EBR-II EXPERIENCE WITH SODIUM CLEANING AND RADIOACTIVITY DECONTAMINATION*

W.E. RUTHER, C.R.F. SMITH
Argonne National Laboratory,
Argonne,
United States of America

INTRODUCTION

The EBR-II is now in its 13th year of operation. During that period more than 2400 subassemblies have been cleaned of sodium without a serious incident of any kind by a two-step process developed at Argonne. Sodium cleaning and decontamination of other reactor components has been performed only on the relatively few occasions in which a repair or replacement has been required. A summary of the EBR-II experience will be presented. A new facility will be described for the improved cleaning and maintenance of sodium-wetted primary components.

SUBASSEMBLY CLEANING

Subassemblies removed from the reactor typically retain from 5-40 g of sodium after draining and argon blowdown. They also may have up to 1.2 kW of residual heat generation. Sodium cleaning is performed in a massive interbuilding coffin (IBC) at a station between the reactor building and the air cell of the Hot Fuel Examination Facility (HFEF). Special provisions are made in the IBC (Fig. 1) to remove the heat generated by means of a recirculating argon cooling system which transfers the heat to the surrounding lead coffin. The IBC is self-powered and includes an emergency mercury flooding system which would operate in the event of failure of the argon pumping system. Thus far, this emergency system has never been activated.

The cleaning process is initiated by introducing humidified air (air bubbled through water) at a rate of 0.4 cfm to the recirculating argon coolant. The amount of water and air available for the oxidation of the sodium is increased gradually because the recirculating cooling argon (30 cfm) is removed at a rate of 0.4 cfm. The rate of reaction is monitored by a thermocouple in the reaction chamber. The temperature is maintained below 150°C by adjustment of the argon flow.

After 15 min of humidified air flow at 0.4 cfm, the flow of the humidified air is increased to 7 cfm with the same restriction on maximum process temperature. This is continued for 15 min.

At the end of this period, the flows of humidified air and recirculating argon are stopped, and the subassembly is rinsed by flowing purified water up through it at a rate of 5 gpm. Twenty-five gal of water are used to rinse the subassembly. The subassembly then is dried by passing dry air through it at a rate of 30 cfm for 60 min. The description above applies to the cleaning of driver subassemblies.

Experimental subassemblies are subjected to a similar operation except that humidified argon only (no air) is used. The 0.4-cfm flow of moist argon is maintained for 60 min and the 7-cfm flow is maintained for 30 min. The subassemblies are rinsed until samples of the effluent rinse have a conductivity less than 0.8 micro mhos/cm and a sodium content less than 0.4 ppm.

Our experience has shown that this process completely removes sodium and sodium products from smooth exposed surfaces but not from tight crevices. Initially there was some concern that the cleaning process might modify the stainless steel surface. Strain[1] studied the effects of sodium and sodium removal by this process on the microstructure of stainless steel tubing. He found no change that could be attributed to the sodium removal process.

---

*Work performed under the auspices of the U.S. Department of Energy, U. S. Government.

Pressurized tubes cycled through the reactor have also been examined after multiple sodium cleanings. Such an experiment,[2] entitled Creep I, has been exposed in-reactor about 22,000 hr at 400°C, receiving about 1×10^{23} nvt, and has been cleaned approximately 15 times with no discernible effect of the cleaning process. Initial hoop stress in the pressurized capsules of Creep I ranges from 10,000 to 35,000 psi.

Unless the experimenter requests otherwise, oxide fuel subassemblies with breached cladding are cleaned by the usual procedure. A breached carbide element in a test subassembly was given only the moist gas treatment without the water rinse. Since the HFEF facilities have been expanded, future breached carbide elements would probably now be transferred without washing to an inert-gas disassembly cell. The latter procedure is also used with breached metallic fuel. After disassembly, the elements are cleaned individually in ethanol in another inert cell.

PRIMARY PUMP

The largest EBR-II components that have been cleaned are the primary sodium pumps[3] (Fig. 2). Binding occurred of the drive shaft both pumps within the first six months of operation. They were removed and modified. Subsequently in 1970, nearly seven years later, pump No. 1 experienced periodic increases in power consumption with no corresponding changes in flow, indicating shaft binding and necessitating its removal from the primary tank again.

The pump was lifted from the primary tank into an inert atmosphere caisson and allowed to cool. Filtered air was slowly introduced to oxidize the surface of the sodium. The pump was then transferred to a plastic bag and gamma scanned. It was then disassembled.

These individual pump parts were cleaned of sodium in fabricated tanks on an isolated exterior pad. Each component, in turn, was placed in a tank, an argon atmosphere was established and the tank remotely filled with ethanol. Since both the component and the ethanol were very cold (December), the reaction proceeded at a leisurely pace. The alcohol was gradually diluted with water to 50-50 mixture and the component soaked until all reaction had ceased. The ethanol mixture was then drained and the pump parts were hot-water washed in the tanks. They were then lifted out and held for decontamination. This was accomplished in the same tanks using an agitated solution of Turco No. 4521 decon solution.

This relatively unsophisticated cleaning procedure worked well, but at one point during the disassembly a small sodium fire initiated in a sodium oxide-sodium mixture on the shaft baffle. It was quickly extinguished by flooding the area with argon gas. It has been our experience that sodium oxide-sodium mixtures spontaneously ignite in air much more readily than sodium.

