958, an examination⁽²³⁾ was made of the Kanigen nickel-plate coating in a vertical section of the main steam line just before it enters the turbine throttle valve. Since welding had been carried out in this area, both heat-treated and unheat-treated Kanigen nickel plate were present.

Knoop hardness determinations showed the heat-treated Kanigen to be harder than the nonheat-treated Kanigen, 8700 vs 7800, respectively. Very little erosion occurred in the heat-treated zones, but considerable erosion occurred in the nonheat-treated areas.

An undesirable characteristic of the hardened (heat-treated) Kanigen plate was the presence of cracks which penetrated through it to the base metal. These cracks could afford ample opportunity for crevice corrosion, especially in the highly oxygenated steam (28 ppm O₂) of EBWR. The general conclusion given was that the Kanigen nickel plating was not holding up under EBWR high-velocity steam conditions, but it appeared to be satisfactory in low-velocity steam.

G. Recirculation Nozzles and Control Rod Bushings

By Spring of 1958, the activity in the subreactor room had built up to the extent that the general radiation level was about 500 mr/hr at head height on the control rod motor platform.⁽²⁴⁾

Most of the activity came from three of the four 6-inch diameter nozzles provided for eventual use with forced circulation, which, pending their eventual use, had been capped. The water in three of the nozzles (those showing high activity) had been stagnant. That in the fourth nozzle had been circulating, because this nozzle was connected to the purification system and startup heater. The surface activity level at this nozzle was only 300 mr/hr, and this was mostly background.

The nine control rod bushings were also very active, with the center rod bushing having the maximum of 65 r/hr at contact.

The buildup of activity in these nozzles and bushings was due to the deposition of suspended corrosion products or crud in these stagnant areas. When the deposits were removed from the nozzles by a flushing procedure, and from the bushings by disassembly, the activity level dropped from 50 r/hr to 4 r/hr at the nozzle and from 65 r at the control rod bushing to 2 r/hr.

About a "fist-full" of crud was removed from each of the three stagnant nozzles. Both spectrographic and wet chemical analytical results on samples of this material are given in the first two columns of Table 18. Results of a wet chemical analysis made later are given in the third column. The predominant activity found in the crud was Co^{58} .

Table 18

ANALYTICAL RESULTS ON DEPOSIT
FROM RECIRCULATING NOZZLE

Element	Spectrographic (%) 5-28-58	Wet Chemical	
		5-28-58	12-22-58
Ni	20	4.5	11.3
Fe	15	7.7	14.0
Mn	0.4	-	-
Cr	0.15	-	-
Mg	0.04	-	-
Al	VS ^(a)	35 ^(b)	26.3
Si	-	<1	

(a) Very strong - predominant.

(b) Aluminum determined on a separate sample.

There are two observations that can be made on the results in this table:

1. Due to the limitation of the spectrographic analysis in accuracy to a factor of two, an incorrect impression is given as to the relative amounts of iron and nickel that are in the deposit.
2. Although the proportions of nickel, iron, and aluminum are of the same order as in the two wet chemical analyses, the actual percentages are different, showing that the two samples do not have the same composition.

After final shutdown for 100-Mw conversion in July, 1959, the recirculation nozzles were cut off. The appearance of the corrosion product deposit in one of them is shown in Fig. 15.

FIG. 15
CORROSION PRODUCT DEPOSIT IN RECIRCULATION NOZZLE

The crud accumulating in the control rod bushings not only resulted in a buildup of activity in this area, but, in addition, caused mechanical interference with the movement of the rod. On one occasion it prevented the rod from going to its complete "in" position by about 2 in. and made necessary some changes in the design of the bushing.⁽²⁵⁾

At several times since startup of the reactor, difficulties with the crud in the bushings occurred, and on some occasions samples of the deposits were taken for analysis. Some of the results of the spectrographic and wet chemical analyses are given in Table 19.

The results in the first two columns, being on a different basis than the others, cannot be precisely compared with them. They do confirm other data in showing that in the deposits examined earlier, the amount of aluminum was relatively less than the amounts of iron and nickel, whereas

about a year later the aluminum was present in greatest amount. A satisfactory explanation of the high proportion of calcium in column three is not available.

Table 19

ANALYSES OF DEPOSITS FROM CONTROL ROD BUSHINGS

Element	Spectrographic			Wet Chemical, 5-28-58 %(b)
	1-7-57, Amount(a)	3-1-57, Amount(a)	10-2-57, %(b)	
Ag	M	--	--	--
Al	M	S	0.3	35(c)
Ca	M	--	43.3	--
Cr	VS	S	6.5	--
Cu	--	W	3.0	--
Fe	VS	VS	31.3	5
Mg	M	--	2.6	--
Mn	M	M	0.9	--
Mo	MW	M	8.7	--
Ni	VS	S	2.6	13.7
Pb	M	M	0.3	--
Sr	--	--	0.3	--
Ti	MS	M	--	--
Zr	MS	M	--	--
Si	--	--	--	<1

(a) VS = Very Strong, S = Strong, M = Moderate, and W = Weak
Accuracy - order of magnitude

(b) Accuracy - factor of 2

(c) Aluminum determined on separate sample.

A control rod bushing deposit sample taken about August 13, 1957, was subjected to radiochemical analysis for β activity, with results given in Table 20.

Table 20

RADIOCHEMICAL ANALYSIS OF DEPOSIT
FROM CONTROL ROD BUSHINGS

Radioisotope	% Total β Activity
Co ⁵⁸ , Co ⁶⁰	11
Fe ⁵⁹	4
Ni ⁶³	1
Cr ⁵¹	0.1
Ce ¹⁴⁴ , Pr	0.15
Gross β Activity: 1.1×10^6 d/m/25.4 mg	

H. Fuel Elements

1. First Fuel Element Deposit Examination

During the July, 1957, shutdown (after 1370 Mw days of operation), one fuel element, ET-5 (see Fig. 16), and two dummy elements were transferred to the fuel storage pit for examination.⁽²⁶⁾ At this time samples of the deposits on these elements were taken. Observations were made to determine the nature of the deposits (adherent or nonadherent), and samples were submitted for spectrographic and radiochemical analyses.

Three samples were taken by underwater wiping with new cellulose sponges:

1. External surface of end fuel plate of fuel element (upper portion).
2. External surface of dummy element.
3. Internal surface of dummy element.

a. Type of Deposit

All three deposits were nonadherent and readily removed by a single wipe of the sponge. The deposit on the fuel plates was reddish-brown in color and the dark grey zirconium oxide film appeared intact when the deposit was wiped off (Fig. 17). The deposit on the external surfaces of the dummy element was greenish-blue when observed in the reactor core; however, when the dummy was moved to the storage pit the color was much lighter, and finally when the deposit was removed it was a dirty white color. The appearance of both the external and internal surfaces of the dummy was not bright when the deposit was wiped off. This would be expected, since the aluminum alloy would have formed an oxide layer before the "crud" or corrosion product layer had been deposited.

FIG. 16
CORE POSITIONS OF EXAMINED FUEL ELEMENTS, AND DUMMY ELEMENTS

FIG. 17
FUEL ASSEMBLY ET-5 BEING REMOVED AFTER
1370 Mwd OPERATION IN EBWR

The deposits were very soft and flocculent in texture, and when washed off the sponge appeared to be very finely divided. An attempt was made to determine the particle size of the fuel element deposit, but by the time the sample was photographed (in the electron microscope) the particles had agglomerated. Later, a sample of deposit from the external surface of the dummy element was prepared and an electron microscope photograph taken at 3100 X. Most of the particles appeared to be less than 1.5 microns in diameter. This deposit should be similar to that on the fuel element.

The physical characteristics of the deposit on the dummy elements changed after drying. When additional samples were taken from the dummy elements recently, the deposit was very adherent and had to be scraped off. Two external surfaces had a dark reddish-brown color and the other two sides had a mottled appearance, with light-colored dots which probably were the aluminum alloy corrosion products.

The amount of deposit on the fuel element could not be determined; however, that on the external and internal surfaces of the dummy element was determined. The amounts in mg/sq in. are given in Table 21. These values were determined by scraping off the deposit and are approximate figures.

b. Spectrographic Analyses of Deposits

The analyses of the three deposits are given in Table 21. Both the fuel element deposit and the deposit from the external surface of the dummy fuel element consisted primarily of iron, nickel, and aluminum oxides in decreasing order of relative amounts present. The aluminum content of the deposit from the external surface of the aluminum dummy was higher than that from the Zircaloy clad fuel, as would be expected. The nickel content of all the deposits was considerably higher than would be expected, and is due to two factors:

1. In the insoluble corrosion products of stainless steel, the nickel content is much higher than the chromium content because chromium is more soluble.
2. The nickel pickup in the steam piping and turbine by the condensate must be quite high, as indicated by erosion of diaphragm nozzles (see Section 11-F).

The deposit from the internal surface of the dummy was quite different from the other two deposits, and the low iron content might be expected since the water in the dummies was stagnant and the deposit primarily due to the corrosion products of the aluminum alloy M-388. The high nickel content of this deposit is of significance in that the ratio of aluminum to nickel in the deposit is not the same as in the alloy (1% Ni-99% Al).

