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# EBR-II: TWENTY YEARS OF OPERATING EXPERIENCE

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## DISCLAIMER

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## 1.0 ABSTRACT

Experimental Breeder Reactor No. 2 (EBR-II) is an unmoderated, sodium-cooled reactor with a design power of 62.5 MWt. For the last 20 years EBR-II has operated safely, has demonstrated stable operating characteristics, has shown excellent performance of its sodium components, and has had an excellent plant factor. These years of operating experience provide a valuable resource to the nuclear community for the development and design of future liquid metal fast reactors.

This report provides a brief description of the EBR-II plant and its early operating experience, describes some recent problems of interest to the nuclear community, and also mentions some of the significant operating achievements of EBR-II. Finally, a few words and speculations on EBR-II's future are offered.

## 2.0 INTRODUCTION

Experimental Breeder Reactor No. II (EBR-II) is an unmoderated, sodium-cooled, reactor which was designed and constructed for the USAEC between 1957 and 1963 at the Idaho National Engineering Laboratory. It has been operated by Argonne National Laboratory since initial power operation began in 1964. During the last 20 years EBR-II has operated

safely and has demonstrated high plant availability and excellent performance of its sodium components.

The original goal of EBR-II operation was the demonstration of the feasibility of a sodium-cooled fast reactor operating as a power plant with fuel-processing capabilities provided by an adjacent Fuel Cycle Facility (FCF) which was designed for a throughout of 1000 kg/yr. During the five years that EBR-II was operated in this mode over 35,000 fuel elements, or roughly five core loadings were made in the FCF and turnaround times as short as one month between fuel out of the reactor to fuel back into the reactor were achieved. This initial phase of operation, involving on-site reprocessing of fuel was completed in 1969.

This phase of EBR-II's operation demonstrated the feasibility of the integrated operation of a nuclear power plant and a fuel reprocessing facility. The advantages of this type of operation were that the rapid fuel turnaround time alleviated the need for large fuel inventories and that there was no need for spent fuel transportation off the reactor site.

After successful demonstration of the integral fuel cycle concept, the role of EBR-II was redirected to that of an irradiation facility for fast reactor fuels and structural materials. The irradiation program has evolved from a very conservative, steady-state program to the present operational reliability testing (ORT) program. The ORT program

