EBR-II OPERATING EXPERIENCE

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ABSTRACT

As originally designed and operated, EBR-II successfully demonstrated the concept of a sodium-cooled fast breeder power plant with a closed fuel reprocessing cycle (mini-nuclear park). Subsequent operation has been as an irradiation facility, a role which will continue into the foreseeable future.

Since the beginning of operation in 1961, operating experience of EBR-II has been very satisfactory. Most of the components and systems have performed well. In particular, the mechanical performance of heat-removal systems has been excellent. A review of the operating experience reveals that all the original design objectives have been successfully demonstrated.

To date, no failures or incidents resulting in serious in-core or out-of-core consequences have occurred. No water-to-sodium leaks have been detected over the life of the plant. At the present time, the facility is operating very well and continuously except for short shutdowns required by maintenance, refueling, modification, and minor repair. A plant factor of 76.9% was achieved for the calendar year 1976.
INTRODUCTION

EBR-II, the nation's only operating fast breeder reactor, is located at the Idaho National Engineering Laboratory, 35 miles west of Idaho Falls, Idaho. EBR-II forms the hub of a complex that is exclusively dedicated to research and development activities under the national Fast Breeder Reactor (FBR) program.

EBR-II and other LMFBR-related facilities at the Argonne-West site on the INEL are operated by Argonne National Laboratory for the United States Energy Research and Development Administration under a contract between ERDA Chicago Operations Office, the Argonne Universities Association, and the University of Chicago. Designed and constructed with the technologies of the 1950's, EBR-II has served and continues to serve as the nation's principal facility for testing LMFBR fuels, materials, and components under conditions approaching those expected for more advanced LMFBR systems.

The purpose of the design and early operation of the EBR-II was to demonstrate the feasibility of a sodium-cooled fast breeder reactor as an integrated plant and fuel-processing facility. In fulfilling the initial goals, the reactor operated successfully up to and including full power for a number of years. More recently, as the only operating sodium-cooled fast breeder reactor in the United States, EBR-II was changed from the role of a demonstration plant to the role of a test and fuels irradiation facility. This role will continue into the foreseeable future. Materials, fuels, and sensors are now being irradiated to elevated temperatures and in a fast flux environment within the reactor. Test regions and cells have been added externally to the reactor to test components, ion chambers, and sensors in the radioactive primary coolant system.

Since the initial approach to power, steps have been taken to increase the plant availability and usefulness of EBR-II. Many plant modifications have been made to the original facility. Some of the most important changes that have been implemented are: (1) shortening of the driver-fuel pin, increasing the enrichment of the driver-fuel material and the replacement of depleted uranium in the upper and lower axial reflectors with stainless
steel; (2) reduction in the number of fueled control rods to eight by the introduction of high-worth control rods to allow installation of special irradiation facilities; (3) enlargement of the core from a 61-subassembly to a 91-subassembly configuration, and finally, to a 127-subassembly configuration; (4) simplification of the plant protection system; (5) drilling of inspection and cleaning holes for the rotation plug seals; (6) upgrading of the fuel-handling system; (7) replacement of the depleted uranium radial blanket with a stainless steel reflector to improve the irradiation capabilities of the core; and (8) installation of water-to-sodium leak detectors in the steam generators. In addition, the burnup limit for the core driver fuel has been raised significantly above its initial conservative level, and the length of reactor runs has been increased to 2700 MW-days to obtain maximum utilization of the fuel.

EBR-II's 13 years of successful power operation have demonstrated that:

- Operating and maintenance personnel can work routinely and safely in close proximity to a sodium-cooled reactor and its ancillary systems.

- A 13-year-old LMFBR power plant can operate with a respectable capacity factor, e.g., 76.9% for 1976. In the absence of downtime needed for experiment handling, capacity factors larger than 80% appear reasonable.

- Components can be routinely removed from sodium systems, cleaned, repaired, and returned.

- Sodium-to-water steam generators can be built and operated safely.

- The consequences of failures in prototypal LMFBR fuel elements are benign.

- An LMFBR such as EBR-II is characterized by extremely stable operation.

- The release of radioactive species from a sodium-cooled system is limited primarily to rare-gas fission products.
II. HISTORY

The initial success of EBR-I in the early 1950's encouraged more visionary plans for the commercial exploitation of the fast-breeding principle. In response to a request from the USAEC, ANL submitted in 1953 a proposal for a second and larger sodium-cooled fast reactor. This system, now known as EBR-II, was conceived as a pilot plant and as an intermediate step between EBR-I and a full scale commercial fast breeder.

The objectives underlying the design of EBR-II were many. Included was the need for higher power densities to evaluate the performance of fuel, cladding, and structural materials under conditions more hostile than those encountered in EBR-I. Vitaliy needed for the design of more advance systems was performance information on items such as pumps, heat exchangers, steam generators, valves, flow meters, control rod drives, seals, etc. EBR-II was, accordingly, designed to fulfill these needs in addition to demonstrating the feasibility of metal-fueled fast reactors for central power plant use.

As a final objective, the subject of on-site pyrometallurgical fuel reprocessing was addressed. Because of the low burnup expected for metal fuels at that time (approximately 1-2%) elementary economics dictated a short turnaround time for reprocessing. Accordingly, a contiguous and integrated facility for reprocessing spent fuel was included in the formal proposal.

Site preparation began in October 1957, and formal construction started in December 1957.

Criticality was reached on November 11, 1963, with a fuel loading of 181.2 kg of $^{235}$U. The approach to power was begun in July 1963, and a power level of 37.5 MWt was reached on October 13, 1964. Power was raised to 45 MWt in March 1964, to 50 MWt in August 1968, and to the full-power rating of 62.5 MWt in September 1969.

In May 1965, EBR-II began its role as an irradiation test facility with the insertion of the first experimental subassembly. Shortly after this time, an ambitious program was launched to irradiate and test a large number of fuels and structural materials.
A chronology of milestones in the construction and operation of EBR-II is presented in Table I.

**Table I. Chronology of Operation of EBR-II**

<table>
<thead>
<tr>
<th>Date</th>
<th>Event Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>July 11, 1955</td>
<td>Original authorization (Public Law 84-141)</td>
</tr>
<tr>
<td>November 15, 1956</td>
<td>Award of architect-engineer contract; start of construction design</td>
</tr>
<tr>
<td>August 21, 1957</td>
<td>Revised authorization to increase scope and funding (Public Law 85-162)</td>
</tr>
<tr>
<td>November 8, 1957</td>
<td>Award of first major construction contract for Reactor Containment Building.</td>
</tr>
<tr>
<td>September 30, 1961</td>
<td>Reactor made &quot;dry critical&quot; without sodium (critical mass, 230.16 kg of $^{235}$U)</td>
</tr>
<tr>
<td>January 29, 1963</td>
<td>Sodium filling of secondary system started</td>
</tr>
<tr>
<td>February 18, 1963</td>
<td>Sodium filling of primary system started</td>
</tr>
<tr>
<td>February 26, 1963</td>
<td>Sodium filling completed (10 railroad tank cars)</td>
</tr>
<tr>
<td>November 11, 1963</td>
<td>Reactor made &quot;wet critical&quot; with sodium (critical mass, 181.2 kg of $^{235}$U)</td>
</tr>
<tr>
<td>April 9, 1964</td>
<td>Combined operation of primary and secondary sodium systems achieved</td>
</tr>
<tr>
<td>July 16, 1964</td>
<td>&quot;Approach to Power&quot; started</td>
</tr>
<tr>
<td>July 29, 1964</td>
<td>Reactor power raised to 10 MWt</td>
</tr>
<tr>
<td>August 5, 1964</td>
<td>Reactor power raised to 20 MWt</td>
</tr>
<tr>
<td>August 13, 1964</td>
<td>Reactor power level raised to 30 MWt</td>
</tr>
<tr>
<td>October 13, 1964</td>
<td>Reactor power level raised to 37.5 MWt</td>
</tr>
<tr>
<td>March 27, 1965</td>
<td>Reactor power level raised to 45 MWt</td>
</tr>
<tr>
<td>August 26, 1968</td>
<td>Regular power operation begins at 50 MWt</td>
</tr>
<tr>
<td>September 25, 1970</td>
<td>Regular power operation begins at nominal 62.5 MWt</td>
</tr>
<tr>
<td>June 30, 1972</td>
<td>Operation with new stainless steel reflector in core</td>
</tr>
<tr>
<td>August 1974</td>
<td>Tenth anniversary of power operation</td>
</tr>
<tr>
<td>January 1, 1977</td>
<td>Achieved a plant factor of 76.9% for the calendar year of 1976</td>
</tr>
</tbody>
</table>
III. PLANT DESCRIPTION

The design of EBR-II is based on the pot-type concept, i.e., all major primary components are immersed in a large double-walled tank that contains 86,000 gallons of 700°F sodium.\textsuperscript{3,4} Principal features are illustrated in Fig. 1.