Table 21

SPECTROGRAPHIC ANALYSES OF DEPOSITS FROM
EBWR FUEL ELEMENT AND DUMMY ELEMENT

(Factor of 2 Accuracy)

Constituents* (as elements)	Deposit from Fuel Plate of Fuel Element (%)	Deposit from External Surface of Dummy Fuel Element (%)	Deposit from Internal Surface of Dummy Fuel Element (%)
Iron	17.4	35	3
Nickel	13.9	13	16
Chromium	0.1	0.3	0.05
Aluminum	5.8	10	20
Copper	0.3	0.4	--
Lead	0.2	0.3	0.1
Boron	0.2	0.5	0.4
Manganese	0.2	0.3	0.04
Sodium	**	1	0.5
Potassium	**	0.1	0.1
Calcium	**	0.08	0.05
Magnesium	**	0.1	0.02
Zirconium	--	--	--
Amount of Deposit:	†	1.1 mg/in. ² 1.4 mg/in. ²	2.2 mg/in. ²
Color of Deposit:	Reddish-brown	Reddish-brown	Dirty white (wet) buff (dry) greenish-blue (viewed through reactor water)

* This table gives major components only - trace constituents not included.

** Sample contaminated by water from the fuel storage pit - determined values not valid.

† Sampled underwater - could not get amount of deposit per square inch.

Note: The individual metal component values do not total 100 as these values are of the metals only. The anions are not included in this analysis. Practically all of the metals were present as oxides and some of the oxides may have water of crystallization present also.

c. Radiochemical Analyses of Deposits

Only long-lived activities were determined, since the reactor had been shut down in June and samples for analyses were not taken until July.

A preliminary gamma-ray spectrometer analysis made on the deposit from the internal surface of the dummy element showed its major activities to be due to Co^{58} and Co^{60} , which are products of nuclear reactions given in Table 7. The measured activity was 91×10^{-6} curie/gm of Co^{58} and 3.5×10^{-6} curie/gm of Co^{60} .

The results of the analysis for beta activity are given in Table 22. They confirm the gamma analysis, indicating that the major activity was due to Co^{58} and Co^{60} .

Table 22

RADIOCHEMICAL ANALYSES OF FUEL ELEMENT AND DUMMY FUEL ELEMENT DEPOSITS

Source of Deposit	% of Total β Activity					Gross β Activity	Uranium
	$\text{Co}^{58,60}$	Fe^{59}	Ni^{63}	Cr^{51}	Ce^{144} and Pr		
Fuel Plate	72*	3	<0.5	<0.1	<0.08	9.2×10^6 d/m/ entire sample	--
Internal Surface of Dummy Fuel Element	49	3	<0.5	<0.1	<0.14	4×10^6 d/m/59 mg	7 ± 3.5 ppm

* Examination by means of a gamma-ray spectrometer showed that the ratio of Co^{58} to Co^{60} was approximately 8 to 1 on October 7, 1957.

d. General Comments

The amount of deposit present on the fuel elements was more than had been expected, since the quality of the reactor water has been maintained at less than one-micromho conductivity during most of the reactor operation.

In view of results of examined later (see page 56), it should be noted that at this time the external surface films on fuel element and dummy appeared to be nonadherent, and that they were composed largely of oxides of iron, nickel, and aluminum, in relative amounts decreasing in that order.

There was considerable speculation about having an accumulation of "crud" in the bottom of the reactor vessel. During the July shutdown, when the reactor was open, an attempt was made to determine whether there was a layer of loose deposit at the bottom. A probe (28 ft long) was put down through one of the outer dummy holes and water

drawn up with a vacuum system. No deposit was drawn up. It could be that the probe, which was quite flexible ($\frac{1}{4}$ -in. stainless steel tubing), did not get past some of the plates located below the grid support plate.

2. Second Fuel Element Deposit Examination

Examination of the EBWR core in January, 1958, after about a year of operation showed the presence of considerable scale on fuel element ET-5 (see Fig. 16). This was evident by the flaking off of the scale from it as it was lifted out of the core toward the surface of the water in the reactor for observation. Large flakes of scale could be seen floating away from the external plate and a relatively large area, estimated to be about 20 sq in., in the highest heat flux zone from which the scale had flaked away could be observed (Fig. 18). Large flakes of scale could be seen resting on top of the assemblies still in the core. Attempts to remove more scale from this plate, around this area and elsewhere, and from the other (opposite) external plate by scraping with a piece of wood on the end of a long piece of pipe were unsuccessful. A piece of the loose scale was fished out and measured with a micrometer. It was found to be 0.003 in. in thickness. Spectrographic analysis of the scale showed 67% aluminum, 25% nickel, and 8% iron. X-ray analysis showed it to be at least 50% boehmite (α $\text{Al}_2\text{O}_3 \cdot \text{H}_2\text{O}$). The X-ray diffraction pattern was identical to that observed from samples of the corrosion layer on the aluminum-nickel dummy fuel elements.

FIG. 18

FUEL ASSEMBLY ET-5, AFTER ONE YEAR OF OPERATION

The aluminum is believed to have come from corrosion of the aluminum-1% nickel dummy fuel elements, the nickel largely from the aluminum-1% nickel, the stainless steel, and possibly from the Kanigen nickel plate on the turbine and piping, and the iron from the stainless steel corrosion products.

The long-lived activities found⁽²⁷⁾ on a sample of crud removed from an EBWR fuel plate during the January, 1958, shutdown along with their probable source are given in Table 23.

Table 23

ACTIVITIES IN EBWR FUEL ELEMENT DEPOSIT

Nuclide	Half-life	Crud, d/(min)(mg)	Probable Nuclear Reaction
Co ⁵⁸	71 d	1.43×10^7	Ni ⁵⁸ (n, p) Co ⁵⁸
Co ⁶⁰	5.3 yr	1.66×10^6	Co ⁵⁹ (n, γ) Co ⁶⁰ Ni ⁶⁰ (n, p) Co ⁶⁰
Fe ⁵⁹	45 d	2.43×10^5	Fe ⁵⁸ (n, γ) Fe ⁵⁹
Mn ⁵⁴	300 d	2.08×10^5	Fe ⁵⁴ (n, p) Mn ⁵⁴

3. Third Fuel Element Deposit Examination

In January, 1958, core loading No. 46 was changed to core loading No. 47. This involved replacing two natural uranium fuel elements in central positions 129-53 and 129-264, with enriched fuel elements from core peripheral locations (27-43 and 27-255).

Fuel Element T-23 removed from position 129-53 was submitted to the Metallurgy Division for destructive examination. Preliminary findings⁽²⁸⁾ indicate a rather nonadherent scale; however (in contrast with the scale on ET-5 described above), no spalling was observed. Spectrographic analysis (factor of 2 accuracy) indicated major constituents as follows:

Aluminum	18%
Nickel	12%
Iron	6%

X-ray diffraction data indicated the scale was largely boehmite, and there was a second phase, resembling aluminum-silicon monohydrate, present.

Although specific pre-irradiation thickness measurements on Plate 606 were not available, averaged data indicate that it increased by 4-5 mils maximum. This increase includes the sum of any volume increase plus corrosion product deposition during in-pile operation.⁽²⁹⁾

After cutting plate No. 606 into coupons, samples from varied flux regions were returned to the Reactor Engineering Division for corrosion product scale examination. Attempts were made to remove the scale by means of alternate oxalic acid-nitric acid treatments as used by WAPD⁽³⁰⁾ in descaling Zircaloy-clad specimens. Satisfactory results were not obtained, due, it is believed, to the high aluminum oxide content of the EBWR fuel element deposits. When this descaling treatment was supplemented by mechanical scraping, chipping, and brushing, the samples finally appeared to be free from scale.

In Fig. 19, the thickness of scale (mg/cm^2) as determined by weight loss was plotted against distance of the sample from the bottom of the plate.⁽³¹⁾ In the same figure the per cent of maximum flux, as determined,⁽²⁹⁾ is plotted against distance from the bottom of the plate. There is quite good correlation between maximum scale thickness and maximum flux or burnup.

4. Fourth Fuel Element Deposit Examination

Fuel element ET-51 occupied position 101-266, next to 129-370 and diagonally second from the central control rod (see Fig. 16), from startup of the reactor in December, 1956, until April 20, 1959, when it was removed. According to a flux survey made on loading No. 46 in February, 1958, this position represented the highest flux on the diagonal radius in the reactor.⁽³²⁾

In April, 1959, while making a flux plot at 40 Mw, a cobalt wire became stuck in element ET-51 and could not be removed. Because of this, and because of the interest in destructively examining a fuel element with maximum burnup, the decision was made to remove ET-51 for examination. When examined by the Metallurgy Division, this element showed areas where scale had spalled off. In the course of cutting up the element, a considerable amount of scale flaked off.⁽²⁹⁾

The thickness of the scale from the general zone of high heat flux was determined to be 0.0046 in. by measurement of photomicrographs of the scale in cross section. The thickness as measured with a micrometer was 0.005 in.,* based on several measurements.⁽³³⁾ Due to surface roughness, these values are larger than those established from photomicrographs.