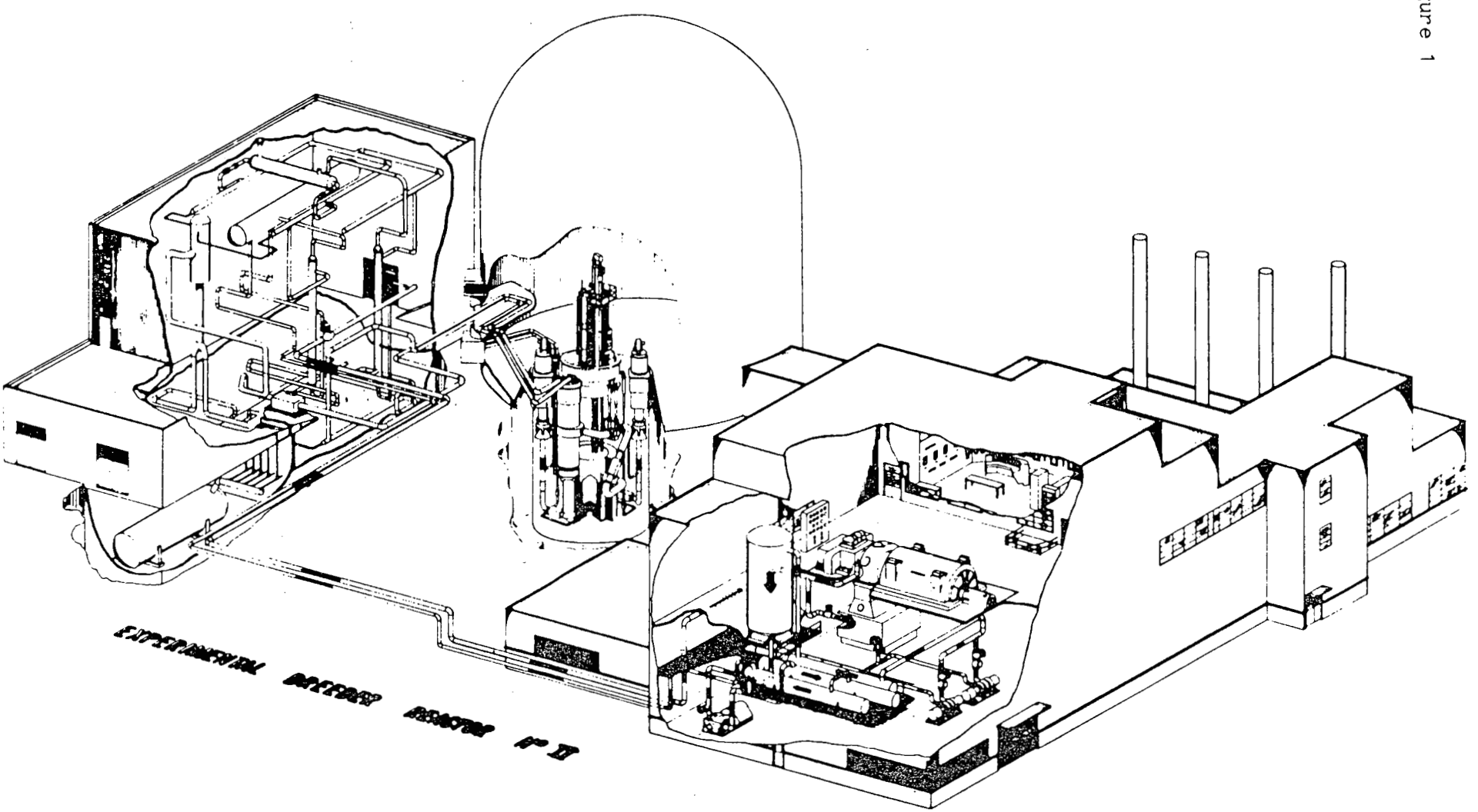
utilizes EBR-II for a more aggressive irradiation program consisting of (a) run-beyond-cladding-breach (RBCB) tests, (b) operational transient tests simulating duty-cycle transients and mild over-power transients on fuel elements, and (c) thermal-hydraulic testing of clusters of fuel elements under both normal and natural convection modes of cooling for testing and verification of shutdown heat-removal codes.

## 3.0 PLANT DESCRIPTION

EBR-II is a sodium-cooled reactor with a design power of 62.5 MWt; a closed, intermediate sodium loop; and a conventional steam plant and turbine generator which produce 20 MW of electrical power (see Fig. 1). The main buildings of the EBR-II plant are the reactor building, sodium boiler building, and power plant building. A hot fuel examination facility (HFEF) adjacent to EBR-II is used to assemble experiments for irradiation in the reactor, and to examine irradiated experiments. Part of the HFEF (HFEF/South) was originally the fuel cycle facility (FCF), which was designed to reprocess spent fuel from EBR-II.

The reactor, major primary-system components and piping, and much of the fuel-handling equipment are submerged in a large, double-walled tank containing over 341 $m^3$ of 473°C sodium as shown in Fig. 2. The two main centrifugal primary pumps, each rated at 0.347 $m^3/s$, take suction from the bulk sodium. This sodium is circulated in a single pass through the reactor, through the single outlet pipe to the intermediate