Two centrifugal pumps, each rated at 4500 gpm, take suction from the bulk sodium. The pumps discharge sodium under a pressure of 45 psi to a plenum under a grid plate that serves two principal purposes: to provide support for core and blanket subassemblies and to regulate the upward flow of coolant. Coolant flows upward through the core and surrounding blanket into an upper plenum through an outlet pipe, and into the intermediate heat exchanger. From here the primary coolant discharges directly to the bulk sodium. Coolant enters the heat exchanger at 883°F and leaves at 695°F. A dc electromagnetic pump, rated at 550 gpm, is located in the outlet pipe. This pump operates continuously with the specific purpose of removing decay heat in the event of a primary pump coastdown.

The secondary sodium cooling system is isolated from the primary system at the heat exchanger. Secondary sodium at 586°F is pumped through the heat exchanger at a rate of 5000 gpm. The exit coolant, at a temperature of 872°F, is pumped by a water-cooled ac linear-induction pump to the sodium-boiler plant. Here the thermally hot but nonradioactive secondary sodium is passed through two superheaters, eight evaporators, and into a surge tank from which the coolant is pumped back to the intermediate heat exchanger to complete the cycle.

Superheated steam at a temperature of 815°F and a pressure of 1250 psi is piped to a 20-MW turbine-generator located in the power plant building. Under full power operating conditions, 19.5 MW of electrical energy is generated. Approximately 4.5 MW is used to satisfy the ANL/W site demand; the remaining 15 MW is fed through a 13.8 kV loop to the INEL distribution system. In addition, approximately 12,000 lb of saturated steam per hr is bypassed for building heat, thereby effecting an annual saving of 280,000 gallons of fuel oil.
Reactivity is controlled by eight fueled control rods with boron-loaded followers and two fueled safety rods. Four other control rod locations are currently occupied by a single nonfueled drop rod for kinetic tests and three instrumented in-core test facilities. The reactivity worths of control and safety rods are, respectively, 0.53 and 0.40% $\Delta k/k$. Any one of the eight control rods may be used for fine reactivity control; all are discharged from the core under scram conditions. The two safety rods are always fully inserted; their principal purpose is to provide shutdown capability in the fuel-handling mode.

Two basic processes of fuel handling are used at EBR-II. Refueling of the reactor consists of transferring subassemblies between the reactor vessel and the subassembly storage basket (located under sodium in the primary tank). The second process transfers subassemblies between the storage basket and the Hot Fuel Examination Facility and is done during reactor operation.

The driver fuel (Mark-II) consists of metallic pins 13.50 in. in height and 0.130 in. in diameter. The composition of the material is 95-wt% uranium metal enriched to 67 wt% and 5 wt% of a mixture of metallic Mo, Ru, Rh, Pd, Zr and Nb. The pins are sodium bonded to Type 304 stainless steel jackets. A standard subassembly consists of 91 elements arranged on a hexagonal pitch. Stainless steel sections below and above the elements serve two purposes: to reduce the neutron fluence on structural components below and above the core and to reflect leakage neutrons back to the core.
IV. PLANT OPERATIONS

The EBR-II plant is operated on a 24-hour, 7-days-per-week basis. Four crews work normal eight-hour shifts according to a rotation schedule. Each crew consists of a shift supervisor, an alternate shift supervisor, a foreman, and nine or ten operators. Support personnel for each operating crew include a radiological safety technician, a coolant chemistry specialist, and at least two maintenance personnel.

Each operator is required to complete a rigorous on-shift training program which qualifies him in five areas of operation. The areas are reactor control console, coolant systems, fuel handling, power plant, and electrical systems. The on-shift training emphasizes practical experience that is blended with intensive classroom work. Two and one-half years are normally required for an operator to become fully qualified. He must demonstrate a high degree of knowledge by successfully completing comprehensive written, oral, and operating examinations. To maintain his proficiency, an operator is given oral and written requalification examinations each year.

Run lengths are nominally 2700 MWd or 43 full-power days. Two factors determine the run length: the need to discharge irradiation experiments at scheduled intervals and the need to replace spent fuel with fresh fuel. Approximately seven days are needed between runs to accommodate refueling operations and to perform minor maintenance activities that cannot be carried out with the plant running.

Each year, the plant is shut down for 4-6 weeks to carry out more comprehensive modification, maintenance, and inspection activities. The sodium systems are cooled to 350°F, the secondary sodium system is drained, and power plant components are cooled to room temperature.

Refueling time between runs is not a serious constraint. The capability of interim in-tank storage permits the preshutdown transfer of fresh fuel subassemblies to the storage basket. Approximately 4 hr after shutdown, spent subassemblies may be transferred from the core to the storage basket and replaced with fresh subassemblies. The turnaround time per subassembly
amounts to approximately 1 hr. In the absence of problems, the time required for end-of-run refueling operations amounts to approximately 24-48 hr.

The interim storage feature is beneficial in another important respect. After fulfilling minimum cooling requirements, spent subassemblies may be transferred from the primary tank and replaced by fresh subassemblies while the reactor is running.

The operation of EBR-II has always been smooth and trouble-free. A relatively prompt negative power coefficient of reactivity, i.e., \(3.6 \times 10^{-5}\) \(\Delta k/k\) per MW, effectively damps the effects of small reactivity changes caused by inlet temperature variations, control rod motion, etc.

Routine operations are occasionally interrupted by the release of gaseous fission products from a failed fuel element. The majority of the releases are the inevitable result of an experiment in which fuel elements are intentionally irradiated to failure. On other occasions, the failure may be initiated by birth defects in the cladding or by premature failure of the cladding.
V. PLANT MAINTENANCE

Thirteen years of operating experience have demonstrated the feasibility of carrying out routine and special maintenance activities on components and equipment that are peculiar to LMFBR systems. In general, preventive maintenance activities are limited to the periodic inspection, repair, and replacement of conventional components that operate in relatively accessible areas. All such activities are carried out in accordance with approved written procedures and at intervals established through accumulated operating experience, the manufacturers' recommendations, and the importance of equipment to plant operation. Although most preventive maintenance work can be scheduled and performed irrespective of reactor operation, there are other activities that require plant shutdown. In either event, work is carefully scheduled to provide the least possible impact on plant availability. For example, activities that require shutdown conditions are scheduled to coincide with those periods in which the plant is routinely shut down for refueling or during the annual maintenance/modification shutdown.5
VI. PLANT PERFORMANCE

The EBR-II plant capacity factors are listed for the years from 1965 through 1975 below in Table II.

<table>
<thead>
<tr>
<th>Year</th>
<th>Plant Capacity Factor, %</th>
</tr>
</thead>
<tbody>
<tr>
<td>1965</td>
<td>26.4</td>
</tr>
<tr>
<td>1966</td>
<td>43.0</td>
</tr>
<tr>
<td>1967</td>
<td>20.1</td>
</tr>
<tr>
<td>1968</td>
<td>41.8</td>
</tr>
<tr>
<td>1969</td>
<td>42.4</td>
</tr>
<tr>
<td>1970</td>
<td>57.9</td>
</tr>
<tr>
<td>1971</td>
<td>39.1</td>
</tr>
<tr>
<td>1972</td>
<td>45.9</td>
</tr>
<tr>
<td>1973</td>
<td>49.9</td>
</tr>
<tr>
<td>1974</td>
<td>58.7</td>
</tr>
<tr>
<td>1975</td>
<td>66.1</td>
</tr>
<tr>
<td>1976</td>
<td>76.9</td>
</tr>
</tbody>
</table>

Plant capacity factor is defined by the following equation:

\[
P.F. = \frac{\text{MW Hours Produced}}{\text{Calendar Hours} \times 62.5 \text{ MW}} \times 100
\]

During its history, EBR-II has achieved a respectable operating record. In particular, EBR-II demonstrated reliability and maintainability by achieving excellent plant capacity factors in 1974, 1975, and 1976 for an experimental reactor. The plant capacity factor is not as high as would be expected for a commercial nuclear power plant. It has been affected primarily by the irradiations program and to a lesser extent by spurious reactor scrams and operational difficulties.

If EBR-II had been operated solely as a power-producing reactor rather than a research reactor, the plant capacity factors could have been 10 to 15% higher than those achieved from 1970 through 1976.
A total of 432 scrams (with the reactor power at critical or above) have occurred since 1965. The number for each year is listed in Table III.