*Scale specimens collected from the bottom of the reactor vessel after shutdown had thicknesses as great as 0.008 in.

FIG. 19
RELATION BETWEEN SCALE BUILDUP AND BURNUP AFTER
OPERATION FOR ABOUT ONE YEAR.

The density of the scale, determined from weight measurements and volume calculation, was 2.5 gm/cm^3 . Pieces of scale tested were found to be attracted by a magnet. Results of spectrographic and wet chemical analyses are given in Table 24. X-ray diffraction measurements indicated the presence of boehmite and of a second, unidentified phase.

The most complete analytical information on the composition of the scale is given in Table 25.

I. Dummy Fuel Elements

Results of chemical analyses of the first dummy fuel element deposits have been given in Table 21. Results of subsequent analyses are summarized in Table 26. A rather wide scatter is shown, particularly with respect to aluminum, iron and nickel; the results differ from those of fuel element deposits of about the same date in that aluminum oxide is predominant over iron and nickel.

Since fuel and dummy elements, from which deposit samples had been removed, had stood for varying periods of time in the water in the fuel storage pit, a sample of this water was analyzed as a control. In most instances the amounts of the various elements in the deposits, other than iron, nickel and aluminum, corresponded roughly with the amounts present in the water. Although there appeared to be no correlation between the calcium content of the water and that of the deposits from the dummy fuel elements, there was a correlation between the high calcium in the water and the calcium in one of the fuel element deposits analyses in Table 24.

In December, 1958, two M-388 aluminum dummy fuel assemblies which had been exposed about one year at operating temperature were removed from EBWR to the hot cell facilities of Argonne National Laboratory.

They were covered with a loose corrosion coating that varied in color from a dark brown to a medium gray.⁽³⁴⁾ Under this loose coating there was an adherent, speckled brown coating.

A sample of the loose coating, on X-ray analysis, was found to be essentially all boehmite ($\alpha \text{ Al}_2\text{O}_3 \cdot \text{H}_2\text{O}$), with only one additional X-ray line which was not identified.

In addition to the uniform attack, two shallow pits (~ 0.02 -inch diameter by 0.002 -inch maximum depth) were noted in the sections examined. These may have resulted from initial rolling inclusions or from poor water quality in the first filling of EBWR.

Table 24

SUMMARY OF FUEL ELEMENT DEPOSIT ANALYSES (IN PER CENT)

Fuel Element No.	Date of Sample	Type of Analysis	Al	B	Ca	Cr	Cu	Fe	K	Mg	Mn	Na	Ni	Pb	Si
ET-5	7- 8-57	Spectrographic(a)	5	0.2	9	0.1	0.3	15	1.5	0.8	0.2	2	12	0.2	
ET-5	1-18-58	Spectrographic(a)	VS	1	0.2	0.5	2	10		0.3			25	0.1	
ET-5	1-18-58	Spectrographic(a)	VS		0.15	0.4		7		0.3			25		
ET-5	1-18-58	Wet chemical	35(b)					6.2					13.6		<1
Unknown	12- 4-58	Spectrographic(a)	VS		0.04	0.2		17		0.03	0.2		14		
T-23	---	Spectrographic(a)	18					6					12		
ET-51	6-10-59	Spectrographic(a)	>50	0.1	0.003	0.04	1	6		0.2			16		
ET-51	6-10-59	Wet chemical	36.3					3.73					10.6		0.8

(a) Factor of 2 accuracy.

(b) Aluminum determined on a separate sample

Table 25

ANALYSIS AND CALCULATED COMPOSITION OF
EBWR FUEL ELEMENT SCALE

Element	Analysis (wt %)	Assumed Compound	X-Ray Identification	Calculated (wt %)
Al	36.3	$\text{Al}_2\text{O}_3 \cdot \text{H}_2\text{O}$	Boehmite	80.6
Ni	10.6	NiO	Not identified	12.6
Fe	3.9	Fe_3O_4	Not identified	5.1
Si	0.8	SiO_2	Not identified	1.6
Total:				99.9

Table 26

ANALYSES OF DEPOSITS FROM DUMMY FUEL ELEMENTS

Date of Sample	Location of Sample	Degree of Accuracy	Al	B	Ba	Ca	Cr	Cu	Fe	K	Li
8-12-57	Storage Pit Water (control)(a)	(b)	1.2	1.3	0.1	65.7	0.1	0.6	1.2	10.3	
8-12-57	Outside of Element	(c)	10	0.5	0.04	0.08	0.3	0.4	35	0.1	
8-29-57	Inside of Element	(c)	20	0.4		0.05	0.05		3	0.1	
10-22-57	Inside of Element	(c)	VS	0.4		0.04	0.06	1.5	2	0.02	
10-22-57	Outside of Element	(c)	VS	0.4		0.15	0.1	0.1	6	0.05	
12- 4-58	Uncertain	(b)	VS			0.08	0.08	0.7	17		

Date of Sample	Location of Sample	Degree of Accuracy	Mg	Mn	Na	Ni	P	Pb	Sn	Zn	Zr
8-12-57	Storage Pit Water (control)(a)	(b)	4.4	0.2	11.7	1.0		0.7	0.1	1.7	0.1
8-12-57	Outside of Element	(c)	0.1	0.3	1	13		0.3			
8-29-57	Inside of Element	(c)	0.02	0.04	0.5	16	<4	0.1			
10-22-57	Inside of Element	(c)	0.03	0.03	0.4	6		<0.1			
10-22-57	Outside of Element	(c)	0.06	0.06	0.4	2		<0.1			
12- 4-58	Uncertain	(b)	0.03	0.2		14					

(a) Average of 2 determinations; per cent calculated from results reported as mg/ml

(b) Order of Magnitude

(c) Factor of 2

Since these elements had been stored for a time in the fuel storage pit, and since the water in the pit was not continuously purified (and is less pure than reactor water)(see Table 26), there is a possibility that the pitting attack may have occurred here. To eliminate this possibility, two dummy fuel elements removed in July, 1959, when the reactor fuel was removed, were wrapped in plastic and stored in a dry condition; they are currently awaiting examination by the Metallurgy Division.

J. Pressure Vessel

In February, 1959, the pressure vessel was opened to install two 3-inch nozzles. This necessitated lowering the water level in the vessel to 10 ft above the core and installing a temporary platform for the workmen. Above the water level, the walls gave measured gamma readings up to 150 mr/hr (hard and soft) at 2 in. The upper surface of the steam ring gave readings up to 750 mr/hr, whereas the lower surface gave lower readings. After wiping up the loose particulate deposit with wet rags, the readings from the walls and steam ring were reduced to 25 and 150 mr/hr, respectively. The ease with which the deposited activity is removed is worthy of note.

Samples of the deposits were analyzed spectrographically and radiochemically. The results are summarized in Tables 27 and 28.⁽³⁵⁾

It is evident from the results that the compositions of these deposits do not correlate too well among themselves or with other deposits that have been described, except for the high content of iron and nickel. The proportions of calcium and lead are unexpected; and aluminum shows a wide variation among these deposits, but is surprisingly low.

In Table 28 two observations are of interest: (1) the high activity in the deposit on top of the steam ring as compared to that on the bottom side or vessel walls; and (2) the ratios of 3:4 between Co^{58} and Co^{60} .

When the core was removed in July, 1959, for 100-Mw conversion, a large number of fuel elements showed the loss of patches of scale, and the water was murky due to the amount of suspended material stirred up or released by the removal operation.

The loose crud was removed from the pressure vessel by flushing with water from a hose down through valves on control rod thimbles and recirculation nozzles provided for the occasion. A manifold system carried the slurry through a full-flow filter to a sump, and then to the retention tank. The filters were changed about a dozen times during the operation. No measurements were made, but an estimate was made that the amount was "less than a cubic foot."

Table 27

SPECTROGRAPHIC ANALYSES OF WALL
AND STEAM RING DEPOSITS

(in per cent - factor of 2 accuracy)^(a)

Element	Deposit from Wall	Deposit from Top of Steam Ring	Deposit from Bottom of Steam Ring
Al	3.8	16.3	10.5
Ca	6.3	16.3	16.7
Cr	5.7	1.6	2.8
Cu	1.3	1.6	1.4
Fe	50.5	37.9	39.0
Mg	1.3	1.1	1.4
Mn	0.6	1.1	0.6
Mo	0.6	0.2	0.1
Ni	3.8	13.6	2.8
Pb	25.3	9.2	23.6
Ti	0.3	0.4	0.4
Zr	0.6	0.8	0.7

(a) Assuming no other elements present.

Table 28

RADIOCHEMICAL ANALYSES OF WALL
AND STEAM RING DEPOSITS

	Activity [d/(min)/(mg)Fe]			
	Co ⁶⁰	Co ⁵⁸	Fe ⁵⁹	Co ⁵⁸ /Co ⁶⁰
Top of Steam Ring	3.3×10^7	1.0×10^8	5.6×10^6	3.0
Bottom of Steam Ring	2.0×10^5	8.0×10^5	2.0×10^5	4.0
Wall of Vessel	1.1×10^6	3.9×10^6	6.2×10^5	3.5

Some observations made⁽³⁶⁾ with reference to deposited activity in the pressure vessel during shutdown for 100-Mw modification were as follows:

A radiation survey made inside the reactor vessel with the water level at the top of the thermal shield gave readings ranging from 75 mr/hr at the vessel top to 1000 mr/hr six inches above the core shroud. The inner surface of the shock shield measured up to 1500 mr/hr at two inches.