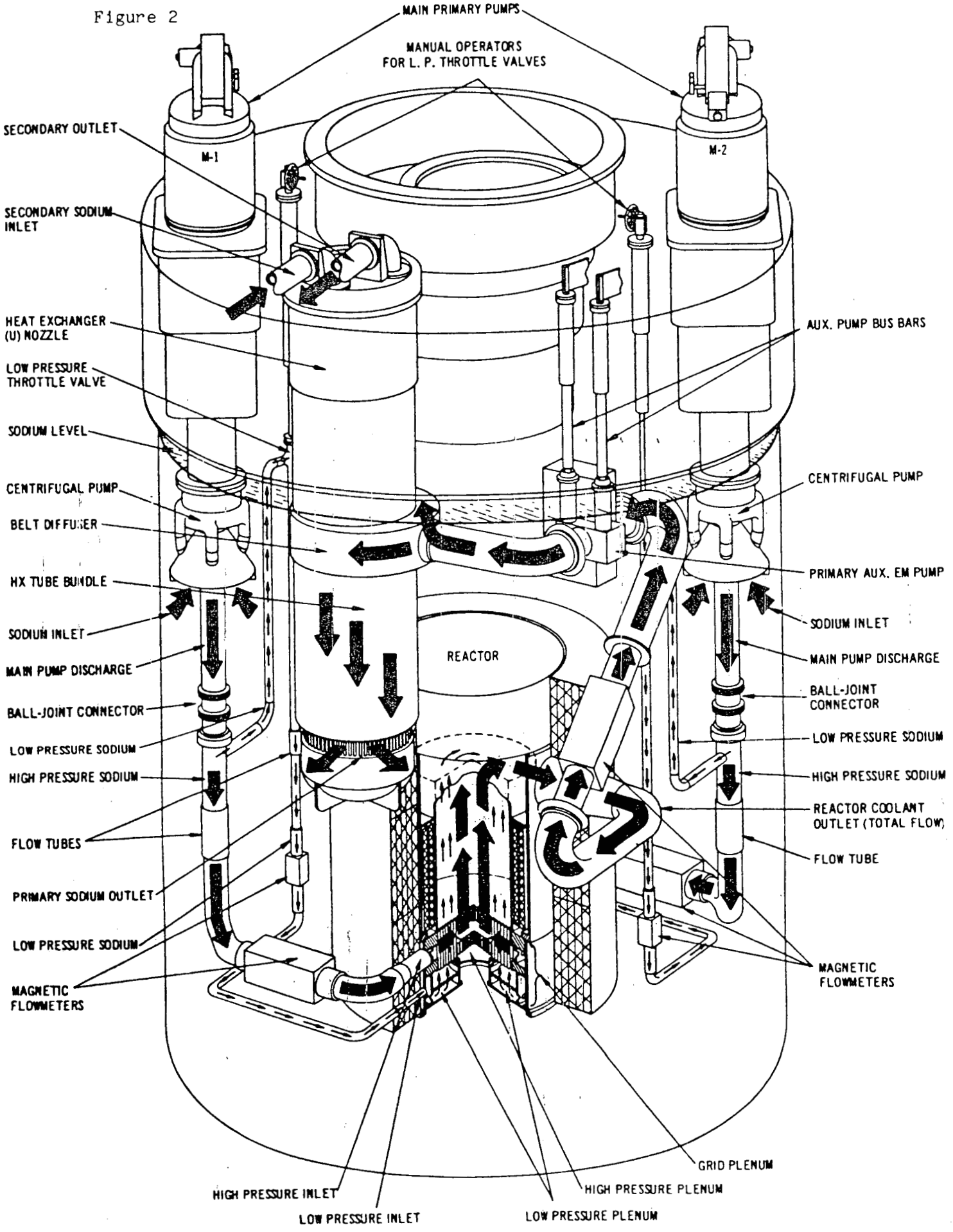
Figure 1



Principal EBR–II facilities. Shown from left to right are the sodium boiler building, the reactor and its containment building, and the power plant building

Primary Tank

Figure 2



heat exchanger (IHX), and back to the bulk sodium. A dc electromagnetic pump, rated at 0.032 $m^3/s$, is in the outlet pipe. The pump operates continuously for the specific purpose of removing decay heat, if both primary pumps should become inoperative.