SECONDARY COLD TRAP

A non-radioactive cold trap removed from the secondary sodium system about 10 years ago was cleaned and examined.[4] The cleaning consisted of the following major steps:

- The trap was mounted on a pivot so that it could be easily turned upside down or right side up. This facilitated draining the sodium and alcohol out of the vessel from the top.
- The sodium was heated using steam in the coil around the trap and it was then drained into a receiving tank.
- The sodium remaining in the trap was then reacted using ethanol. The alcohol was circulated and heated to 60°C. The first batch of alcohol (250 gallons) became saturated with sodium and alcohol reactant products and a second batch of clean alcohol (∿200 gallons) was used for the final cleanup. The only difficulty in this step was the reaction rate was too high on the first addition of alcohol and the vessel pressure

went to approximately 20 psig which exceeded the alcohol supply pump discharge pressure rating and stopped the flow of alcohol. After the pressure decreased only small batches of alcohol were added until the trap was filled with alcohol. From the amount of sodium reactant in the alcohol it was estimated that 90 pounds of sodium remained in the trap after draining and was reacted by the alcohol.

After the alcohol was circulated and heated to 60°C the cold trap was drained and purged with argon gas. The head was cut-off and the mesh removed. There was no evidence of sodium in the trap.

This operation was considered very satisfactory snd additional traps are intended to be cleaned in this manner. The complete reaction of the sodium verified the acceptability of the heated alcohol process.

SMALL COMPONENTS

In the past there has been no consistent procedure used for sodium removal. In general, ethanol has been used if the component was to be reused and might be damaged by a rapid reaction. For other situations superheated steam followed by a water rinse also has been used frequently.

NEW FACILITIES

EBR-II is upgrading their capabilities for sodium removal from components in order to facilitate component disassembly, inspection, repair, and reassembly. Improved capabilities will allow sodium removal activities to be conducted anytime of the year regardless of weather conditions, will allow for better control of the cleaning processes and will allow for controlled disposal of gaseous and liquid effluents.

Sodium removal will be conducted in the Sodium Components Maintenance Shop (SCMS) at EBR-II. This facility will provide for the removal of radioactive and nonradioactive sodium from large and small components. Sodium removal will be accomplished by the use of ethyl alcohol or by the use of steam and/or water.

Ethyl alcohol will be used on components that may be damaged by the use of steam or water and consequently impair the function of the component. Steam and/or water will be used on components where the function will not be impaired, on components that will not be reused, and for the disposal of bulk sodium.

The alcohol cleaning system will consist of two small component cleaning systems, one for radioactive components and one for nonradioactive components; a third cleaning system will be used for large components. The alcohol cleaning system will be housed in a metal building with the storage facilities located outside away from the building. The storage facilities will provide for the storage of fresh alcohol and the sodium-alcohol reaction products.

Installation of the small component system is presently underway and installation of the large component cleaning system is scheduled to begin in the spring of 1978. Both systems are scheduled to be operational by the summer of 1978.

The design of systems for sodium removal by steam and/or water is scheduled to begin in the spring of 1978 and to be operational by the fall of 1979. EBR-II also intends to provide an alcohol recovery facility which will provide for the concentrating and discharging of radioactive contaminates to drums for disposal and which will allow the recovery of the unreacted alcohol.

RADIOCHEMICAL HISTORY OF EBR-II SODIUM

The radiochemical history of EBR-II begins in July 16, 1964. Important factors that influence the production of radionuclides are (a) reactor power level, (b) plant capacity factor, and (c) neutron energy spectrum. Table I shows how the EBR-II power level and capacity factor have changed since 1964.

Reactor power reached design level (62.5 MWt) in 1970. The annual capacity factor varied from 20 to 60% until 1972 when efforts to improve the capacity factor resulted in a steady increase to 77% in 1976. In mid-1972 the depleted-uranium blanket surrounding the active core was replaced by a stainless steel reflector. This softened the neutron-energy spectrum and thereby increased the rate of formation of products of the (n,γ) reactions. All three of these factors have contributed to increasing radioactivity in the EBR-II primary system.

In addition, the increased number of experimental high temperature subassemblies has increased the amount of activated corrosion products in the primary system.

Table II lists the longer-lived radionuclides found in the primary sodium. (Radionuclides of the noble gases, found principally in the cover gas, are omitted). Analysis for $^{239}\mathrm{Pu}$ in sodium is also made monthly but, to date, none has been found (to the limit 8 mBq/gram Na). Of the radionuclides listed, $^{3}\mathrm{H}$, $^{54}\mathrm{Mn}$, $^{125}\mathrm{Sb}$, $^{131}\mathrm{I}$, and $^{137}\mathrm{Cs}$ exhibit reversible, temperature-dependent segregation.[5] We observe that if the temperature of the primary sodium is reduced to 177°C (350°F) (The conventional temperature when components are to be removed from the primary tank) these nuclides tend to leave the bulk sodium and concentrate at all surfaces exposed to sodium.

Of the radionuclides listed in Table II, fifteen-hour, sodium-24 is overwhelmingly the dominant radioactivity. At equilibrium its activity is a factor of 1×10^{4} greater than $^{137}\mathrm{Cs}$, the next most radioactive nuclide. Its principal effect on operations is to necessitate adequate shielding around primary sodium components and to delay contact maintenance. For example, it is customary to delay five days before removing samples from the primary sodium sampler for analysis; entry to the purification vault (the location of the cold trap) is not permitted until 10 days following isolation of the cold trap or shutdown of the reactor.