Washing the walls in the upper part (steam zone) of the pressure vessel with wet rags four times reduced the radiation from the surface by about 50%.

Washing the steam ring with wet rags removed an estimated 70-80% of the activity. The highest percentage of activity on the steam ring was on the upper horizontal surface as compared with the lower surface.

Washing the shock shield repeatedly (four times) reduced the activity by only about 10%. After removing the shock shield in sections from the pressure vessel activity measured showed up to 1250 mr/hr at two inches.

Chemical cleaning (H_3PO_4 + Versene) of sections of the shock shield removed from the reactor pressure vessel failed to give a noticeable reduction in activity.

Radiochemical analysis of grinding and cutting chips from the shock shield showed the presence of only Co^{58} and Co^{60} activities. The ratio between Co^{58} and Co^{60} , calculated back to per cent total activity at time of shutdown (July 15, 1959), was roughly between 4 to 1 and 5 to 1.⁽³⁷⁾

K. Corrosion Specimens

In Section I.D.4.c a corrosion experiment was described in which corrosion samples of structural materials representative of those on the turbine were placed in a holder in the 6-inch steam line just before the dryer in November, 1958. During the next 19 months samples were removed three times for examination of the corrosion rate. At the end of these examinations the samples were examined for activity deposits. The major activities were found to be Co^{58} and Co^{60} in a ratio of about 6 to 1.⁽¹⁰⁾

L. Deposition Coupons

In connection with fuel defect studies [described in Section I.D.4.d(4)] samples representative of structural materials were placed in the steam line ahead of the dryer and in the condenser to study the activity deposited on them. Results of examination by the Chemical Engineering Division may be summarized as follows:^(38,39)

1. First Defect Test

Duration of the test was 18,800 Mw-hr.

Samples from top tubes in the condenser were examined as to nature of deposit and associated activities; the results are summarized in Table 29.

Table 29

NATURE OF DEPOSITS AND ACTIVITIES ON SAMPLES IN CONDENSER, FIRST DEFECT TEST

Metal	Appearance	Absorbed Activity, counts/(min)(cm ²)	Major Activities
Mild steel	Pitted and streaked	764	Co ⁵⁸ and Cu ⁶⁴
Stainless steel	Brown film	300	Co ⁵⁸ and Cu ⁶⁴
Copper	Brown film	330	Co ⁵⁸ and Cu ⁶⁴

Approximately the same amounts of Cu⁶⁴ activity were deposited on all samples, but a much larger amount of Co⁵⁸ was absorbed on mild steel than on either copper or stainless steel. Both wet-ground and as-received surfaces on mild steel and stainless steel samples were tested. Results showed about 10-15% less activity on wet-ground surfaces than on as-received surfaces.

2. Second Defect Test

During the second defect test, which had a duration of 15 weeks and 51,214 Mw-hr, samples of eight structural materials were exposed in the steam line ahead of the steam dryer, and in the condenser.

The materials in the steam line were:

- (a) Structural steel ASTM A113.
- (b) AISI type 4140.
- (c) 13% chromium steel.
- (d) Chrome-moly steel (%):

Carbon	0.45 (max)
Manganese	0.70-1.00
Silicon	0.20-0.35
Chromium	1.0-1.5
Molybdenum	0.45-0.65
Vanadium	0.15 (min)

- (e) Low carbon steel.
- (f) 13% chrome nonhardening steel.
- (g) Type X-8001 aluminum-1% nickel.
- (h) Type A-198 aluminum-1% nickel-0.1% titanium.

Examination of activities on a single channel analyzer showed that about the same amount of gamma activity was absorbed on all of the steel specimens, but about twice as much gamma activity was absorbed on the aluminum alloy specimens as on the steel specimens.

Materials in the condenser were:

- (a) Type 304 stainless steel.
- (b) Mild steel.
- (c) Copper.
- (d) Type A-198 aluminum-1% nickel-0.1% titanium.
- (e) Type X-8001 aluminum-1% nickel.

The samples were examined very shortly after reactor shutdown in order to pick up short-lived activities. The total counting rate for each specimen at two different times was determined to give a semi-quantitative measure of absorption of each active nuclide. The results are tabulated in Table 30.

Table 30

ACTIVITY ON SPECIMENS IN CONDENSER
DURING SECOND DEFECT TEST

Material	Counts/min/sample	
	After 0 hr ^(a)	After 72 hr ^(b)
Type 304 stainless steel	210	100
Mild steel	780	70
Copper	380	130
Type A-198 aluminum	6040	250
Type X-8001 aluminum	6890	180

(a) Major activity = F^{18} .

(b) Major activity = Co^{58} .

These results show that about ten times as much F^{18} activity is absorbed on aluminum as on the next greatly absorbing specimen, mild steel; and that stainless steel absorbs the least activity.

No F^{18} activity was detected on samples from the first defect test because, being short-lived, it had decayed out by the time the samples were examined.

The failure to detect Co^{60} in the condenser is believed to be due to its lower concentration as compared to Co^{58} and to the lower level of activity found in the condenser as compared to that in the steam line.

It is of particular significance that no fission product activity was reported on any of these samples.

M. Plant Activity Levels after Shutdown

In order to get an indication of the activity levels reached as a result of deposition of activity with time, a survey was made March 25, 1960, by Health Physics. This survey, which was made about nine months after shutdown for 100-Mw conversion, represents long-lived activity buildup around the plant following about two and one-half years of operation. The results obtained by survey meter are given in Table 31. Due to changes made below the reactor for 100-Mw conversion, no readings are reported in that area.

Table 31

EBWR BUILDING ACTIVITY SURVEY

(March 25, 1960)

	Hard Radiation	
	mr/hr	in.
4th Level - Main Floor		
Background: 1. Immediately inside airlock	0.08	
2. Between turbine and reactor	0.2	
3. Near storage well	0.17	
Main steam header:	0.17	1
Main steam chest:	0.11	1
Exhaust hood (max):	0.14	1

Table 31 (Cont'd.)

	Hard Radiation	
	mr/hr	in.
3rd Level - Condenser Floor		
Background: 1. At east stairs	0.5	
2. At west stairs	0.09	
3. Opposite air ejectors	0.3	
Steam dryer and emergency cooler:	1.3	1
Main steam line from dryer (top):	0.4	1
Relief steam line from dryer (bottom):	0.3	1
Feedwater filter No. 1:	1.5	1
Feedwater filter No. 2:	9	1
Air ejector after cooler:	0.3	1
Air ejector inter cooler:	0.3	1
Desuperheater (max):	0.4	1
2nd Level - Pump Floor		
Line from purification system on reactor face:	0.5	1
Start-up heater:	16	1
Shield cooling manifold:	0.4	1
Rod cooling water line under manifold:	6	1
Air ejector discharge line:	0.2	1
Air ejector flow meter:	0.2	1
Twin strainer:	0.25	1
Feedwater pump No. 1:	0.18	1
Feedwater pump No. 2:	7	2
Feedwater pump filter No. 1:	7	2
Feedwater pump filter No. 2:	7	1
Background: 1. Between feedwater pump and face of reactor:	0.3	
2. East stairwell:	0.7	

Table 31 (Cont'd.)

	Hard Radiation	
	mr/hr	in.
1st Level - Basement		
Retention Tank No. 1:	22	2
Retention Tank No. 2:	20	2
Background: 1. West stairwell:	2	
2. East stairwell:	0.9	
Prefilter No. 1:	18	2
Prefilter No. 2:	5	1
Ion exchanger No. 1:	21	2
Ion exchanger No. 2:	17	2
After filter No. 1:	32	2
Regenerative Cooler No. 1: 1. left end	100	2
2. right end	15	2
Secondary cooler No. 1: 1. left end	310	2
2. right end	180	2
Regenerative cooler No. 2: 1. left end	130	2
2. right end	20	2
Secondary cooler No. 2: 1. left end	240	2
2. right end	100	2

N. Conclusions

The decontaminating effect of the boiling process as carried out in EBWR has been extremely effective in preventing the spread of activated corrosion products from the reactor to and through the steam system. This has been brought out in the experimentally determined decontaminated factor of about 5×10^3 and in the many observations as to the difference in dose level between deposits from reactor water, such as those in purification system filters and ion exchanger, startup heater, control rod drive thimbles, and recirculation nozzles; and those in the steam and condensate systems such as turbine, condenser, and feedwater filters.

Corrosion product deposits up to 0.008 in. in thickness have been observed on EBWR fuel element plates or in the form of scale flaked off the plates. Scale thickness appears to be a resultant of two processes: a tendency to build up in thickness as a function of fuel burnup, and a tendency to spall or flake off in areas where high buildup has occurred. Since the scale on analysis showed about 80% boehmite, α - $\text{Al}_2\text{O}_3 \cdot \text{H}_2\text{O}$, corrosion of the aluminum-nickel alloy, M-388, dummy fuel elements is believed largely responsible. Other elements in the scale were iron, nickel, and silica in a form not identified. Both Co^{58} and Co^{60} in the scale were responsible for the long-lived activity, the high level of activity being due to their activation while deposited on the fuel elements. Activation on the fuel followed by spalling and deposition or settling in low velocity areas was responsible for high activity in the circulation thimbles and control rod bushings.