The secondary system is an intermediate closed loop between the primary system and the steam system. The portion of the secondary system which interfaces with the steam generating equipment is located in the sodium boiler building. The secondary sodium loop contains about 50 $m^3$ of sodium.

The secondary sodium is pumped from a surge tank to the intermediate heat exchanger in the reactor building at a maximum rate of 0.41 $m^3/s$ by an alternating current, electromagnetic pump. It is heated from 305°C to about 467°C in the IHX. The heated sodium then flows through two parallel superheaters and seven parallel evaporators before returning to the surge tank at 305°C.

Superheated steam is supplied to the power plant building at about 8.62 MPa. The superheated steam either can be dumped to the condenser or used to produce 20 MW of electricity via the turbine generator. The generated electrical power is distributed to a 138 kV commercial power loop; however, the in-plant distribution system will automatically isolate the EBR-II site from commercial power during periods of electrical disturbances on the commercial loop and allow EBR-II to continue uninterrupted operation on its own power.

The reactor was designed to be operated with twelve fueled control rods. Two fueled safety rods were also provided for a shut down method independent of the control rods during reactor operation and to provide removable reactivity during fuel handling. Control rods and safety rods are similar to standard driver subassemblies, but contain only two-thirds the number of fuel elements. At present, four of the control rods have been removed to accommodate in-core test facilities. Any one of the remaining eight fueled control rods may be used for reactor control and all of the control rods are used for reactor scram. One control rod can be driven by a computer to provide automatic power control and shaping of power transients for fuel testing. The drive system for the two safety rods has been modified to allow them to be used for reactivity shim control.

## 4.0 OPERATING EXPERIENCE

EBR-II has achieved a respectable operating record over the last 20 years. During the first several years, some operational and equipment problems were experienced; however, these were primarily due to design deficiencies and were corrected as they occurred. Since the late 1960's, EBR-II has been a very reliable facility. The performance of the reactor has been very good, the reactor has been easily controlled and operation has been stable. Performance of the major systems and components has also been very good. Even with EBR-II's excellent record, however, there have been some operating and equipment problems which should be of interest to the nuclear community. The equipment and

operating problems of interest and significant achievements are briefly discussed in the following paragraphs.