Table III. EBR-II Scrams Since 1965

<table>
<thead>
<tr>
<th>Year</th>
<th>Number of Scrams</th>
</tr>
</thead>
<tbody>
<tr>
<td>1965</td>
<td>83</td>
</tr>
<tr>
<td>1966</td>
<td>51</td>
</tr>
<tr>
<td>1967</td>
<td>35</td>
</tr>
<tr>
<td>1968</td>
<td>56</td>
</tr>
<tr>
<td>1969</td>
<td>47</td>
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<tr>
<td>1970</td>
<td>31</td>
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<tr>
<td>1971</td>
<td>35</td>
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<tr>
<td>1972</td>
<td>22</td>
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<tr>
<td>1973</td>
<td>24</td>
</tr>
<tr>
<td>1974</td>
<td>18</td>
</tr>
<tr>
<td>1975</td>
<td>16</td>
</tr>
<tr>
<td>1976</td>
<td>14</td>
</tr>
</tbody>
</table>

Spurious instrumentation trips have caused 70% of the scrams. Efforts have been initiated to reduce the number of spurious scrams by upgrading the equipment for the plant protection system and reducing the number of redundant and/or unnecessary functions that can cause scrams. When the plant was designed, numerous anticipatory scram functions were originally included in the scram system because of the uncertainty associated with the performance of the scram system. Recent testing has revealed that the anticipatory scram functions can be removed without introducing any new safety hazards.

To simplify the shutdown system, removal of the anticipatory scram functions began in late 1974 and was completed in 1976.

Other causes of scrams have been the commercial power system, operating problems, and manual scrams. Several scrams have occurred each year because of loss of power or perturbations in the commercial power system supplying the Argonne National Laboratory-West site. Manual scrams have been required during special tests, to end reactor runs, and when equipment failures that
did not cause reactor scrams necessitated a rapid shutdown of the reactor. Operating problems causing scrams include operator error, erroneous set points, and adjustments to instrumentation.

A. Primary System

The performance of the primary system and reactor has proven very good. Throughout its operating history, the reactor has been highly stable and readily controllable. The mechanical stability of core components has also been excellent. No indication of vibration or coolant-flow blockage has been observed during the years of operation, nor has any indication of gas entrainment been detected in the coolant.

No incidents have occurred, either in the reactor core or primary tank, that were not readily repairable. Two subassembly upper adapters were inadvertently twisted during fuel handling in the core, but these subassemblies were removed from the primary tank with little difficulty and by normal procedures. A control rod thimble was damaged slightly in 1967; this problem also occurred during fuel-handling operations. A special thimble-removal tool was fabricated and used to remove the damaged thimble by normal fuel-handling procedures. During early operation of the oscillator rod, a linear ball bushing in the oscillator drive shaft failed and released a number of small steel balls about 1/8 in. in diameter. Some of the steel balls became lodged between two control rod drives adjacent to the oscillator rod and were removed with minor difficulty. A total of 40 of the original 105 steel balls of the linear bushing were recovered. The unaccounted-for steel balls are assumed to be located in areas of the primary tank where they can do no harm. No additional problems have resulted from the unaccounted-for steel balls.

Removal of fuel-handling and primary components from the primary tank is a job that has been done routinely and without significant problems. Since the primary tank was filled with sodium, the core gripper has been removed four times, the transfer arm has been removed once, control rod drives have been replaced sixteen times, and primary pumps have been removed and reinstalled three times. Mechanical failures have occurred ten times in control rod drives, three times from gripper-jaw failures, and seven times from bellows failures in the driveshafts.
Only one incident has occurred during the many interchanges of components within the primary tank. The auxiliary gripper plug was being reinserted when a chemical reaction occurred in the hole for the auxiliary gripper plug. A small amount of sodium was expelled around the plug shaft and up through the hole. Investigation indicated the auxiliary gripper plug had not been adequately dried after cleaning and the residual alcohol-water cleaning solution reacted with the sodium. Damage to equipment due to the expelled sodium was insignificant and contamination was minor. The auxiliary gripper plug was not damaged. Since this incident, more stringent rules have been imposed to ensure adequate drying of each component prior to installation into the primary tank.

B. Secondary System

The secondary system has been essentially trouble-free except for an early failure of the main secondary sodium pump (due to a design deficiency), a pinhole leak (due to a faulty weld) of water to atmosphere in one of the sodium-to-water steam generators, and a sodium leak of 80 gal during maintenance activities. The piping in the secondary system has never experienced a weld or pipe failure. However, several ring-joint flanges have had minor sodium leaks. These leaks were detected by flange leak detection or during maintenance inspection.

Since the initial problem with the secondary sodium pump, performance has been very reliable under all conditions up to and including full flow. The steam generator with the pinhole leak was readily repairable, and all the steam generators (eight evaporators and two superheaters) have performed in an excellent manner. No evidence of leakage of water to sodium has been encountered.

C. Power Plant

The power plant is essentially a conventional steam plant. Performance of the systems has been very good and the power plant has seldom been responsible for loss of plant availability.
D. Fuel Handling

The fuel-handling system has been used extensively. As of January 1, 1977, operating experience with the fuel-handling system consisted of over 17,000 separate operations. The unique design of this system, which provides for very rapid refueling of the reactor, results in efficient use of the reactor, and permits interim storage within the primary tank. Most of the fuel-handling components are original equipment and have operated many thousands of times with only minor difficulties. Some design changes were required after initial check-out, and a number of changes were made after operating experience was gained; but, in general, the system has performed very well.

E. Instrumentation

1. General

Throughout its operating history, EBR-II has lost a number of primary flow, pressure, and temperature-sensing devices that are not replaceable. Few of the key parameters have failed. However, additional failures could eventually affect plant operation if all monitoring capabilities for any key parameter such as flow or core △T were completely lost. Alternative methods are being studied for monitoring both core △T and primary flow.

2. Flow Instrumentation

Magnetic flowmeters were installed in the high-pressure-plenum and low-pressure-plenum inlet lines from each primary pump and in the reactor outlet piping (Fig. 1). These were chosen because of their simplicity and linear output for a wide range of flow rates.

At the time of installation, the state-of-the-art required that the magnetic flowmeters be calibrated in place. To provide this capability and to increase system reliability, venturi flow tubes were installed in series with each of the magnetic flowmeters. All of the venturi tubes have failed except the one located in the reactor coolant outlet piping. In addition to the losses of the venturi tubes, the EM flowmeter on one high-pressure sodium-inlet line failed and the EM flowmeter on the sodium-outlet line from the reactor failed. The single remaining flowtube is being used for indication of sodium flow out of the reactor.
3. **Temperatures**

An extensive network of thermocouples and some resistance thermometers were installed in the EBR-II primary system to monitor temperature conditions. Resistance thermometers were installed in areas where high accuracy in measurement was desired; however, the majority of the temperature sensors were thermocouples because of the lower costs and greater ease of installation. Among the components monitored are the primary pumps, the intermediate heat exchanger, and the coolant inlet and outlet lines, the instrument thimbles, subassembly outlet temperatures, sodium temperature, and the primary tank.

To date, many of the thermocouples have failed and none of the original ten resistance thermometers are working. Because of the redundancy used, however, these failures have only posed a problem in monitoring one parameter -- coolant outlet temperature. Originally, five thermocouples and two resistance thermometers were available in the outlet piping. Now only two thermocouples remain operable, and one of those is showing signs of failure.

4. **Pressure**

Sixteen pressure sensors were installed in the primary system. These sensors transmit primary sodium pressure via a NaK-filled (22% sodium, 78% potassium alloy) capillary to a pressure transmitter outside the primary tank. Of the 16 installed pressure sensors, only three are still operable. Two of these are at the discharge of the primary pumps and can be replaced, if necessary, whenever the primary pumps are removed. The third is in the coolant outlet plenum.

Of the 13 failed sensors, nine were associated with the reactor vessel inlet and outlet coolant pressures and the other four were in the primary pump high pressure and low pressure plenum discharge lines. None of these is replaceable.

5. **Level**

Two types of sodium-level detectors were originally used in the primary and secondary systems. Resistance type probes were installed in the primary tank, secondary storage tank, and secondary surge tank. A pressure sensor that included a temperature compensator was also installed in the primary tank to measure the static pressure head of the sodium. The resistance probes are still used as
the main level indication in the secondary system and as a backup for the primary tank. Several problems have occurred with the resistance probes. They indicate level only in stepwise increments of several inches as displayed by a series of lights. Erroneous readings have frequently occurred because of insulation breakdown of the probes or bridging of sodium or sodium oxide from one probe to another.