III. THERMAL CONDUCTIVITY OF EBWR FUEL ELEMENT SCALE

A. Introduction

The scale deposit on the EBWR fuel elements (see Section II-G) has a pronounced effect on the operating temperatures of the individual fuel plates. The maximum EBWR fuel centerline temperature is given as 767°F, based on a total heat generation rate of 100 Mw and a maximum heat flux of 485,000 Btu/(hr)(ft²).⁽⁴⁰⁾ The calculation of this temperature involves the assumption that there is no scale deposit on the surfaces of the fuel plates. It is apparent, therefore, that any additional thermal resistance present on the surface of the fuel plate will increase this centerline temperature by an amount essentially equal to the temperature drop through the added resistance, assuming no changes in heat flux or surface temperature.

It was believed that, as fuel temperatures approached 900°F, growth of the uranium alloy may be observed.⁽⁴¹⁾ It became necessary, then, to determine the thermal conductivity of the scale deposit in order to make a good estimation of the maximum fuel temperature that may be present under varying conditions of scale formation.

B. Test Procedure

Samples of the EBWR scale were recovered from the pressure vessel during the July, 1959, shutdown of the EBWR. These pieces of scale ranged in thickness from 0.004 in. to 0.008 in. The surface areas of the specimens used were measured by tracing the outline of the scale on graph paper and counting squares. This procedure was used because of the irregularity of the surface and because its radioactivity precluded direct handling of the scale.

The scale was then pressed between the two copper rods of a thermal conductivity apparatus. This apparatus is described in detail in Section C below. The heat flux and temperature drop between the copper rods were determined in the following manner:

After the system had come to equilibrium for each of the experimental runs, four temperatures were measured, as shown in Fig. 20. Since T_1 and T_2 are spaced the same distance apart as T_3 and T_4 , one would expect that $T_1 - T_2$ would equal $T_3 - T_4$ if there were no heat losses. Actually, the latter drop is always less than the former, as one would expect in a real system.

Since the temperatures in the upper rod are usually much higher than those in the lower rod, it is reasonable to assume that most of the heat loss occurs from the former. Hence, in calculating the heat flux through the scale, only the temperature drop $T_3 - T_4$ was considered,

and heat losses were neglected. The product of the temperature gradient and the thermal conductivity of copper gives the heat flux through the scale.

The temperature drop across the gap, $T_2 - T_3$, was corrected to allow for the drop across the $\frac{1}{4}$ in. of copper below T_2 and above T_3 , and this corrected drop was considered the drop across the scale, ΔT .

FIG. 20
TEMPERATURE MEASUREMENT SCHEMATIC

C. Thermal Conductivity Apparatus

The apparatus used in this test provides a means for pressing a specimen of fuel element scale between the faces of two copper rods. It is required that the copper rods be concentric and the faces parallel.

A drawing of this apparatus is given in Fig. 21. Four tie rods extend from the base plate and hold the upper support plate in a fixed position. The heated section and upper copper rod are integral. The heated section is held to the upper support plate by three bolts.

FIG. 21
THERMAL CONDUCTIVITY APPARATUS

An insulating spacer of Micalex with uniform cross section thermally isolates the heated section from the upper support plate; the only direct paths for heat flow from the heated section to the upper support plate are via the three bolts previously mentioned. Although it is not indicated on the diagram, a coil of water-cooled copper tubing is pressed against the upper support plate to prevent a large temperature increase of this component.

The lower copper rod and its cooling block are bolted to the lower support plate. The lower support plate assembly contains four ball bushings which slide over the tie rods. The lower support plate is spring-loaded, and this makes it possible to press a scale specimen between the faces of the copper rods. This spring loading also provides for expansion of the copper rods as heat is applied.

Thermal insulation is used only around the heated section and the copper rods. The tie rods and other sections are kept cold to prevent thermal warping with consequent misalignment of the copper rods.

The faces of the copper rods are lapped optically flat. Iron-constantan thermocouple junctions are at the axis of each copper rod. The thermocouples lead in through radial holes of small diameter.

A Teflon sleeve is held around the gap that exists between the faces of the copper rods. This sleeve makes it possible to maintain gas atmospheres other than air between the faces. This provision is not indicated in Fig. 21.

D. The Comparative Method for Determining Thermal Conductivity of a Thin Specimen

The main problem that one encounters in a thermal conductivity measurement of a thin material by the means described above is that it is difficult to evaluate the contact resistance at the copper-scale interface. The data from this test were used to determine the thermal conductivity of the scale by a novel method, which we have called the comparative method for want of a better term. After completion of these experiments, it was discovered that a similar method had been employed 20 years previously to obtain the thermal conductivity of rocks.⁽⁴²⁾

If the scale and the copper rod were in perfect contact, there would be no contact resistance at the interface. However, due to the irregularity of the scale surface and the usual human limitations in making anything "perfect," there will always be some gaps, often immeasurable, between the surfaces in contact. These gaps will normally contain air, which has a thermal conductivity that is low relative to metals and even to oxide films similar in composition to the EBWR scale. Hence, the temperature drops across these gaps may be significant compared to the drop across the scale.

FIG. 22
FIRST CONTACT RESISTANCE MODEL

Two models for the contact resistance were assumed in order to evaluate the thermal conductivity of the scale. The first of these is illustrated in Fig. 22. Here, one surface of the scale is assumed to be in "perfect" contact with the copper, and the other surface in zero contact with a gap between scale and copper.

The heat flow analysis of this model is as follows:

$$Q = q_c'' A_c + q_s'' A_s \quad , \quad (1)$$

where

Q = total heat flow through copper, Btu/hr

A_s = area of scale, ft²

A_c = area of copper - area of scale, ft²

q_c'' = heat flux through A_c , Btu/(hr)(ft²)

q_s'' = heat flux through A_s , Btu/(hr)(ft²) .

Now,

$$q_c'' = k_g \frac{\Delta T}{L} \quad , \quad (2)$$

where

k_g = thermal conductivity of the gas between copper faces,
Btu/(hr)(ft)(°F)

ΔT = temperature drop between copper faces, °F

L = distance between copper faces, ft ,

and

$$q_s'' = \frac{\Delta T}{\left(\frac{t}{k_s} + \frac{x}{k_g} \right)} \quad , \quad (3)$$

where

t = scale thickness, ft

k_s = scale thermal conductivity, Btu/(hr)(ft²)(°F)

x = gap thickness = $L - t$, ft .

Substituting Equations (2) and (3) in Equation (1) and multiplying by $1/A_s \Delta T$,

$$\frac{Q}{A_s \Delta T} = \frac{A_c}{A_s} \frac{k_g}{L} + \frac{1}{\left(\frac{t}{k_s} + \frac{s}{k_g}\right)} , \quad (4)$$

Let

$$f = \frac{A_c + A_s}{A_s} = \frac{A}{A_s} ,$$

where

A = area of copper rod, ft² .

Then

$$\frac{Q}{A_s \Delta T} = \frac{fQ}{\Delta T A} = q'' \frac{f}{\Delta T} , \quad (5)$$

where

q'' = heat flux through copper rod, Btu/(hr)(ft²) .

Now,

$$A_c/A_s = f - 1 . \quad (6)$$

Substituting Equations (5) and (6) into Equation (4) and rearranging,

$$q'' \frac{f}{\Delta T} - \frac{(f - 1)k_g}{L} = \frac{1}{\left(\frac{t}{k_s} + \frac{x}{k_g}\right)} , \quad (7)$$

Since $x \ll t$, we have $L \approx t$ and the left side of Equation (7) can be considered constant. We can measure all the quantities on the left, and for a given piece of scale and set of experimental data, this constant will change only if the gas atmosphere or thermal conductivity is changed. It develops that for a given gas and scale specimen, the constant remains essentially the same regardless of heat flux. We shall call this constant the correlating parameter C .

It is apparent that Equation (7) contains two unknowns, x and k_s . Therefore, if experimental runs are duplicated, using first air and then helium to blanket the heat transfer surfaces, we can solve the resulting two independent equations for these unknowns, and in particular for k_s .

This model and procedure were employed in conjunction with the measurement of the thermal conductivity of Pyrex Brand No. 7740 glass. It was thought that a test on such a standard specimen as Pyrex might give some indication of the worth of the apparatus and method. However, the reported values of thermal conductivity of Pyrex⁽⁴³⁾ vary too much to draw any concrete conclusions.

The data from this series of tests are shown plotted in Fig. 23. Theoretically, the data for any one gas and glass specimen should coincide, but due to (1) certain simplifying assumptions in the data reduction indicated in Section C above; (2) variation of gas thermal conductivity with temperature; and (3) the usual experimental errors, scatter of the experimental points can be expected. The values for thermal conductivity according to these data lie between 0.430 and 0.550 Btu/(hr)(ft)(°F), with a mean value of 0.490 Btu/(hr)(ft)(°F). Using the most recently reported values of thermal conductivity⁽⁴³⁾ and averaging between the actual temperatures of this test, a mean value of 0.629 Btu/(hr)(ft)(°F) was obtained. This represents an error of about 22%, which is not unreasonable considering the simplicity of the model and the uncertainty of the true value.