Thirty-year cesium-137 is the second greatest radioactivity in the primary sodium. Cesium-137, a fission product, enters the sodium from the fuel following

TABLE I

Power Level and Capacity Factor

	Reactor Power Level (MWt)	Capacity Factor (%)
7/29/64 - 8/5/64	10	
8/5/64 - 8/13/64	20	
8/13/64 - 10/13/64	30	
10/13/64 - 3/27/65	37.5	
3/27/65 - 8/26/68	45	
1965		26.4
1966		43.0
1967		20.1
8/26/68 - 9/25/70	50	
1968		41.8
1969		42.4
9/25/70 - Date	62.5	
1970		57.9
1971		39.1
6/30.72	Operation with a new stainless steel reflector in core	
1972		46.9
1973		49.9
1974		58.7
1975		66.1
1976		76.9

---

[a] Factors are calculated in accordance with the Glossary of Important Power and Rate Terms, dated 1965, promulgated by the Federal Power Commission and prepared by the Ad Hoc Committee of the Inter-Agency Committee on Water Resources.

TABLE II. Longer-Lived Radionuclides Found in the EBR-II Primary System

Radio-Nuclide	Source	Produced By	t 1/2	Where Identified	Activity Per Gram of Sodium	
					1970	1977
^{3}H	Fuel; Control Rods	Fission; $^{10}B(n,2\alpha)^{3}H$	12.3 y	Bulk Na; cover-gas	3.0 kBq	2.7 kBq
^{22}Na	Sodium	$^{23}Na(n,2n)^{22}Na$	2.6 y	Bulk Na; cold trap; cover-gas surfaces; in-vessel components; seal alloy of rotating plugs	1.8 kBq	4.1 kBq
^{24}Na	Sodium	$^{23}Na(n,\gamma)^{24}Na$	15 h	Bulk Na; in-vessel components	52 MBq	99 MBq
^{51}Cr	Stainless Steel	$^{50}Cr(n,\gamma)^{51}Cr$	27.7 d	Plugging valve[a], deposition samples, in-vessel components		
^{54}Mn	Stainless Steel	$^{54}Fe(n,p)^{54}Mn$	313 d	Bulk Na; cover-gas surfaces; deposition samples; in-vessel components	1.7 Bq	7.9 Bq
^{58}Co	Stainless Steel	$^{58}Ni(n,p)^{58}Co$	71 d	Plugging valve; SURV-3 subassembly; deposition samples		
^{60}Co	Stainless Steel Haynes Stellite	$^{59}Co(n,\gamma)^{60}Co$	5.27 y	Cold trap; cover-gas surfaces; plugging valve;[a] deposition samples; in-vessel components		
^{65}Zn	Unknown	$^{64}Zn(n,\gamma)^{65}Zn$	244 d	Cold trap; plugging valve[a]		
^{110m}Ag	Unknown	$^{109}Ag(n,\gamma)^{110m}Ag$	252 d	Bulk Na	20 Bq	94 Bq
^{113}Sn	Seal Alloy	$^{112}Sn(n,\gamma)^{113}Sn$	115 d	Bulk Na	0.20 kBq	1.1 kBq
^{117m}Sn	Seal Alloy	$^{116}Sn(n,\gamma)^{117m}Sn$	14 d	Bulk Na	0.16 kBq	0.56 kBq
^{134}Sb	Stainless Steel	$^{123}Sb(n,\gamma)^{124}Sb$	60 d	Cold trap	0.44 na	
^{125}Sb	Seal Alloy	$^{124}Sn(n,\gamma)^{125}Sn(-\beta^{-})^{125}Sb$	2.73 y	Bulk Na; cold trap	16 Bq	27 Bq

Radio-Nuclide	Source	Produced By	t 1/2	Where Identified	Activity Per Gram of Sodium	
					1970	1977
^{131}I	Fuel	Fission	8.0 h	Bulk Na	(c)	(c)
^{134}Ca	Fuel	Fission; $^{133}Cs(n,\gamma)^{134}Cs$	2.06 y	Cold trap; deposition samples; in-vessel components		
^{137}Cs	Fuel	Fission	30.1 y	Bulk Na; cold trap; cover-gas surfaces; deposition samples in-vessel components; seal alloy of rotating plugs	0.44 kBq	10 kBq
^{210}Po	Seal Alloy	$^{209}Bi(n,\gamma)^{210}Bi(-\beta^-)^{210}Po$	138 d	Bulk Na; pump impeller	0.9 Bq	3.5 Bq
^{182}Ta	Cladding of neutron source; SURV subassemblies	$^{181}Ta(n,\gamma)^{182}Ta$	115 d	Pump; plugging valve[a]; gage subassembly; cold trap		
^{132}Te	Fuel	Fission	78 h	Bulk Na		
^{140}Ba	Fuel	Fission	12.8 d	Plugging valve[a]		
^{89}Sr	Fuel	Fission	50 d	Inlet valve in purification system		
^{90}Sr	Fuel	Fission	29 y	Inlet valve in purification system		
^{59}Fe	Stainless Steel	$^{58}Fe(n,\gamma)^{59}Fe$	45 d	YY02		
^{233}Pa	welding rod	$^{232}Th(n,\gamma)^{233}Th(-\beta^-)^{232}Pa$	27 d	Gage subassembly		

[a] Co-deposited with copper in plugging valve, September 1968.

[b] Examples of in-vessel components are: YY03 experimental subassembly, SURV subassemblies, core gripper liner, gage subassembly, INCOT thimble.

[c] Base level is < 2 Bq/gram Na; following fission product releases, maximum ^{131}I activity was 37 Bq/gram in 1970 and 368 Bq/gram in 1977.

[d] Average valve when bulk sodium is at 371°C (700°F).