The pressure sensor in the primary tank did not give the desired accuracy or the reliability for a continuous level indicator and was replaced by a float device in 1969. The float device was developed at EBR-II. It consists of a partially submerged buoyancy cylinder hanging from a force transducer. This device has operated successfully since installation and has demonstrated an accuracy of ±1/4 in. over a measuring range of 20 in. Similar measuring devices are scheduled to be installed in the secondary storage and surge tanks.
VII. INDIVIDUAL COMPONENT & SYSTEM PERFORMANCE

A. Primary Pumps

The primary system uses two centrifugal pumps. They are operated in parallel at approximately 800 rpm to supply 9000 gpm of 700°F sodium at 46 psig to the reactor core. Each pump is purged with 5 cfm of argon to prevent the migration of sodium vapor into the motor case and to retard the deposition of sodium vapor on the pump shaft.

Initial operation of the primary sodium system began in April 1963. Binding occurred in both pumps during their first six months of operation. Failure was caused by excessive rubbing between the pump shaft and the lower labyrinth. Both pumps were removed and modified to eliminate the rubbing. Modification consisted of replacing the pump shafts and increasing the clearance between the shaft and lower labyrinth and baffle plates.

Since initial modification, primary pump No. 2 has performed satisfactorily for over 53,000 hr. Pump No. 1 experienced two additional incidents of binding attributed to the buildup of sodium and/or sodium oxide in the running clearance between the pump shaft and lower labyrinth. After each binding, the shaft was manually rotated by the application of about 200 ft-lb of torque to begin rotation. The shaft then rotated freely. The pump subsequently performed satisfactorily until mid-1970, when periodic increases in power consumption were noted with no corresponding change in primary sodium flow. Further investigation indicated that excessive torques were required to rotate the pump shaft. Although some success was achieved in providing freer rotation of the shaft, the pump was scheduled for removal.

Inspection after removal and disassembly revealed a considerable buildup of sodium and sodium oxide on the pump shaft, inside the lower labyrinth, and on the shield-plug liner above the labyrinth. Shaft rubbing had occurred in these areas. Sodium oxide had severely corroded the internal serrations of the lower labyrinth. The lower labyrinth was replaced with a unit of modified design, and minor rework was done on the baffle-assembly drain holes to improve the argon purge in order to reduce the buildup of sodium oxide.
Since the repair of these pumps, no other problems, except minor malfunctions of the electrical control components, have been encountered.

B. Primary Auxiliary Pump

The primary auxiliary pump is a permanent-magnet dc electromagnetic unit located in the discharge pipe of the reactor. This pump provides a minimum coolant flow through the reactor at all times when the reactor vessel cover is in its down position. Originally, about 8 in. of exposed copper extended from the bottom of stainless steel bus bar tubes. This copper was in direct contact with the sodium and was the electrical power supply for the pump.

After three years of operation of the pump, copper was noted in the plugging meter for the primary system. This copper originated from dissolution of the bare copper bus bars; about 10 lb of copper was dissolved in the primary sodium.

Both bus bars were repaired by machining the bare copper bus bar ends to a smooth surface. Stainless steel sleeves were machined and pressed over the copper bus bar ends to provide a tight shrink fit. The resulting repair ensures that no copper or copper alloy comes into contact with the sodium. Subsequent examination of the stainless-steel-clad bus bars showed them to be in excellent condition, with no evidence of further dissolution or attack.

C. Electromagnetic Pumps for Primary Auxiliary Systems

Three primary auxiliary sodium loops use basically the same type of pump. These are the sodium purification system, the radioactive sodium chemistry loop (RSCL), and the fuel element rupture detector loop (FERD). The pumps used in these systems are dc electromagnetic pumps rated at 100 gpm.

During the initial operational checkout, chronic gas entrainment in the FERD pump made it difficult to establish and maintain stable flow in the system at low flow rates. Two months after initial operation, a sodium leak developed in the pump duct during startup. All leaking sodium was contained within the pump housing. The failure was due to gas entrainment causing overheating in the pump duct.
The FERO loop was redesigned and a new pump was relocated at an elevation below the original pump. These changes eliminated the problem of gas entrainment within the pump.

The only other difficulty with this type of pump has been in the bus bar connections, internal and external, which caused high-resistance connections. In three instances, external connections had to be modified and internal connections of two pumps had to be changed from bolted to welded connections. In general, the auxiliary sodium pumps have performed exceptionally well.

D. Intermediate Heat Exchanger (IHX)

The IHX is located within the primary tank. The primary sodium from the reactor passes through the shell side of the tube bundle and returns to the primary tank. The nonradioactive secondary sodium flows through the tube side to remove heat from the primary system.

Except for a single event involving the sodium drain tube, operation of the IHX has been trouble-free.

In November 1970, a loud banging noise was heard in the vicinity of the IHX. Indications were that the noise source was within the IHX inlet pipe. After evacuating the secondary sodium from the IHX, a port for access to the IHX internals was installed on the inlet-pipe elbow over a rectangular hole that was cut in the elbow. Visual examinations, by the use of both a periscope and a remote TV system, revealed that the two support clips holding a 1-in.-diam drain tube in place were not in their originally installed condition. The top clip was loose and the bottom clip was missing. This condition allowed the sodium tube to move and vibrate against the wall of the 12-in.-diam inlet pipe. Evidence of wear on both the 12-in. and 1-in. tube was found.

The upper clip was removed; the sodium tube was cut at the top and bottom and removed. The lower clip was not found. The section cut out of the inlet elbow was rewelded in place, and the secondary system was restored to operational status. Quiet operation of the IHX verified that the repair was successful.
E. **Primary Sodium Leaks**

Two leaks of primary sodium to air have occurred. The first one occurred in 1969 and the second one occurred in 1971. Both were in the purification system, were small, and resulted in fires, but occurred when the reactor was shut down and the activity of the sodium was low. Hence, no significant radiological problems were encountered.

The first leak was from a 3/4-in. electromagnetic pump, which operated at a flow rate of less than 5 gpm and was used for sodium sampling. This leak occurred as a result of the fracture of the braze between the pump electrodes and the pump duct. Subsequent electrical arcing from one of the electrodes to the duct melted a small area of the duct. An estimated 3.5 lb of sodium leaked from the system and caused the small sodium fire in the vicinity of the pump. Damage was light. Examination of the failed pump revealed two apparent reasons for the failure of the braze. The pump was deformed, rather than being rectangular in cross section, it bulged. This deformation occurred while the sampling system sodium was being melted in preparation for sampling. The other factor contributing to failure was inadequate wetting of the braze between the electrodes and the pump duct.

The second leak was from a small sodium-to-air heat exchanger used in the plugging loop. Damage was light, having been confined to the lower portion of the plugging loop. The total amount of sodium which leaked from the system was less than 8 lb. The heat exchanger consisted of a piece of 1/2-in. stainless steel pipe with eight cooling fins attached along its length. To accommodate the cooling fins, channels had been cut longitudinally along the surface of the pipe. To secure the fins in the channels, each end of the fins had been tack-welded to the pipe. A crack was found at the base of one of the channels. Precision measurements, made in the area of the failure, showed that the channel had been machined almost completely through the wall of the pipe. It appeared that fatigue cracking occurred in this thin-walled section of piping, and was due to thermal cycling of the sodium-air heat exchanger. Further examination showed that the machined thinning of the channels was due to a bow in the pipe. In the process of machining, the depth of cut varied as a function of bow. Too deep a cut was taken on one side of the bowed region and too shallow a cut was taken on the other.
F. Primary System Auxiliaries

1. Sodium Purification and Monitoring

The EBR-II primary sodium is normally continuously purified by cold trapping of those impurities that have solubility limits. The 500-gal cold trap is NaK-cooled and of the counter-flow type with the inlet flow passing down through the outer shell and outlet flow passing up through a mesh-packed inner chamber. The normal flow rate is about 22 to 24 gpm, and the minimum temperature is controlled at about 240°F. During cold-trap operation at a minimum of 240°F, oxygen and hydrogen (two important impurities) are purified to <1.0 ppm and <0.08 ppm, respectively, values which are quite close to the solubility limits at 240°F for each of the species. In EBR-II's history, the primary cold trap has been replaced two times after a temporary cold trap was used to clean up the initial sodium fill of the reactor. The lift of the cold trap can be limited by impurity loading or radioactivity buildup that would restrict maintenance operations. So far, impurity loading has been the cause of the end of cold-trap life at EBR-II. Because of the large number of run-to-cladding failure experiments now in the reactor, future replacement of the cold trap will probably be due to radioactivity buildup of long-half-life isotopes.