FIG. 23
MEASURED THERMAL CONDUCTIVITY OF PYREX (7740)

Further tests were made to determine the thermal conductivity of helium by using glass spacers of negligible area between the copper rods. The mean of the reported values of thermal conductivity for helium⁽⁴⁴⁾ at three test temperatures is 0.102 Btu/(hr)(ft)(°F), while the mean of the calculated values at these temperatures is 0.112, an error of only 10%.

Finally, a series of runs were made on the EBWR scale. Some of the pertinent data obtained are listed in Table 32 and plotted in Figs. 24 and 25.

Table 32

EXPERIMENTAL DATA FOR THE DETERMINATION OF SCALE THERMAL CONDUCTIVITY

Curve	Gas	k_g , Btu/(hr)(ft)(°F)	q'' , Btu/(hr)(ft ²)	f	ΔT	$\frac{fq''}{\Delta T} - (f-1)\frac{k_g}{t}$ (correlating parameter, C)	$t \times 10^4$, ft
A	Helium	0.11	100,000	2.38	174	954	3.67
B	Helium	0.1	52,200	2.38	93	954	3.67
C	Helium	0.11	100,000	2.38	170	986	3.67
D	Air	0.0186	38,800	2.38	183	436	3.67
E	Air	0.022	75,500	2.38	315	486	3.67
F	Air	0.0186	38,800	2.38	176	456	3.67
G	Helium	0.098	48,800	2	97	755	3.84
H	Helium	0.094	32,200	2	66	730	3.84
J	Helium	0.108	93,200	2	167	838	3.84
K	Air	0.0192	37,800	2	199	329	3.84
L	Air	0.0215	63,200	2	302	364	3.84
M	Air	0.0174	22,200	2	126	307	3.84

FIG. 24
MEASURED THERMAL CONDUCTIVITY OF SCALE FOR $f = 2.38$

FIG. 25
MEASURED THERMAL CONDUCTIVITY OF SCALE FOR $f = 2.0$

FIG. 26
SECOND CONTACT RESISTANCE MODEL

The values of thermal conductivity ranged from 0.398 to 0.487 Btu/(hr)(ft)(°F). The mean was 0.44 Btu/(hr)(ft)(°F). If we allow an error of 20%, to be consistent with the glass data, we conclude that the thermal conductivity of the scale is 0.44 ± 0.09 Btu/(hr)(ft)(°F).

The second contact resistance model considered is shown in Fig. 26. Whereas in the previous model the scale was assumed to be in zero contact with the face of one of the copper rods, the second model allows a fraction of scale to be in perfect contact with that face. The analysis of the heat flow in this model follows, with the nomenclature remaining unchanged:

$$q_s'' A_s = q_0'' A_0 + q_p'' A_p \quad , \quad (8)$$

where

$$\begin{aligned} A_0 &= \text{area of scale in series with contact resistance } x, \text{ ft}^2 \\ A_p &= \text{area of scale in perfect contact with upper copper rod, ft}^2 \\ q_0'' &= \text{heat flux through } A_0, \text{ Btu}/(\text{hr})(\text{ft}^2) \\ q_p'' &= \text{heat flux through } A_p, \text{ Btu}/(\text{hr})(\text{ft}^2) \end{aligned}$$

Now,

$$q_0'' = \frac{\Delta T}{\frac{t}{k_s} + \frac{x}{k_g}} \quad (9)$$

and

$$q_p'' = k_s \frac{\Delta T}{L} \quad (10)$$

Substituting Equations (9) and (10) into Equation (8) and multiplying by $1/A_s \Delta T$,

$$\frac{q_s''}{\Delta T} = \frac{1}{\left(\frac{t}{k_s} + \frac{x}{k_g}\right)} \frac{A_0}{A_s} + \frac{k_s}{L} \frac{A_p}{A_s} \quad (11)$$

Let

$$F = \frac{A_p}{A_s} = \frac{A_s - A_0}{A_s} = 1 - \frac{A_0}{A_s} \quad (12)$$

Then

$$\frac{q_s''}{\Delta T} = \frac{(1 - F)}{\left(\frac{t}{k_s} + \frac{x}{k_g}\right)} + \frac{F k_s}{L} \quad (13)$$

Equation (7) may now be rewritten as follows:

$$q'' \frac{f}{\Delta T} - \frac{(f - 1)k_g}{L} = \frac{1 - F}{\left(\frac{t}{k_s} + \frac{x}{k_g}\right)} + \frac{F k_s}{L} \quad (14)$$

As one would expect, Equation (14) reduced to Equation (7) if $F = 0$.

The correlating parameter on the left side of Equation (14) is now a function of F , k_s , and x , all of which are unknown. Three independent

equations will result if data are taken using three different gases with significantly different thermal conductivities. Three such gases are air, helium, and krypton.

One run was made with each gas. The resulting equations were solved on the IBM-704 with the following answers:

$$k_s = 0.43 \text{ Btu}/(\text{hr})(\text{ft})(^\circ\text{F})$$

$$F = 0.077$$

$$x = 4.84 \times 10^{-5} \text{ ft}$$

E. Fuel Temperature Calculations

Based upon the above measurement of k for the EBWR scale, one can calculate the maximum temperature that the fuel will attain under a variety of scaling conditions. Using the values of 767°F and $485,000 \text{ Btu}/(\text{hr})(\text{ft}^2)$ for the maximum fuel centerline temperature and heat flux, respectively, $15 \text{ Btu}/(\text{hr})(\text{ft})(^\circ\text{F})$ and $8 \text{ Btu}/(\text{hr})(\text{ft})(^\circ\text{F})$ for the thermal conductivity of fuel alloy and Zircaloy-2, respectively, and assuming that the heat generation rate is uniform through any cross section, the maximum cladding surface temperature was calculated to be 504°F . The following calculations were made for a thick (0.280 in.) fuel plate with the same physical constants as above and assuming that the surface temperature remains constant at 504°F .

Case I. 0.002-in. of Scale Deposited on One Heat Transfer Surface

Figure 27 illustrates the heat transfer model and a representative temperature distribution in which the maximum temperature is shifted from the fuel centerline due to the asymmetrical thermal resistances.

FIG. 27
REPRESENTATIVE TEMPERATURE DISTRIBUTION
CASES I AND II

Here $T_2 = T_5 = 504^\circ\text{F}$. The temperature distribution in the fuel satisfies the equation

$$\frac{d^2T}{dx^2} = -\frac{Q}{k_f} \quad , \quad (15a)$$

where

Q = rate of internal heat generation, $\text{Btu}/(\text{hr})(\text{ft}^3)$

k_f = thermal conductivity of fuel, $\text{Btu}/(\text{hr})(\text{ft})(^\circ\text{F})$.

The temperature gradient is, after integrating Equation (15a) once,

$$\frac{dT}{dx} = -\frac{Qx}{k_f} + C_1 \quad . \quad (15b)$$

Now,

$$q_u'' = k_f \left(\frac{dT}{dx} \right)_{x=0} = k_f C_1 \quad (15c)$$

and

$$q_s'' = k_f \left(\frac{dT}{dx} \right)_{x=a} = k_f C_1 + Qa \quad , \quad (15d)$$

where

q_u'' = heat flux on unscaled surface, $\text{Btu}/(\text{hr})(\text{ft}^2)$

q_s'' = heat flux on scaled surface, $\text{Btu}/(\text{hr})(\text{ft}^2)$

a = meat thickness, ft.

From Equation (15b) we can evaluate the distance x at which T is maximum as follows:

$$\frac{dT}{dx} = -\frac{Qx}{k_f} + C_1 = 0 \quad , \quad (15e)$$

from which

$$x = \frac{C_1 k_f}{Q} \quad . \quad (15f)$$

Using Equations (15c) and (15d) we can now write

$$x = \frac{q_u'' a}{q_u'' + q_s''} \quad . \quad (15g)$$

If we sum the temperature drops on the unscaled side of T_0 , we get, after substituting the appropriate constants,

$$(T_0 - 504) = q_u'' \left(\frac{x}{30} + \frac{1}{4800} \right) \quad (16)$$

Similarly, on the scaled side of T_0 ,

$$(T_0 - 504) = q_s'' \left(\frac{0.02 - x}{30} + \frac{1}{4800} + \frac{1}{2640} \right) \quad (17)$$

Now, equating Equations (16) and (17) and noting that

$$q_s'' + q_u'' = 970,000 \text{ Btu}/(\text{hr})(\text{ft}^2) \quad ,$$

there is obtained

$$q_s'' \left(\frac{0.02 - x}{30} + 5.56 \times 10^{-4} \right) = 970,000 - q_s'' \left(\frac{x}{30} + 2.08 \times 10^{-4} \right) \quad (18)$$