[e] Average valve in 1971.

a breach in the fuel cladding. This nuclide was first detected in EBR-II sodium in May 1967, but the principal increases in cesium-137 activity began after May 1976 when the reactor was operated with high-burnup, breached-fuel in-core. Figure 3 shows the activity of cesium-137 in sodium since 1967. It can be seen that, following the major increases in cesium-137 activity there is usually a small remission. We attribute this to removal of the cesium by the cold trap and by condensation of the cesium on surfaces in the cover-gas region.

The third most radioactive nuclide is 2.6-year sodium-22. This nuclide is formed by an energetic (n,2n) reaction, and, therefore, can only be formed in-core. The increased activity of sodium-22 from 1970 to 1977 is attributed to its continuing approach to saturation and to the increasing capacity factor of EBR-II (Table I).

The fourth most radioactive nuclide is 12.3-year tritium. Tritium is a low energy (0.0186 MeV) beta-emitter. As such it is easily shielded and becomes an insignificant source of radiation unless it enters the body as tritiated water (HTO or T_2O). In this form it is considered an important radiological hazard. The radiological concentration guide for tritium in air is 207 Bq/L of air (5.6×10^{-6} µCi/mL of air). Prior to September 1972, analyses for tritium in EBR-II sodium showed considerable scatter, and are not considered reliable. Subsequent analyses have indicated an approximately constant activity of 2.7 ± 0.6 kBq/ gram for tritium in primary sodium. This indicates that the sources for tritium (fusion and ^{10}B (n, 2α) ^{3}H reactions) are in approximate balance with sinks for tritium.

RADIONUCLIDES ON COMPONENTS

Other than ^{22}Na which pervades the entire primary system, the principal radionuclides found on components in the primary system at EBR-II are ^{54}Mn, ^{60}Co, ^{137}Cs, and ^{182}Ta. All of these nuclides are formed (or released) in-core, then transported by the sodium and deposited on the surfaces of various components. In addition, ^{60}Co is formed by (n,γ) reactions in stainless steel and in other cobalt-bearing material in the primary tank. Cobalt-60 that is deposited on surfaces from sodium is not completely removed by our non-destructive cleaning techniques. To distinguish between ^{60}Co that is deposited on surfaces and ^{60}Co that is formed in situ by neutron activation, it is necessary to measure its activity after removing the surface layer by etching or by electropolishing.

Like ^{22}Na, ^{137}Cs pervades the entire primary system. However, because of its higher vapor pressure, cesium preferentially deposits on surfaces in the cover-gas region. Deposits of sodium taken from the cover-gas region show $^{137}Cs/^{22}Na$ ratios up to 200 times the $^{137}Cs/^{22}Na$ ratio found in 700°F primary sodium.[3] The average ratio is approximately 40.

Both ^{54}Mn and ^{137}Cs segregate from the bulk sodium to surfaces in contact with the sodium when the sodium is cooled. For example, in December 1970 to February 1971, the primary sodium in EBR-II was cooled from 371°C to less than 177°C. During this period, the activity of ^{54}Mn decreased from 1.8 Bq/g to < 0.7 Bq/g and the activity of ^{137}Cs decreased from 1.4 kBq/g to 0.1 kBq/g. This means that components removed from primary sodium following a cool-down of the sodium will have a higher burden as ^{54}Mn and ^{137}Cs (and any other segregating species) than they would have had at operating temperature. Another example of ^{137}Cs segregation is the radiometric examination of the pump duct of the fuel-element rupture detector loop.[6] This duct was heated (temperature unspecified) and drained of sodium twice before examination. No ^{137}Cs was found; evidently it had left the surface with the heated sodium.

The presence of ^{182}Ta on the surfaces of some components in the EBR-II primary system is probably not typical of most other LMFBR's. The sources for ^{182}Ta in EBR-II are the tantalum cladding of the ^{124}Sb neutron source and the tantalum specimens in the materials-irradiation (SURV) subassemblies. The tantalum-182 is formed by the reaction ^{181}Ta (n,γ) ^{182}Ta. It has been found on the surface of the pump-case-manifold subassembly and on the shroud of the SURV-3 subassembly.

RADIOCHEMICAL RESIDUE AFTER CLEANING

1. Irradiated-Fuel Monitor Thimble

This thimble entered the primary tank about 11 feet from the center-line of the reactor core. The lower 3m of the thimble were submerged in 371°C liquid sodium. Above that, about 46 cm of the length were exposed to the cover gas before the thimble entered the shield plug. The thimble was made from series 18-8, type 304 stainless steel. It had been installed 1.8 years before its removal in November 1970.

Using a GeLi detector, axial gamma scans of the lower 3.8 meters of the thimble were made before and after steam cleaning.[6] Besides ^{22}Na, which was completely removed by steam cleaning, the principal radioisotopes found on the thimble were ^{54}Mn, ^{60}Co, and ^{137}Cs. The activities of ^{54}Mn and of ^{137}Cs on the uncleaned thimble were roughly constant over the sodium-immersed region, but sharply lower in the cover-gas region. The activity of ^{60}Co, on the other hand, decreased roughly exponentially from bottom to top. The results of steam cleaning are given in Table III.

TABLE III. Decontamination of the Irradiated-Fuel Monitor-Thimble by Steam Cleaning

	Percent of Radioactivity Removed by Steam Cleaning	
Nuclide	From Surface Exposed to Cover Gas	From Surface Immersed in Liquid Sodium
^{54}Mn	31	23
^{60}Co	51	-1.6
^{137}Cs	90	93

Steam cleaning removed about 90% of the ^{137}Cs-activity from stainless steel surfaces of the irradiated-fuel-monitor thimble exposed to either liquid sodium or to argon cover gas. It is less effective in removing ^{54}Mn and ^{60}Co. The mechanism by which ^{54}Mn reaches surfaces in the cover-gas region is not known; possibly it diffuses upward on surfaces wetted by sodium. Steam cleaning was 50% effective in removing ^{60}Co-activity from surfaces exposed to cover gas but ineffective in removing ^{60}Co from surfaces immersed in bulk sodium. We did not attempt to determine the proportion of ^{60}Co activity formed in situ by (n,γ) reactions and that formed by transport-deposition.