A rather crude and insensitive plugging temperature indicator (PTI) was used on a once-per-shift basis to monitor the effectiveness of the cold trap operation. The PTI uses a slotted valve stem to collect the precipitating impurities. A new, improved PII was installed in 1977, which uses a 60-micron micro-metallic filter element for the plugging component (Fig. 2). This filter is more sensitive to precipitated impurities than the old slotted plugging valve and provides a more precise measurement of the plugging temperature. The initial cleanup of the primary sodium involved the operation of a temporary cold trap with PTI monitoring of the extent of cleanup. The cleanup required of the stainless steel primary system was less extensive than for the 2 1/4 Cr. 1 Mo steel of the oxidized and scaled secondary system.

Plugging temperatures up to 400°F were measured during the initial cleanup of the primary sodium. Major maintenance work such as repairs to the primary pump and the IHX have resulted in plugging temperatures up to 280°F at the restart of cold-trapping operations. Normally, no flow changes (plugging)
are noted for temperatures down to 225°F. A notable exception occurred in 1967 when sufficient amounts of copper dissolved in the primary sodium to form a permanent plugging condition in the primary PTI. The source of copper was traced to the exposed copper electrodes of the primary auxiliary pump.

Since 1972, on-line oxygen and hydrogen meters have been in use to continuously monitor those purities. The oxygen concentration is normally 0.8 to 1.0 ppm and hydrogen 0.06 to 0.08 ppm. The reactor operating limits are 2.0 ppm and 0.2 ppm, respectively, corresponding to the 300°F plugging temperature limit.

Metallic and radioactive impurities are periodically measured by taking and analyzing appropriate sodium samples. The major metallic impurities are Bi (2.6 ppm), K (145 ppm), Fe (0.2 ppm), Pb (10 ppm), Si (0.2 ppm), Sn (40 ppm), and C (0.2 ppm). The major radioactivity impurity is $^{24}$Na, which has a 15-hr half-life, and thus is a problem only when the reactor is operating and shortly thereafter. At full power operation, the $^{24}$Na activity is about 2.8 mCi/g. Other major isotopes with longer half-lives are $^3$H (100 nCi/g), $^{22}$Na (80 nCi/g), and $^{137}$Cs (40 nCi/g). Twelve other radionuclides introduced to the sodium by fuel-cladding breaks or by neutron activation of structural and other materials are also periodically monitored.

2. Cover-gas Monitoring and Control

The EBR-II primary cover gas is continuously monitored by two on-line gas chromatographs which sense helium, hydrogen, nitrogen, and oxygen in the argon. Oxygen is normally detectable for only one datum point on the chromatograph recorder after air is introduced, because oxygen reacts so rapidly and completely with the sodium. The hydrogen concentration (from moisture in the air) is reduced exponentially, but does not reach equilibrium for 8 to 10 hr. Helium is normally zero and is only an indicator of leaks in certain experimental subassemblies or possibly from canned graphite shielding. No leaks have occurred in the canned graphite. Because of the inadvertent admission of small amounts of air, nitrogen concentrations over 10,000 ppm are observed during and after fuel-handling operations. The normal nitrogen concentration is about 2000 to 4000 ppm.
Gas samples are taken periodically for analysis of carbon-bearing species, and for radioactive Ar, Xe, and the Kr fission-product gases and $^3$H. Methane concentrations of 1 to 10 ppm with trace amounts of CO and CO$_2$ and other hydrocarbons are normally found. Methane concentrations over 50 ppm have been measured and related to inadvertent introduction of lubricants to the primary sodium from the primary pumps and fuel-handling components. During full-power operations, $^{135}$Xe and $^{133}$Xe activities of $1.5 \times 10^3$ nCi/l and $1.1 \times 10^3$ nCi/l, respectively, are normally produced from "tramp" fissionable materials. These levels may be exceeded by several orders of magnitude during fuel-cladding breaches that release gaseous fission products. The $^3$H concentration of primary cover gas normally runs about 50 nCi/l.

G. Secondary Electromagnetic Pump

Sodium is circulated through the secondary loop by an ac linear-induction electromagnetic pump rated at 6500 gpm. The pump is enclosed and water-cooled. There are no valves or other control components in the secondary loop, other than this pump to control flow. The pump is also able to provide a reverse pumping force. This feature is necessary when the reactor is shut down, and it is desirable to keep heat losses from the primary tank at a minimum by retarding thermal convection flow in the secondary system.

During April of 1964, a sodium leak occurred in the pump duct after 1385 hr of operation. This leak was caused by three separate fatigue cracks in the flat stainless steel duct sheet. Inspection of the top duct sheet showed two small cracks about 15 in. from the inlet end. One crack was found in the lower duct sheet, in about the same region as those on the top sheet. The location and geometry of the cracks indicated that the failure was probably caused by fatigue rather than material defects. Repair consisted of removing most of the cracked areas from the duct with a hole saw, then welding disks of new metal into the holes.

To eliminate the low differential pressure across the duct that caused the failure, a vacuum control system for the pump case was installed. This system maintains the interior of the pump case at a vacuum and keeps the pump duct tightly expanded against the stators to eliminate any possible flutter or vibration of the duct. No further failures have occurred since the vacuum
control system was installed. Each year since 1964, the pump has operated continuously except for one or two months when the secondary system was drained. Of the approximately 100,000 hr of pump operation, about 49,000 hr have been at flows above 400 gpm. The remaining 51,000 hr have been primarily in a reversed-head condition at low flow.

H. Steam Generators

The EBR-II steam generator is a natural recirculation-type unit that employs eight shell-and-tube type evaporators and two similar superheaters to produce superheated steam. The evaporators and superheaters use duplex tubes, some metallurgically bonded and some draw-bonded, with double tube sheets. This construction minimizes the possibility of water-to-sodium leakage. The evaporators, superheaters, and steam drum are fabricated of Croloy 2.1/4.

In February 1965, a water leak was found in the air space between the water and sodium tubesheets of one of the evaporators. The details of the evaporators are shown in Fig. 3. The steam system was drained and dried. The riser pipe was cut in two places for access to the steam tubesheet for inspection. A defect was found in a tube-to-tube sheet weld near the center of the tubesheet. The defect was removed by grinding, a weld repair was accomplished with the unit in place. At that time, two removable flanges were installed in the riser between the evaporator and the steam drum to eliminate cutting and welding of the riser for subsequent inspection of the steam side of the evaporator. Operation of the plant was resumed in March 1965. No further difficulties have been experienced with either the evaporators or superheaters.

I. Water-to-sodium Leak Detection System

The EBR-II water-to-sodium leak detection system uses in-sodium, on-line monitors designed to sense the hydrogen generated by water-sodium reactions. The leak detection system consists of twelve hydrogen meter leak detector (HMLD) subsystems; one subsystem is attached to the outlet pipe of each of eight evaporators and two superheaters. Two additional HMLD's are installed at the outlet manifolds of each bank of evaporators. In the event of a leak in a steam generator water tube, the hydrogen meters will provide rapid detection
(less than 2 min) so that the secondary system may be shut down prior to extensive damage. The sensitivity of the hydrogen meter allows detection of leaks as small as $10^{-7}$ lb H$_2$O/s (0.16g H$_2$O/hr).

As illustrated in Fig. 4, the hydrogen meter consists of a sodium system; a vacuum system, and a control system. The sodium system provides a sample of sodium from the main sodium outlet line of each steam generator. A concentric counterflow design, requiring only one penetration of the secondary system pipe, allows incoming sodium to be heated by external heaters and by regenerative heat from the internal return flow. Sodium flow is maintained and controlled by a simple linear induction pump. The incoming sodium is heated from an evaporator outlet temperature of approximately 550°F (or superheater outlet temperature of approximately 780°F) to the 900°F nickel membrane temperature. The membrane temperature is controlled by an automatic heater control circuit.

The vacuum system of the detector consists of a nickel membrane, vacuum isolation valve, connecting vacuum pipes, and an ion pump. The nickel membrane is a bellows-shaped tube with an 9.375-in.-i.d. x 0.010-in. wall. The membrane forms a boundary between the sodium and vacuum. The membrane temperature is controlled at 900°F ± 2°F so as to sustain a constant hydrogen flux.

The ion pump maintains the dynamic vacuum at a pressure of $10^{-6}$ to $10^{-8}$ torr. The current produced by the pump is directly proportional to the hydrogen flux through the nickel membrane from the sodium side.