Rearranging Equation (18), we get

$$q_s'' = 138,000 + 2.21 \times 10^7 x \quad (19)$$

Combining Equations (15g) and (19), we get

$$q_s'' = 398,000 \text{ Btu}/(\text{hr})(\text{ft}^2)$$

and

$$q_u'' = 572,000 \text{ Btu}/(\text{hr})(\text{ft}^2) \quad .$$

Then

$$x = 0.0118 \text{ ft} \quad .$$

From Equation (16) ,

$$(T_0 - 504) = 344^\circ\text{F} \text{ and } T_0 = 848^\circ\text{F} \quad .$$

Case II. 0.004 in. of Scale Deposited on One Heat Transfer Surface

This case is the same as Case I except that the scale thickness has been doubled. Equation (16) remains unchanged, but Equation (17) becomes

$$(T_0 - 504) = q_s'' \left(\frac{0.02 - x}{30} + \frac{1}{4800} + \frac{1}{1320} \right) \quad (20)$$

Solution of these equations as before yields

$$q_s'' = 339,000 \text{ Btu}/(\text{hr})(\text{ft}^2)$$

$$q_u'' = 631,000 \text{ Btu}/(\text{hr})(\text{ft}^2)$$

$$x = 0.0130 \text{ ft}$$

and

$$T_0 = 909^\circ\text{F} \quad .$$

Case III. 0.002 in. of Scale Deposited on Both Heat Transfer Surfaces

In this case, the temperature distribution is symmetrical about the fuel centerline as shown in Fig. 28. The total temperature drop through each half of the fuel plate is

$$(T_0 - 504) = q'' \left[\frac{b}{2k_{\text{fuel}}} + \frac{c - b}{K_{\text{Zr-2}}} + \frac{t}{k_{\text{scale}}} \right] \quad . \quad (21)$$

Substituting the appropriate constants, we get

$$T_0 = 949^\circ\text{F} \quad .$$

FIG. 28
REPRESENTATIVE TEMPERATURE DISTRIBUTION
CASES III - V

Case IV. 0.004 in. of Scale Deposited on Both Heat Transfer Surfaces

This case is the same as Case III except the scale thickness is doubled. Equation (21) then gives

$$T_0 = 1133^{\circ}\text{F} \quad .$$

Case V. 0.008 in. of Scale Deposited on Both Heat Transfer Surfaces

This case probably represents the worst condition of scale formation. Equation (21) gives the result

$$T_0 = 1499^{\circ}\text{F} \quad .$$

All of the above T_0 temperatures have been recalculated, allowing the thermal conductivity of the scale to vary $\pm 20\%$. The maximum and minimum T_0 temperatures thus calculated are listed in Table 33.

Table 33

CALCULATED MAXIMUM AND MINIMUM T_0 TEMPERATURES

Case	T_0 (Maximum), $^{\circ}\text{F}$	T_0 (Minimum), $^{\circ}\text{F}$
I	864	834
II	935	892
III	998	919
IV	1229	1070
V	1692	1375

F. Conclusions

The highest fuel temperature attainable, 1692°F , was calculated using the most pessimistic values of scale thickness and thermal conductivity. Even the most optimistic calculation considering the thinnest scale specimen recovered (0.004 in.) gives a temperature bordering on 900°F . If the scale does not flake off the fuel plates at these elevated temperatures, and if the premise concerning fuel plate growth at temperatures above 900°F is correct, then one would indeed expect growth problems to develop as the EBWR begins 100-Mw operation.

The thermal conductivity of the EBWR scale measured using the comparative method appears to be independent of the choice of the contact resistance model as presented above. This is probably due to the fact that the scale area in contact with the upper copper rod is only about 8% of the total area. Conceivably, if a smoother specimen were used this percentage would be larger and the effect on the value of the thermal conductivity would be greater.

The analysis presented above assumes that the thermal conductivity of the scale is constant regardless of the gas medium in the pores of the scale. In view of the high conductivity of the scale relative to the gases used in this test and the estimated low porosity, this is not an unreasonable assumption. Since the temperature of the scale throughout these tests was only within about a 200°F range, temperature variation of the thermal conductivity has been neglected.

The comparative method of determining the thermal conductivity of thin specimens appears to show some promise for future applications. Some idea of the accuracy of this method may be obtained if one were to find a material whose value of k/t is about the same as for the scale and whose thermal conductivity could be guaranteed within 20% or less. This material could then be used as a standard for error determination. Unfortunately, the Pyrex used in the preliminary tests of this method cannot be so described.

Further improvement in the apparatus may be effected by devising a more positive means of keeping different atmospheres in contact with the specimen regardless of its thickness. In this case, t could be larger than about 0.010 in. and the value of k/t could thus be adjusted to give suitable temperature drops for any heat flux.

IV. THE EFFECTS OF HIGH-TEMPERATURE HEATING ON EBWR FUEL ELEMENT DESCALING AND GROWTH

A. Introduction

The scale which has deposited on the EBWR fuel plates has proven to be extremely adherent. While the scale itself is very brittle, hammering on the fuel plate has had no observable effect on scale cracking or loosening. The only chemical means of removing the scale cannot be employed without attacking the Zircaloy-2 cladding.

It was felt that the extremely good bond between scale and cladding was somehow dependent upon the water of hydration of the boehmite ($\text{Al}_2\text{O}_3 \cdot \text{H}_2\text{O}$). If the water could be driven off, the scale might flake off, or at least become loosened to the point where wire brushing might be effectively employed. On the basis of this theory, coupons from fuel element ET-51 were heated to 850°F in an inert atmosphere for 8 hr, after which time the scale did, in fact, curl and flake off. This may be seen in Fig. 29.

FIG. 29

COUPON FROM FUEL ASSEMBLY ET-51
HEATED 8 hr AT 850°F IN ARGON.

After the encouraging results with the fuel plate coupons, a program was set up to test this method of scale removal on a complete subassembly and eventually to employ this means on a production basis to descale all the fuel elements in the EBWR fuel storage pit. The following discussion is concerned with two phases of this program: (1) the heating of subassembly H-18 to 850°F for descaling purposes; and (2) the post-irradiation heating of fuel plates from this fuel element for growth studies.

A complete description of the oven used in these tests is given in Section G below. The work was carried out in the hot-cell facilities of Argonne National Laboratory.

B. Test Procedure for Descaling Operation

Measurements of the overall width of the fuel subassembly between the outermost fuel plates were made before and after heating. It was hoped that these data would furnish some information about the growth of an irradiated fuel subassembly due to heating.

The fuel element was slowly heated to 850°F, held at this temperature for twenty-five hours, and then allowed to cool slowly. Throughout the heating and cooling, the subassembly was blanketed with argon gas. Photographs of the fuel plates were taken before and after the test and are shown in Figs. 30 and 31. Figure 32 shows end views of the fuel assembly before and after heating.

C. Results of Descaling Operation

The most significant result of the descaling test was that the $\frac{1}{16}$ -inch thick Zircaloy-2 side plates buckled along their lengths between the spot welds on the fuel plates. This can be seen in Figs. 33 and 34. The buckling amounted to at least $\frac{1}{8}$ in. on each side. Even if this treatment had been successful in removing scale, the side-plate deformation would preclude subsequent use of the fuel element in the reactor due to insufficient clearances.

One theory concerning the cause of the plate buckling considers that the fuel plate has a coefficient of thermal expansion greater than that of Zircaloy-2. As the fuel plates heat up and expand, they stress the side plates beyond the yield strength. Upon cooling, the greater fuel plate shrinkage causes the side plates to buckle outward from the spot welds. Since the cladding is in compression at room temperature and since the predicted radiation growth has not materialized, it is quite possible that the cladding never becomes stressed beyond the yield strength, and thus maintains its integrity.

FIG. 30

UPPER AND LOWER SECTIONS OF FUEL PLATE "A" BEFORE
AND AFTER HEATING FOR 25 HR AT 850°F

FIG. 31

UPPER AND LOWER SECTIONS OF FUEL PLATE "C" BEFORE
AND AFTER HEATING FOR 25 HR AT 850°F

BEFORE

AFTER

FIG. 32

END VIEW OF FUEL ASSEMBLY BEFORE AND AFTER HEATING
FOR 25 HR AT 850°F

BUCKLED SIDE
PLATES

FIG. 33

EBWR FUEL ASSEMBLY IN OVEN

FIG. 34
CLOSEUP VIEW OF SIDE PLATE BUCKLING

The thin scale coat which was present on fuel assembly H-18 was not removed by heating. This can be seen by examining Figs. 30, 31, 33, and 34. Marks on several of the plates are due to scratching. The designations "A" and "C" refer to each of the outside fuel plates.

It is believed that scale removal by this method is a function of scale thickness. While the coupons from assembly ET-51, which had a thick scale coat (~ 0.005 in.), were successfully descaled, there were some coupons with a much thinner coat which were not descaled. Similarly, sub-assembly H-18 had been in a region of low thermal flux, (positions 27-219 and 27-255)* achieving a burnup of about 0.1 atom %⁽⁴⁵⁾ and had a scale coat of less than 0.002 in. If the test had been a success otherwise, it was planned to repeat this test using a fuel element with high burnup and hence a thick scale coat, to prove or disprove this thesis.

Finally, the measurements of the assembly previously described indicated little or no fuel plate deformation caused by the heating. Table 34 summarizes these data.