2. FERD Loop Pump Duct

The fuel-element-rupture-detector (FERD) loop is a system for detecting ruptured fuel by sensing the emission of delayed neutrons. A continuous stream of primary sodium is removed near the outlet of the intermediate heat exchanger (IHX) and is pumped through a loop past a series of delayed-neutron detectors and then returned to the primary tank. The FERD loop pump is a d.c., conduction-type, electromagnetic pump that delivers 371°C sodium at 6.3 L/sec (100 gpm) at an estimated head of 30 k pascal (4.3 psi). The pump duct examined here was made of series 18.8, type 304 stainless steel. It was installed in 1964 and removed in April 1970 after 35,000 hrs of operation. Except for those surfaces from the active zone of the core to the outlet of the IHX, the FERD loop is exposed to the most radioactive sodium in the primary system. The pump duct is sufficiently removed from the reactor core so that it is not exposed to neutrons other than a very low source from delayed-neutron precursors. In effect, the only radioactivity found at the pump duct results from transport-deposition.

The FERD loop pump duct was twice heated and drained of sodium before an axial gamma scan was made with a GeLi detector.[6] The only radionuclides found on the duct were ^{54}Mn and ^{60}Co. The ^{137}Cs activity had evidently gone with the sodium during the hot-draining. The interior of the duct was then washed with ethyl alcohol and water until no reactions or reaction products were seen in the wash solution, and the gamma scan was repeated. The alcohol-water washes removed 28% of the ^{54}Mn and 52% of the ^{60}Co activities.

3. Hexagonal Shroud of the SURV-3 Subassembly[6]

The SURV subassemblies are a set of 10 materials irradiation subassemblies that were loaded into the reactor core and storage basket in March 1965. These subassemblies contain 15 materials used in the construction of the primary system. The SURV-3 subassembly was removed from its reactor position in February 1971 after 2188 days in-core at an estimated fluence of 1×10^{22} n/cm^2.

The subassembly was removed from the reactor building to the Hot Fuels Examination facility (HFEF) in a transfer cask. Moist argon was used to convert residual sodium to sodium hydroxide. While still in the cask the subassembly was repeatedly washed with 20L (5 gal.) batches of water until a total of about 100L (25 gal.) of water were used. The subassembly was then transferred to the HFEF air cell for remote disassembly. One-inch square (6.45 cm^2) samples were cut from the hexagonal shroud of the subassembly for further examination. (Both the inner and other surfaces of the shroud were exposed to flowing sodium during irradiation).

The principal radioactivities found in the first 40L (10 gal.) of wash water were:

γ activity: ^{182}Ta = 18.5 MBq/L (0.5 μCi/mL)

α activity: unknown = 41 MBq/L (1.1 μCi/mL)

No uranium was found (to a limit of < 0.1 mg/L).

Since it was impractical to examine the SURV-3 subassembly by a direct gamma scan, there was no way to measure the effectiveness of the cleaning procedure. However, the samples of hexagonal shroud were examined for radioactivity in the Argonne West Junior Caves with the following results:

Radioactivity Removed by 50 mL of Rinse Water

	kBq/L
^{54}Mn	53
^{58}Co	6.7
^{60}Co	16
^{137}Cs	530

Subsequently the specimen was electropolished in 10 stages, the radioactivity in the electrolyte from each stage was analyzed. The results showed that all ^{132}Ta was removed in the first stage [depth removed from one side, 13 μm (0.0005 in.)] but that ^{54}Mn, ^{58}Co, and ^{60}Co were uniformly distributed through in the electrolytes from all 10 stages. An estimated depth of 127 μm (0.005 in.) was removed from each side.

The conclusions following from these tests are that ^{154}Mn, ^{58}Co, ^{60}Co, ^{137}Cs, and ^{182}Ta are all found on the surfaces of the hexagonal shroud, exposed to flowing sodium. Washing the shroud with water removes all of the ^{137}Cs and part of the ^{54}Mn, ^{58}Co, ^{60}Co, and ^{182}Ta. A light electrolytic etch of the shroud removes all of the ^{182}Ta and probably all of the ^{54}Mn, ^{58}Co, ^{60}Co remaining on the surface. Successive electrolytic etches of the shroud show that it is uniformly permeated with ^{54}Mn, ^{58}Co, and ^{60}Co formed from in situ, neutron reactions with ^{54}Fe, ^{58}Ni, and ^{59}Co respectively.