The instrument and control system provides controls and alarms needed to ensure proper operating conditions (i.e., sodium flow rate and system temperatures). The ion pump current is transmitted by high voltage coaxial cable to an ion pump controller which processes the signal to the EBR-II data acquisition system (DAS). The DAS provides alarm functions for (1) high hydrogen level, (2) high rate-of-rise in hydrogen level, and (3) system temperatures. The DAS is also capable of data storage and display of other key operating parameters.

After the initial operation, eight of ten units experienced sodium-to-vacuum leaks. Spare units were installed in a sodium test loop and were operated to
failure. Inspection revealed the cause of the failures to be inclusion stringers in the stainless steel membrane fitting to which the nickel membrane was attached via a transition weld.

Six modified units were installed in 1976 and six more in 1977. All of the original units were replaced and two new units were installed on the downstream of each bank of four evaporators. All units have performed satisfactorily, and the leak detection system is currently being placed in operation.¹⁴

J. Secondary Sodium Leaks

Only one significant leak of sodium to the atmosphere has occurred. This leak occurred in February 1968, while the reactor was shut down. During the replacement of a small valve in the plugging temperature indicator (PTI) system, a "freeze seal" failed, and this resulted in the release of approximately 80 gal of sodium. The sodium release occurred about 20 min after a fan, used for maintaining a freeze seal between the open pipe and flowing sodium, had been turned off to permit welding of the replacement valve. The sodium, initially under a pressure of about 13.5 psi and at a temperature of 509°F, streamed out of the open valve body, and almost immediately ignited. The released secondary sodium contained an insignificant amount of radioactivity. The reactor, which is in another building, was not operating at the time and was not placed in jeopardy by the fire.

The three men who were making the installation received minor burns. The damage, although unsightly, was not extensive, and cleanup and repair of equipment systems began immediately. Cleaning, repairing, and replacing equipment necessary to return the facility to operation cost approximately $25,000. The entire plant was returned to operation 13 days after the incident. As a result of this fire, more stringent rules were implemented that cover all maintenance work on sodium systems.

K. Secondary System Auxiliaries

1. Sodium Purification and Monitoring

The secondary system is normally continuously purified by cold trapping, as in the case of the primary sodium. The cold trap is similar to the one in use in the primary system, except that its volume is about 150 gal and operates at a
flow rate of about 11 gpm. Cooling to the sodium is provided by Dowtherm, which in turn is water cooled. The trap is operated to maintain a minimum sodium temperature of 240°F. In EBR-II's operating history, the secondary cold trap has been replaced four times since the cleanup operations that followed the initial startup of the system. In the secondary system, the life of the cold trap is limited by impurity loading, primarily hydrogen (as NaH) that is introduced to the sodium by diffusion of hydrogen produced by water-side corrosion in the superheaters and evaporators. A measurement of this source rate of hydrogen at full reactor power gave 0.2 lb of hydrogen per year, or a burden of about 5 lb of NaH per year to the secondary cold trap.

As for the primary sodium, a plugging temperature indicator (PTI) using a slotted valve stem was used on a once-per-shift basis to monitor the effectiveness of the operation of the cold trap. During the initial cleanup of the secondary sodium system, plugging temperatures of up to 500°F were recorded. About 130 lb of Na₂O was removed by cold trapping. The source of oxygen was iron oxide from the ferritic steel piping and heat exchangers. As in the primary sodium, the normal PTI monitoring indicates no flow changes (plugging) at temperatures down to 225°F. The reactor operating limit for the plugging temperature for the secondary system is 325°F. The higher limit for the secondary system is required because of the hydrogen source rate. This higher limit also permits plant operation with the cold trap off (for maintenance and other work) for reasonable lengths of time. A more sensitive PTI was installed on the secondary sodium system in early 1976 and is identical to the new one installed in the primary system.

The secondary sodium is also being continuously monitored by on-line oxygen and hydrogen meters. Metallic and radioactive impurities are periodically measured by the analysis of appropriate sodium samples. The major metallic impurities are iron (0.2 ppm), lead (0.4 ppm), silicon (1.0 ppm), and potassium (145 ppm). Because one of the purposes of the secondary sodium system is to prevent possible radioactive transport to the steam, few radioisotopes are present in the sodium. In fact, only two are measurable; $^{24}$Na from the fact that the intermediate heat exchanger sees some neutrons that leak from the core, and $^3$H that diffuses from the fuel via the primary sodium and intermediate heat exchanger to the secondary sodium. The concentration of $^{24}$Na is nominally 30 nCi/g at full power and $^3$H is nominally 2-3 nCi/g.
2. Cover-gas Monitoring and Control

The secondary cover gas is continuously monitored for hydrogen, oxygen, and nitrogen by two gas chromatographs. Oxygen and nitrogen are normally undetectable or very low, because there is no opportunity for air inleakage. Hydrogen is also normally <4 ppm. At high concentration, hydrogen would be an indicator of water-to-sodium leak, and therefore is followed closely by the operators.

Periodic gas samples are taken for analysis of carbon-bearing species and $^3\text{H}$; $\text{CH}_4$ and other carbon-bearing species are normally less than 5 ppm. $^3\text{H}$ is normally about 20 nCi/g for the secondary cover gas.

L. Fuel-handling Equipment

1. General

Fig. 5 shows the location of the major fuel-handling components in the primary tank. The core gripper mechanism inserts subassemblies into and out of the reactor vessel grid. It grasps the upper adapter of the subassembly and moves the subassembly vertically. The holddown mechanism operates in conjunction with the gripper mechanism. During the insertion or removal of a subassembly from the reactor vessel, the holddown mechanism functions as a guide funnel and prevents inadvertent extraction or movement of adjacent subassemblies. The transfer arm is a manually operated mechanism that is electrically interlocked. It transports subassemblies and other core components within the primary tank. The storage basket provides temporary storage of new and spent subassemblies to be inserted or removed from the primary tank.

2. Core Gripper Mechanism

In January 1964, the core gripper was removed from the primary tank to investigate the cause of sticking. A large quantity of sodium and sodium oxide was found on the shaft and guide tube at a point just above the sodium level. To minimize the buildup of sodium oxide between the gripper shaft and its guide tube, three gas-vent holes were drilled in the guide tube. These holes allowed the argon cover gas to enter and pressurize the previously stagnant space in the guide tube annulus and greatly reduced the tendency for air inleakage due to a pumping action when the gripper shaft was raised and lowered.
Later the same year, the gripper was again removed. Damage had occurred to the gripper guide funnel when the funnel interfered with the holddown while the gripper was being lowered. The guide funnel was redesigned to have less tendency to catch on the holddown.

The gripper operated routinely until 1967, when difficulty was encountered in releasing a subassembly to the transfer arm. The gripper was removed and a visual inspection was made remotely of the gripper jaws and sensing blade. This inspection revealed that the gripper was in proper working order; the difficulty was found to be due to the subassembly top adapter, not the gripper.

No problems were encountered for the next seven years. In 1974, the gripper and guide tube were both removed following sticky operation of the tripper shaft and an infrequent malfunction of the gripper jaws. The gripper jaws were inspected with mirrors and a camera, and were found to be in satisfactory condition. Once again, a buildup of sodium oxide was found on the gripper shaft and guide tube just above the sodium level. After removal of the sodium oxide buildup, three slots were milled in the guide tube to facilitate drainage. The gripper and guide tube were then reinstalled. The removal of the sodium oxide buildup eliminated the sticky operation of the gripper; however, the infrequent malfunction has persisted with the gripper jaws.

3. Holddown Mechanism

The holddown mechanism has operated virtually trouble-free with one exception. Early in 1970, the holddown began sticking near the upper end of its vertical travel. The problem appeared to be similar to the problem with the gripper mechanism in 1964. A sodium oxide buildup was apparently forming in the cover-gas space between the holddown shaft and the guide tube.

The design of the holddown provided for removal of the guide tube without removal of the holddown shaft. To provide better circulation of cover gas in the guide tube, the guide tube was cleaned, modified, and reinstalled.

4. Transfer Arm Mechanism

One difficulty required the transfer arm mechanism to be moved from the primary tank. In February 1966, the sensing pin mechanism failed because of a defective weld and severe oxidation of the shaft bushings. These bushings
were exposed to a 700°F air environment inside the transfer arm. The drive mechanism was redesigned; the transfer arm was also modified to provide an internal argon atmosphere to eliminate the oxidation problem encountered with the previous 700°F air environment. The transfer arm has operated reliably since it was reinstalled.

M. Rotating Plugs

The rotating shield plugs have been a continuing source of difficulty since the system became operational. The large rotating plug, which has a maximum diameter of 12 ft and weighs 58 tons, is mounted in the primary tank cover with its vertical axis coinciding with the vertical axis of the reactor core. The small rotating plug, which has a maximum diameter of 7-1/2 ft and weighs 45 tons, is positioned eccentrically within the large plug. At the periphery of each plug is a dip ring or blade which dips into a seal trough.