These data are necessarily limited due to the inaccessibility of individual fuel plates for measurement. For this reason, further tests were conducted on the entire subassembly and on individual fuel plates from that subassembly. Details of these tests follow.

*Fig. 16.

Table 34

HEAT-PRODUCED CHANGES IN FUEL PLATE DIMENSIONS

Position*	Distance across Fuel Plates (in.)	
	Before Heating	After Heating
1	3.526	3.523
2	3.524	-
3	3.522	3.523
4	3.526	3.512
5	3.533	3.516
6	3.519	3.518

*Positions along the fuel element are arbitrary.

D. Test Procedure for Growth Studies

After completion of the descaling operation, EBWR fuel element H-18 was heated to 600°C for 100 hr. Measurements of the lengths of the two outermost fuel plates were made before and after heating. Two adjacent fuel plates were then broken loose from the side plates. The thickness of plate No. 558 was measured at twenty locations, after which this plate was heated to 700°C for twenty hours. Its length was measured before and after heating.

The length of plate No. 528 was measured and the two plates were then heated together to 700°C for three hours. Length measurements were made and then the plates were heated to 700°C for 46 additional hours. After this heating, the length and several thicknesses of plate No. 528 were measured as was the length of plate No. 558.

E. Results of Growth Studies

Heating the fuel element to 600°C for 100 hr caused additional buckling of the side plates to about $\frac{1}{4}$ in., lending credence to the buckling theory proposed above. In addition, white oxide of zirconium was found beneath the black oxide initially present, as shown in Fig. 35. This was observed after the black oxide was removed by mechanically abrading the surface of the side plates. The oxidation of the zirconium may be attributed to either or both of the following factors:

1. The high temperature caused excessive buckling of the furnace, thus breaking the seal (see Section G below) and allowing relatively cool air to blanket the assembly.

BEFORE

AFTER

FIG. 35

EFFECT OF MECHANICAL ABRADING ON SIDE PLATES
HEATED FOR 100 HR AT 600°C

2. The argon gas used was welding-grade argon which contains less than 1% oxygen. This oxygen content might well have been enough to cause oxidation under such stringent heating conditions. Some idea as to the amount of oxidation which occurred may be obtained by

observing the quantity of oxide scale which was left in the furnace by the fuel plate facing the bottom of the furnace. This is shown in Fig. 36, which was taken immediately upon removal of the fuel element from the furnace.

FIG. 36

OXIDE SCALE IN FURNACE AFTER REMOVAL
OF FUEL ELEMENT H-18

The length of plate No. 558 before and after heating at 600°C was 54 in. and $54\frac{3}{32}$ in., respectively, indicating a lengthwise growth of 0.174%. Plate No. 503 measured 54 in. both before and after this test. Although thickness measurements of fuel plate No. 558 were made after this heating, no quantitative conclusions of growth in thickness can be drawn due to the previously mentioned oxide scale formation and flaking off.

The length of plate No. 558 after heating at 700°C for 24 hr was $54\frac{3}{8}$ in., indicating a lengthwise growth of $\frac{9}{32}$ in., or 0.520% from its previous condition. This heating caused the continued formation of zirconium oxide on the fuel plate, which then exhibited the frost-like appearance shown in Fig. 37. Visual inspection of the fuel plate revealed no failure or bulging of the cladding.

FIG. 37

FUEL PLATE NO. 558 HEATED FOR 20 HR AT 700°C

Prior to the initial 700°C heating of the two plates together, the length of plate No. 528 was measured as $54\frac{3}{32}$ in. After three hours at 700°C, neither plate showed a noticeable increase in length. Forty-six additional hours of heating at 700°C caused plate No. 528 to grow to a length of $54\frac{1}{4}$ in., or an increase of 0.278% in its initial length. There was no indication of cladding swelling or failure.

This last heating, however, caused spectacular failure of plate No. 558. The excessive oxidation of the Zircaloy-2 resulted in the failure of the cladding and the subsequent abnormal swelling of the fuel alloy in two regions of the fuel plate. This plate failure is illustrated in Fig. 38. The final length was about $55\frac{1}{4}$ in.

F. Analysis of Growth Data

Prior to insertion in the reactor, the EBWR fuel elements were heated into the gamma phase of the fuel alloy and then beta quenched.⁽⁴⁶⁾ It is believed that this heat treatment, to a large degree, removed the anisotropy of alpha uranium. Hence, it has been assumed that growth of an irradiated fuel element due to heating is uniform in all directions.

SIDE

FRONT

BACK

FIG. 38

VIEWS OF PLATE NO. 558 FAILURE AFTER
HEATING FOR 69 HR AT 700°C

If this is the case, then we can calculate the per cent change in volume of a fuel plate by measuring only the per cent change in length. In this manner we can arrive at quantitative conclusions about growth without having to worry about how much zirconium oxide was formed and removed from the fuel plate. Table 35 summarizes the measured and calculated data from this test.

Table 35

FUEL PLATE GROWTH DATA

Plate No.	Length, in.					$\Delta L/L$, %*				Total $\Delta L/L$, %**	$\Delta V/V$, %***	
	After		After 700°C			After 600°C	After 700°C				After	
	450°C	600°C	20 hr	49 hr	69 hr		20 hr	49 hr	69 hr		600°C	700°C
503	54	54	-	-	-	0	-	-	-	0	0	-
558	54	$54\frac{3}{32}$	$54\frac{3}{8}$	-	$54\frac{1}{4}$	0.174	0.520	-	1.61	2.315	0.523	7.107
528	-	$54\frac{3}{32}$	-	$54\frac{1}{4}$	-	-	-	0.278	-	0.278	-	0.836

NOTE: Plate Nos. 503 and 558 were outside plates.

*L = Length measured prior to each heating; ΔL = Change in length after each heating.

**L = Initial measured length; ΔL = Total change in length throughout test.

***V = Initial volume; ΔV = Total change in volume throughout test.

G. Oven Design

The oven used in this test was designed with two problems in mind. First, due to the space limitation in the cave facility, the fuel sub-assembly could not be inserted in an autoclave-type oven through an open end. It was necessary, therefore, to construct the oven in two sections. The fuel element would be lowered lengthwise into one section and the second section would then be lowered onto the first.

The second problem to be resolved concerned the need for an inert atmosphere around the fuel subassembly to prevent zirconium reacting with air. While a positive seal would have been desirable, the importance of immediate results precluded spending a great deal of time on such a development. Even with a poor seal, however, there should be no in-leakage of air unless there is complete separation of the two sections. The only disadvantage of a non-positive seal is that the inert purge gas cannot be collected and analyzed for radioactivity and/or moisture content.

The seal finally decided upon consisted of a tongue-and-groove type arrangement in which the groove was filled with sand. The groove ran around the periphery of the bottom section of the oven and the tongue likewise on the top section. There was sufficient clearance between the tongue and groove to allow for thermal expansion.

The lower section of the oven consisted of a carbon steel box lined with firebrick; there was a long, rectangular channel in the middle. A stainless steel liner (see Fig. 39) was inserted in this channel after flat-plate ceramic resistance heaters had been placed along the sides and bottom of the channel. This sheet metal liner was integral with the seal groove mentioned above. The liner had three argon inlets along one side and three thermocouples welded along the opposite side.

FIG. 39
LOWER SECTION OF OVEN WITH
STAINLESS STEEL LINER

The upper section of the oven consisted of an inverted stainless steel pan, the edges of which formed the tongue of the seal. A row of heaters was placed along the pan parallel to those of the bottom section, and then covered with firebrick. Figure 40 illustrates the entire oven after assembly in the cave.

The twenty heaters used in this test were divided into three banks, each controlled manually by a Variac. Each thermocouple was associated with temperature measurement at each of these banks. The temperatures were recorded on a Brown recorder and manual adjustments made with the Variacs to keep the temperatures close to 850°F. By this means the heat generation was kept uniform.

FIG. 40
ASSEMBLED OVEN PRIOR TO HEATING

Argon gas was used to blanket the fuel subassembly. The purge gas was metered at about one liter per minute and then flowed to each of the three inlets via a stainless tube header.

H. Conclusions

The EBWR fuel elements cannot be descaled by heating to 850°F due to side plate deformation at this elevated temperature. Other means must be found to remove this tenacious scale. At present, a test is underway to determine whether the scale can be removed by wet abrasive blasting of the fuel plates with an aluminum oxide slurry.

At 600°C, the growth of fuel plate No. 558 was 0.523% $\Delta V/V$, while that for plate No. 503 was zero. At 700°C, plate No. 528 grew 0.836% $\Delta V/V$ while plate No. 558 grew 7.107% $\Delta V/V$. It is clear from these data that swelling and distortion of fuel plates at the temperatures indicated are limited and reduced by a sound cladding.

It would be of interest to perform another series of similar tests on high burnup fuel plates. It should not be difficult to construct an oven for heating individual fuel plates in which one could guarantee no oxygen contamination. By virtue of being able to measure the three dimensions before and after heating, one cannot only calculate the changes in gross and local volumes at varying temperatures and burnup, but can also determine the degree of residual anisotropy in the fuel plates.

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