4. Primary Pump No. 1

As described in an earlier section, Primary Pump No. 1 was removed in December 1970. On January 6, 1971 (52 days after reactor shutdown) an axial scan of the radiation from the pump was made with a survey meter; samples of deposits were taken from various surfaces for analyses; an axial gamma-scan was made of the pump with a GeLi detector. Figure 4 shows the pump assembly (shield plug in-place), the location of various samples taken for analysis (encircled numbers), and the results of the axial scan using the GeLi detector. The results of this study are reported in detail in Reference 6. The most notable discoveries are summarized below:

a. The principal radioisotopes found in the deposits sampled were: ^{22}Na, ^{54}Mn, and ^{137}Cs. Polonium-210 was found on sample #27 scraped from the pump impeller.

b. Ratios of ^{137}Cs/^{22}Na were measured and compared with the same ratio found in the primary sodium before reactor shutdown and sodium cooldown. These comparative ratios found in the cover-gas region (samples 1-24) ranged from 0.88 to 197 and averaged 46. This shows how the higher vapor pressure of cesium tends to concentrate ^{137}Cs in the cover gas region.

c. Similarly, the ratio of ^{137}Cs/^{22}Na on surface deposits (samples 3, 25-27) that has been under sodium when compared with the same ratio in primary sodium before reactor shutdown ranged from 26-31 and averaged 31. This shows the effect of the segregation of ^{137}Cs from the primary sodium as the sodium was cooled from 371°F (700°F) to 177°F (350°F).

d. Surprisingly, a negligible increase in ^{60}Co activity was found in the vicinity of the stellite bearing of the pump (at 10 ft elevation in Figure 3). Stellite is an alloy containing 64% cobalt.

After the gamma-scan of the uncleaned pump, the pump was disassembled and an axial scan was for γ activities was made on each major component prior to decontamination.

Decontamination of the components consisted in immersing them in a series of solutions as described below. Each solution was agitated by flowing argon gas through it.

a. Each component was immersed in 95% ethyl alcohol. Gradually this solution was diluted to 50% alcohol by adding water.

b. Each component was rinsed in hot water.

c. Each component was soaked for four hours in 65°C (150°F) solution of Turco Decon 4521* (20 g/L or 2.7 oz/gal) and ammonium citrate (9.6 g/L or 1.3 oz/gal). Exposed surfaces of the components were scrubbed during this time. The bath was allowed to cool for 12 hours to 4°C (40°F).

d. Each component was washed with hot water, rinsed with commercial (95%) alcohol, and drip-dried in air.

e. Each component was wrapped in polyethylene sheet and again scanned with the GeLi detector.

Comparison of the gamma-scans before and after decontamination are given in Table IV.

---

*Turco Decon 4521 is a proprietary product of Turco Products, Inc., P.O. Box 1055, Wilmington, Ca. It is an inhibited acid containing oxalate, citrate, and ammonium ions.

Percent Radioactivity Removed by Pump-Cleaning-Procedure

	Shaft-Impeller Subassembly	Pump-Case Manifold Subassembly	Baffle Subassembly
^{54}Mn	53	44	67
^{60}Co	92	42	88
^{137}Cs		65	
^{182}Ta		negl.	

The decontamination procedure used here appears to be more effective for removing ^{54}Mn and ^{60}Co than the steam-cleaning procedure used with the irradiated-fuel-monitor thimble or the alcohol-water procedure used for the FERD-loop pump duct. Better effectiveness should be expected for removing ^{137}Cs and ^{182}Ta.

5. Baffle Subassembly Capscrews

Forty-eight capscrews that joined the baffle subassembly to the case-manifold subassembly and to the shield plug were replaced during the reassembly of the primary pump. Twenty-four of the capscrews had been below the level of the primary sodium; twenty-four had been in the cover-gas region. These screws, of type 316 stainless steel were hex-head, 1-in. diameter x 3 3/4-in. or 3 1/4-in. length. For convenience, the screws submerged in sodium are hereafter called Series-A screws and those from the cover-gas region are called Series-B screws. All of the screws had been in use for seven years. Only the heads of the series-A screws were exposed to flowing sodium. Only the heads of Series-B screws were freely exposed to argon cover gas containing sodium vapor, vapor condensate, sodium oxide and volatile ^{137}Cs. The shanks of both series were relatively shielded from their respective environments. The estimated neutron flux at the capscrews (fast and thermal) was less than 10^5 n/cm^2-sec.

Radiation surveys of the capscrews showed significant differences between the two series and also between the exposed heads and the shielded shanks. This observation led to a detailed examination which is described fully in Reference 6. The capscrews had already been cleaned with an alcohol-water wash. This precluded a measurement of the decontamination factor. Also, unexposed specimens of the capscrews were not available for comparative examination. Results of this examination relating to decontamination are summarized below:

a. The hex-head of the capscrew was rough-surfaced relative to the machined shank. Manganese-54 and ^{137}Cs tend to accumulate on these rough surfaces.

b. Series-A capscrews were about five times as radioactive as Series-B capscrews (following alcohol-water wash); this is partly due to neutron activation (in-sodium material being closer to the neutron source), partly due to deposition of radionuclides from sodium.

c. The (exposed) heads of the capscrews were typically 10-50 times as radioactive as the (unexposed) shanks. This indicates radioactivity from transport-deposition was many times as significant as radioactivity from neutron activation in the region of the capscrews.

d. Activities found on Series-A capscrews were ^{54}Mn (major)* and ^{60}Co (minor), ^{137}Cs (minor). Activities found on Series-B capscrews were ^{137}Cs (major), ^{60}Co (minor), ^{58}Co (trace).

---

*The strong ^{54}Mn-activity may have obscured a weak ^{58}Co-activity in Series-A capscrews.

e. A series of acid washes of Series-B capscrews (50% CH_3COOH, IN HNO_3, 16N HNO_3, 6N HCL) removed 93% of the ^{58}Co, 18% of the ^{60}Co, and 88% of the ^{137}Cs activities. Most of the ^{60}Co activity remaining (following the initial alcohol-water wash) was formed in situ by (n,γ) reactions whereas the ^{54}Mn and ^{58}Co [(n,p) reactions] were somehow transported into the cover-gas region. (It is surprising that 12% of the ^{137}Cs activity remained on a Series-B capscrew, even after repeated etches with strong mineral acids).