The 8-in.-deep tin-bismuth eutectic alloy in the trough must be molten during fuel handling to permit plug rotation. When the reactor is operating, the alloy must be half molten with the lower portion molten for a gas seal and the surface frozen to prevent the seal from being displaced in the event that an abnormally high cover-gas pressure occurs in the primary tank.

A copper ring was originally used at the bottom of the blade to provide an even temperature distribution in the lower part of the seal. Before the filling of the primary tank with sodium and during initial checkout of the fuel-handling system, plug rotation became increasingly difficult. Both plugs were finally removed when rotation became impossible. The copper rings were found to be badly eroded, particularly in the vicinity of the heaters. At some heater locations, the ring was completely broken. This condition had caused severe binding between the blade and trough wall. Replacement of the copper rings with stainless steel rings alleviated the erosion problem, and heat distribution was not noticeably affected.

After the primary tank was filled with sodium, difficulty with plug rotation again increased. Additional heat was usually used to enable the plugs to be moved, since it became harder to obtain proper seal temperatures. The time needed to effect rotation steadily increased. Finally, almost a full
day of seal melting was required before the plugs could be moved, and manual force had to be routinely applied to the large plug to achieve free rotation.

Inspection of the air side of the seals revealed that considerable oxidation of the tin-bismuth alloy had taken place. A dry, black, powdery oxide was found on top of the seal alloy on the air side of each seal. To improve the access to the air side of the seals, a 3/4-in.-diam hole was drilled in 1966 through the steel and high-density concrete of each of the shield plugs over the outer annuli of seals. Initial attempts were then made to clean the seals through these holes.

The first operation performed through the new access holes was a vacuum-cleaning process to remove as much dross or oxide material as possible from the surface of the alloy. This vacuuming process and exploratory probing of the alloy in the outer annuli of the troughs disclosed the presence at moderately compacted deposits of oxide-like material existing about 8 in. down in the trough, and except for a few discontinuities, apparently extending around the trough circumference.

The solid areas were found difficult to penetrate. Finally, a 1/2-in. steel tube was hammered through one of these areas and, upon removal from the alloy, the solid substance remained in the tube and could then be removed by impacting the end of the tube in a container. This procedure was soon adopted as the initial seal-cleaning method, and the complete outer annulus of each plug was cored in this manner.

When the coring was completed, it was evident that a large quantity of loose oxide remained in the seal alloy and should be removed. Temperature profiles of the seal alloy were taken through the new access holes, and it became evident that the trough thermocouples were indicating low temperatures. Oxide deposition appeared to be insulating the thermocouples from the alloy temperature, and an attempt was made to brush off the oxide with a steel brush. That effort was partially successful and some of the temperatures began to be indicated more accurately. More important, though, was the discovery that the oxide adhered to the steel brush when it was removed from the alloy. The brush-cleaning technique evolved from that incident. Cleaning is accomplished
by rotating the plug in 1/2° increments, inserting and removing the brush at least once at each increment, and cleaning the dross from the brush. After the initial cleaning and refilling of the seal trough, plug rotation became much easier.

Although crude and time-consuming, a program of periodic brush cleaning of the seals proved effective and provided relatively trouble-free plug rotation. The brush cleaning was used until November 1972 when a 3-in.-diam access hole was drilled through the plug support structure to the air side of the seal for the large plug. The method of cleaning was changed to a more direct means of skimming the large plug seal with specially designed tools.

In April 1973, a similar access hole was drilled in the small plug to the air side of the seal. The same skimming technique is now used on the small plug seal. The new access holes also permitted scraping of the seal ring, which had oxide accumulations as thick as 5/16 in. adhering to it, and provided a means of removing these scrapings from the seal. The new seal-cleaning technique resulted in a significant savings in seal alloy material, which needed to be added after each cleaning, and sufficiently reduced the time required for cleaning the seal.

Even though the air sides of the seals were being maintained in a good condition, chronic sticking of the large plug continued and became progressively worse. To achieve free rotation initially, the seal has required heating to elevated temperatures (400 to 500°F), and manual force of up to 8,000 lb has been applied to the large plug. In April 1975, a 3-in.-diam hole was drilled through the large plug. This provided better access to the argon side of the seal.

Observations through the access hole revealed that the argon side of the seal contained large accumulations of dross, but it was immediately apparent that the large plug was sticking because of large accumulations of material in the plug annulus as shown in Fig. 6. The annulus is the clearance between the large rotating plug wall and the rotating plug support structure ("z" ring) and communicates directly to the primary-tank argon blanket region.
Samples taken of this observed material consisted of 40% sodium; the remainder was mostly seal alloy material that had apparently spilled over the inside of the seal wall during seal-filling operations. The melting point of this material is approximately 800°F and it was estimated that approximately 250 to 300 lb of material was present in this annulus area.

The large-plug annulus was thoroughly cleaned using a glovebox (to prevent spread of contamination) and digger tool. Two hundred and ninety pounds of material were removed in a five-day period. Plug rotation has been trouble-free since the cleanup.

An access hole, similar to the one drilled in the large plug, was drilled in the argon side to the small plug in 1977. Initial inspection of the argon side revealed buildup of dross, but not to the extent of that observed in the large plug. Thorough cleaning is planned at a later date; to date, however, rotation difficulties with the small plug have been minimal.

A program of periodic cleaning of seals (air and argon sides) and the plug annuli should provide a relatively trouble-free plug operation in the future.
The EBR-II driver fuel element consists of a uranium, 5 wt% fissium metallic fuel pin, sodium bonded to stainless steel cladding. Three driver-fuel designs, designated Mark-I, Mark-IA, and Mark-II, have been used. Mark-I was used during the first 2-1/2 yr of operation. It was replaced by the Mark-IA that has a shorter fuel pin and a thinner restrainer, resulting in an increased plenum-to-fuel volume ratio. The Mark-II element has a thicker cladding for higher strength, a larger fuel-cladding annulus to allow greater fuel swelling prior to fuel-cladding contact, and a larger plenum volume. These fuel-element design changes have been aimed at increasing burnup potential.

Currently, Mark-II fuel elements are used for driver subassemblies, and Mark-IA elements are used for control-rod and safety-rod subassemblies that require the shorter element design. This is an especially important consideration for the higher-worth control rods (HWCR) which employ natural B4C follower capsules located axially immediately above the fuel elements. Currently, the reactor is operating with two safety rods, one standard control rod, and seven higher-worth control rods.

The current limits for fuel life is 2.6 and 6.0 at.% maximum burnup for Mark-IA and Mark-II, respectively. These limits have been determined to be quite conservative, and a detailed program is in progress to increase the burnup limits. Preliminary results indicate that the burnup limit of Mark-II can be increased to about 8.5 at.% maximum burnup within a few years.
IX. IRRADIATION PROGRAM

A. General

The use of EBR-II as an irradiation facility for the U. S. FBR program began in May 1965 with the insertion of two experimental subassemblies that contained various structural specimens and prototypal fuel rods (mixed PuO$_2$-UO$_2$ and U-Pu alloys). Since that time, the complement of experimental subassemblies has grown to as many as 65. Up to June 1977, a total of 9033 individual experiments were either completed or in the process of being irradiated. (An experiment is understood to be a single fuel element or a capsule.) Of the 9033 experiments, 2648 were mixed oxide fuels; 463 were either carbides, nitrides, or cermets; 4442 were metallic fuels; 1027 were various cladding and structural materials; 195 were concerned with control materials, e.g., boron, europium, tantalum, etc.; and 258 were other types. In the course of the irradiation program, peak fuel burnups of 17 at.% and cladding temperatures of 1500°F have been achieved for mixed oxide fuels. Peak fluences of $1.7 \times 10^{23}$ nu have been reached for structural materials, and plenum pressures as high as 2900 psi have been achieved for metallic fuel elements.

Without exception, all of the above experiments were uninstrumented. To obtain continuous readings of temperature, pressure, flow rate, etc., three control-rod locations have been sacrificed and converted to facilities that permit the installation of instrument leads. Two types of instrumented facilities are available. The in-core subassembly test (INSAT) facility provided the means for monitoring operating parameters of fuel and material specimens, instruments, etc. After completion of the experiment, leads to the facility, which superficially resembles a standard core subassembly, are cut and the experiment is removed with the standard fuel-handling equipment.