REFERENCES

1. R. V. Strain, "Effects of Sodium and Sodium Removal on the microstructure of type 304L Stainless Steel Tubing," ANL/EBR OOG, August 1969.

2. L. C. Walters, Personal Communication.

3. B. C. Cerutti et al, "Removal and Repair of EBR-II Primary Sodium Pump No. 1," ANL-7835, May 1972.

4. J. D. Leman, Personal Communication.

5. C. R. F. Smith and J. T. Holmes, "Purity of EBR-II Sodium: 1967-1974," ANL 75-26, December 1975.

6. C. R. F. Smith, "Radioactivity of EBR-II Components: I Examination in FY 1971, ANL 7873 September 1972.

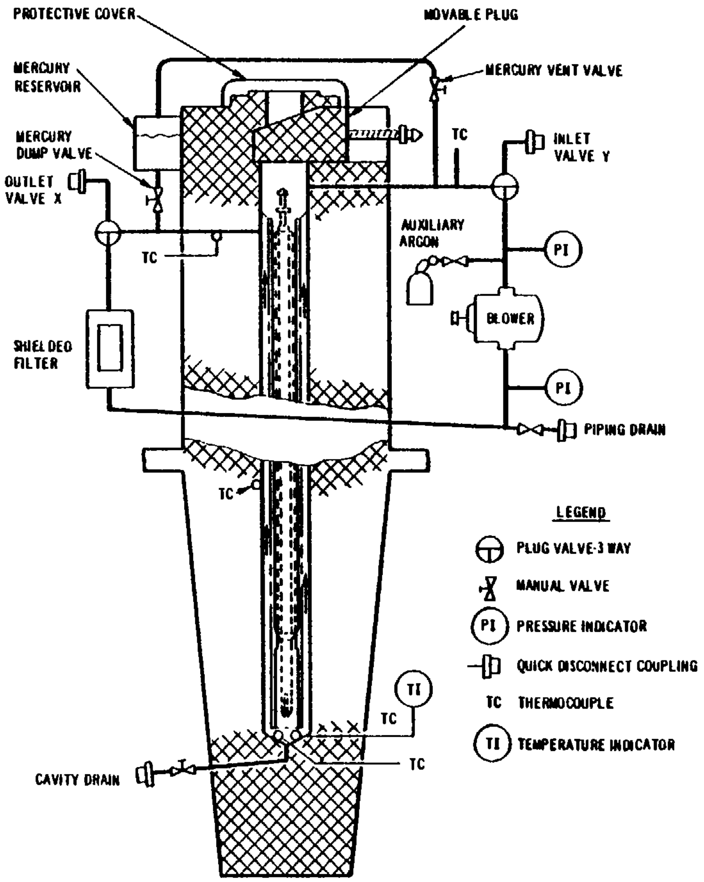


Figure 1. Interbuilding Coffin Recirculating Cooling System Flow Diagram

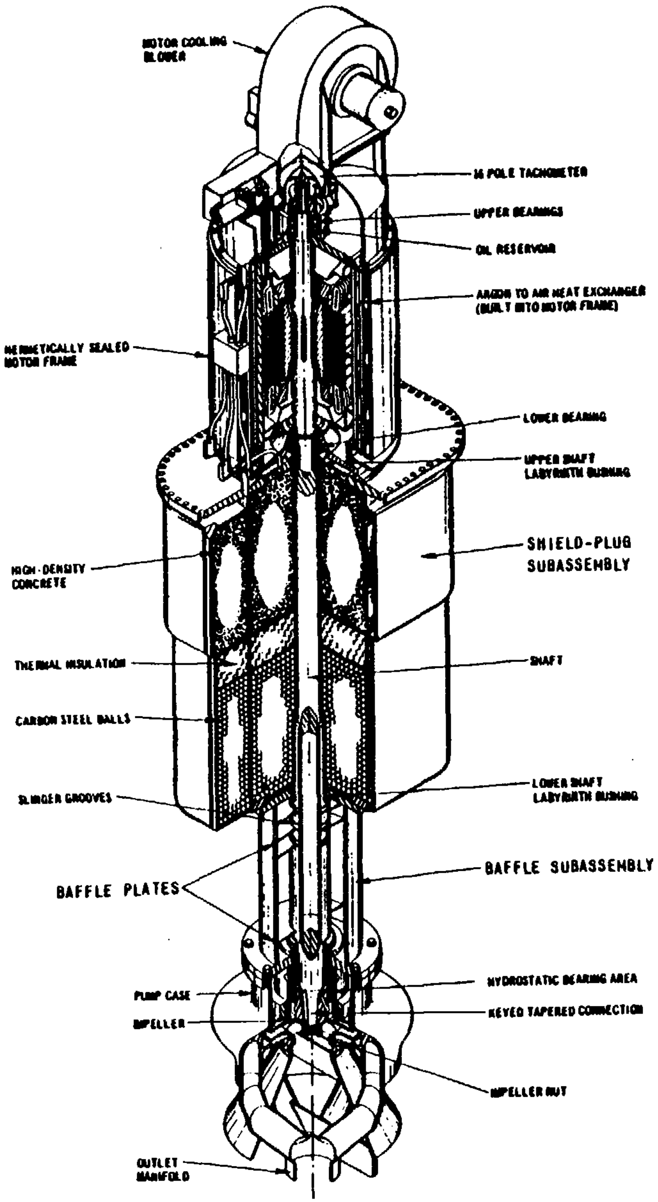


Fig. 2 Cut Away View of the EBR II Primary Pump No. 1.

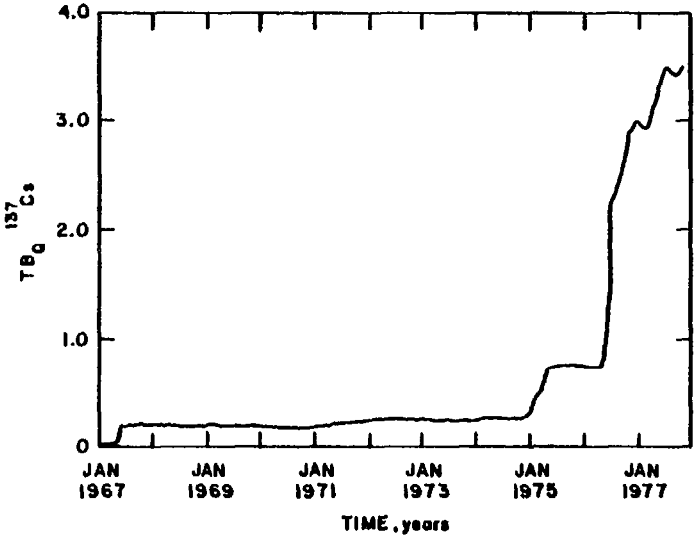
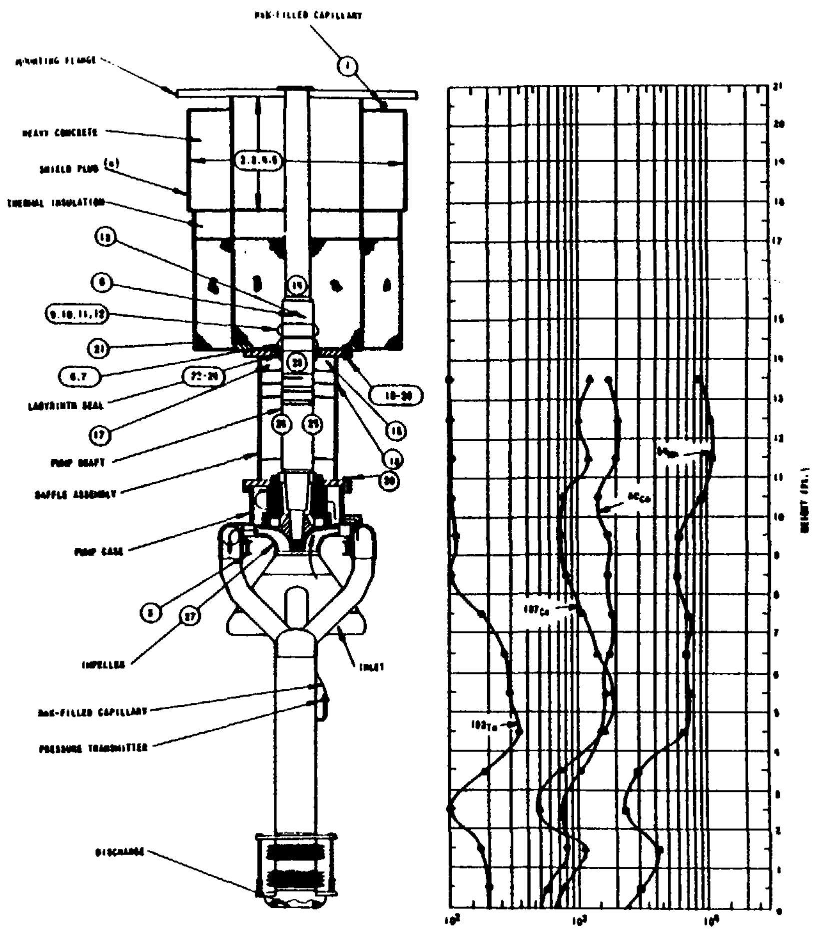


Fig. 3 Activity of Cesium---137 in EBR II Primary Sodium



(a) SHIELD PLUG SUBASSEMBLY WAS REMOVED PRIOR TO SCAN OF RADIOISOTOPE ACTIVITY

SECTION OF EBR-II PRIMARY PUMP SHOWING:

(A) LOCATION OF DEPOSITS SAMPLED FROM PUMP ASSEMBLY SURFACES

(B) RADIOISOTOPE ACTIVITIES MEASURED WITH GeLi DETECTOR (44" FROM PUMP CENTERLINE)

Fig. 4 Radioactivity Scan of EBR II Primary Pump No. 1

AN EXPERIENCE OF CLEANING AND DECONTAMINATION OF THE BN-350 REACTOR COMPONENTS*

K.T. VASILENKO, L.A. KOCHETKOV, V.M. ARKHIPOV,
R.P. BAKLUSHIN, A.I. GORLOV, G.V. KISELEV,
P.S. REZINKIN, A.A. SAMARKIN, N.D. TVERDOVSKY
Union of Soviet Socialist Republics

Abstract

In the course of start-up, adjustment and operation of the BN-350 reactor there arose a need for cleaning from sodium and decomntamination of primary and secondary equipment components. Design schemes of the systems provided for this purpose as well as those specially designed for cleaning of steam generator evaporators are considered. Technological processes of cleaning and decontamination for some reactor components (removable parts of circulating pumps, evaporators, valves) are described, the results are presented.

I. Introduction.

The sodium-cooled fast reactor BN-350 was put into operation on June 16, 1973. In the course of start-up and adjustment work and initial power operation a need for cleaning of components from sodium residues has repeatedly arisen. Some-

---

* The report was distributed at the meeting and was not presented and discussed.