The in-core instrument test (INCOT) facility provides a direct access to the core through a control rod penetration for the in-situ testing of in-core instruments. The facility consists essentially of a thimble that extends from the top of the biological shield, through the bulk sodium and reactor cover, to the reactor core. Instrument sensors and test materials may be installed in the thimble and irradiated under either gaseous or sodium environments under core flux and elevated temperature conditions. During fuel-handling
operations, the entire assembly is raised upward to provide clearance for the fuel-handling equipment. A similar arrangement is provided for the INSAT experiments. Thus far, a total of 212 individual experiments (capsules, creep specimens, fuel elements, sensors, etc.) have been irradiated and monitored in a total of seven INSAT and seven INCOOT facilities.

Other out-of-core facilities are available for testing materials, sensors, and components under conditions of interest to LMFBR technology. Prominent among these are the radioactive sodium chemistry loop (RSCL), for experiments involving primary sodium, the nuclear instrument test facility (NITF) for testing sensors and cabling in the instrument thimbles, and the experiment equipment building.

B. Experience with Breached Elements

During the 13 years that EBR-II has been used as an irradiation facility, 38 fuel elements have developed cladding breaches and an additional six elements had fission-gas releases due to diagnostic tests. Of the above total, some 24 were releases from experiments approved to run to cladding breach.

Detection of cladding breaches is made using several on-line or continuous fission-gas detectors that monitor the primary cover gas. These detectors are (1) fission-gas monitor (FGM), (2) reactor cover gas monitor (RCGM), and (3) germanium lithium argon scanning system (GLASS). Delayed-neutron monitoring is done with the fuel element rupture detector (FERD) system. Also, routine cover gas and primary sodium samples are taken for analysis.

The FGM is a charged-wire electrostatic precipitation apparatus which responds in a pseudo-integral manner to the Cs and Rb decay products of $^{88}$Kr, $^{89}$Kr, and $^{138}$Xe. The RCGM consists of an NaI detector connected to three single-channel pulse-height analyzers set to monitor $^{133}$Xe, $^{135}$Xe, and $^{85m}$Kr. The GLASS system uses a Ge(Li) detector and multichannel pulse-height analyzer. Isotopes of primary interest that are monitored with GLASS are $^{85m}$Kr, $^{87}$Kr, $^{68}$Kr, $^{133}$Xe, $^{135m}$Xe, and $^{138}$Xe. The FERD (delayed neutron detector) system consists of six BF$_3$ proportional counters that are 2 in. in diameter and 12 in. long and grouped two per recording channel. Samples of argon cover gas are quantitatively analyzed with a mass spectrometer. Primary sodium coolant
samples taken periodically are radiochemically analyzed for specific soluble fission products ($^{131}$I and $^{137}$Cs) and for fuel material ($^{235}$U and $^{239}$Pu).

Identification of a subassembly that contains a breached element was initially a lengthy process. It was accomplished by removing one or more subassemblies and then operating the reactor to check for continued high fission-gas activity. This process sometimes required as many as six reactor startups before the leaking subassembly was removed. However, since 1972, increasing use of xenon "tags" in experimental elements has reduced identification in most cases to a matter of hours and has resulted in a corresponding reduction in the number of reactor startups required.

The cladding breaches that have occurred to date have not raised any unanswered questions regarding the safety of the reactor. In particular, post irradiation examinations of subassemblies and elements have revealed no evidence of pressure pulses resulting from breached elements, and no evidence of any breached element causing adjacent elements to fail in a similar manner. Despite the considerable plenum pressures in some of the elements, none of them has released all its fission-gas inventory at one time, and in all cases breaching of the cladding has been a benign process.

C. Run-Beyond-Cladding-Breach (RBCB) Program

An extensive RBCB program is just being implemented in EBR-II. The program as presently identified includes the following experiments:

- Two with predefected mixed oxide fuel elements
- Six with naturally defected mixed oxide elements (two experiments will be removed for transient testing when breach occurs)
- One with a predefected blanket element
- One with a predefected carbide element.

The RBCB program was planned to provide experimental support for the Clinch River Breeder Reactor Project (CRBRP) operation; the objectives were to:

- Provide a prototypic fuel and fission-product source to
  1. Monitor Pu burden in Na
2. Confirm capability of reactor instrumentation to monitor effects of sodium-fuel contact
3. Determine (infer) deposition behavior of fuel released from fuel pin
4. Correlate reactor instrumentation and breached pin behavior
   • Establish breached fuel pin behavior by
     1. Measuring reaction-induced swelling
     2. Measuring loss of fuel and fission products
     3. Determining "effective" area of exposed fuel
     4. Determining extent and mode of crack extension
     5. Transient testing breached fuel pins
   • Confirm that post breach irradiation is nonpropagating by
     1. Detailed examination of neighboring fuel pins
     2. Mechanical property tests of cladding from neighboring pins
     3. Verifying absence of coolant channel blockage.

Before any RBCB experiment could be conducted in EBR-II, tests had to be conducted to determine response of the fission-product monitoring systems to oxide elements containing cladding defects of known geometry. Of particular interest were the delayed-neutron signals to be expected from the elements. Two experiments were irradiated, each containing one predefected UO₂ fuel element, to acquire the necessary data.

In order to operate the reactor for extended periods with defective fuel, a number of modifications to the plant were necessary; the major ones were:
   • Installation of a cover-gas cleanup system (CGCS)
   • Reduction of the leakrate of the cover gas to the reactor building
   • Improvement of capability to control and detect fission products in sodium.

The CGCS is designed to remove xenon and krypton from the cover gas by means of a cryogenic distillation process. The system is designed to maintain the cover gas at low activity levels while the reactor is operating with up to 12 (U, Pu)O₂ defected elements. The system also incorporates a xenon-tag-trap subsystem to provide continuous, computer-controlled recovery of xenon-tag
isotopes for the identification of on-line failed elements. Each experimental fuel subassembly contains a unique mixture of xenon isotopes that can be identified using a mass spectrometer.

The primary tank cover was not designed to be leaktight, and as a result, the historical leak rate has exceeded 1 liter/min. Extended operation with defective fuel would lead to unacceptable radiation levels in the reactor building, even with the CGCS in operation. After an extensive effort of leak identification and repair, the leak rate to the building has been reduced to <300 ml/min.

Of the fission products released from a breached fuel element, cesium is the most troublesome. It is the major contributor to coolant activity during shutdown periods and is controlling on maintenance activities. The primary purification system is being modified to include a cesium trap.19

A vacuum distillation sampler has been installed to sample sodium for particulates. Sampling is accomplished by flowing radioactive sodium to a cup where the sodium is then removed by distillation. The cup, with only nonvolatiles, can then be removed without waiting for radioactive decay, and its contents can be analyzed.

The first RBCB experiment, containing a preirradiated mixed-oxide element with a machined slit in its cladding and measuring 0.031 in. wide and 0.87 in. long, was irradiated June 20, 1977. The test was terminated after approximately 7-1/2 hr of full-power operation, when the delayed-neutron signal reached a preset limit.

Initial examination after irradiation revealed that the machined slit had widened uniformly to 0.040 in. and had extended by thin cracks to 2.3 in. There was visual indication of a gray reaction product in the slit, and the geometric area of the slit had increased 47% (from 0.19 cm$^2$ to 0.28 cm$^2$). Significantly, the mean delayed-neutron countrate from the element showed a 48% increase from the time of achieving full power to the time of test termination.

The second test containing a predefected element is scheduled for irradiation in August 1977.
X. CONCLUSION

The results of many tests and experiments conducted at EBR-II under the auspices of the formal irradiation programs, coupled with over a decade of operating and maintenance experience, have contributed heavily to national LMFBR technology. Essentially no area of LMFBR technology has remained untouched. The task of converting a plant built with the technologies of the 1950's into a sophisticated irradiation facility has been challenging, but the challenge has been successfully met. Until its replacement by more sophisticated test facilities, EBR-II will continue to serve as the nation's prime facility for testing fuels, materials, concepts, and technologies for the nation's increasingly important FBR program.
PLUGGING INDICATOR
SYSTEM UTILIZING
MICRO-METALIC FILTER
EBR-II EVAPORATOR
SODIUM INLET END DETAILS

FIGURE 3
HYDROGEN METER LEAK DETECTOR (HMLD) FOR STEAM GENERATORS

Main Sodium Pipe

Linear Induction Sodium Pump

Heat Exchanger 1" SCH 40 Pipe

Ion Gauge (Optional)

Nickel Membrane

Isolation Valve

Roughing Port Valve

Figure 4
Fuel Handling System

FIGURE 5
CROSS-SECTION OF LARGE PLUG SEAL, SHOWING LOCATION OF DEPOSITS RELATED TO THE ARGON SIDE OF SEAL
REFERENCES