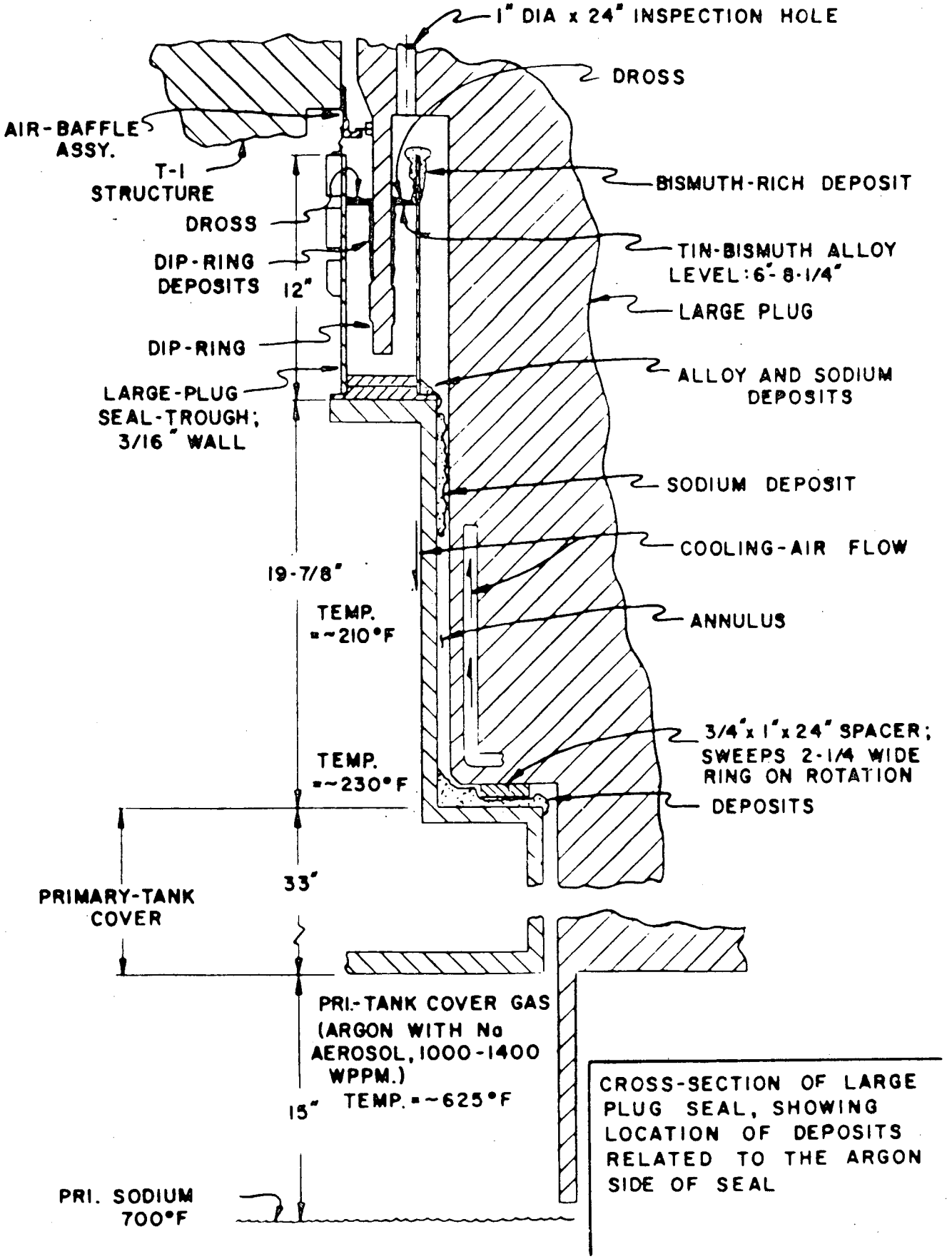
## 4.1 Recent Equipment and Operational Problems

### 4.1.1 Sticking of the Large-Rotating Plug

The large rotating plug is supported by the primary tank support structure and the small rotating plug is positioned off-center and supported by the large rotating plug. Around the periphery of each plug is a dip ring or blade that dips into a seal trough filled with a tin-bismuth eutectic alloy. The alloy provides the seal between the cover gas and the reactor building atmosphere both during fuel handling in the core when it is molten and during reactor operation when it is solid. Figure 3 shows a cross section of the large rotating plug seal.

Rotational sticking problems with the rotating seal plugs have persisted since the primary tank was filled with sodium. During the first eight to ten years of operation, the sticking problems were primarily caused by the formation of a dry, black, powdery material (oxidized alloy) on the surface of the seal alloy on the air side of the seals. The powdery oxide would mix with the molten seal alloy when the plugs were rotating and would form a mixture of oxides and seal alloy. When the oxide layer became thick enough, it would provide enough insulation that the mixture of oxide and alloy could not be easily melted with the seal heaters and would remain as a hard crust after the rest of

Figure 3


1" DIA x 24" INSPECTION HOLE
DROSS
AIR-BAFFLE ASSY.
T-1 STRUCTURE
BISMUTH-RICH DEPOSIT
DROSS
TIN-BISMUTH ALLOY LEVEL: 6"-8-1/4"
DIP-RING DEPOSITS
12"
LARGE PLUG
DIP-RING
ALLOY AND SODIUM DEPOSITS
LARGE-PLUG SEAL-TROUGH; 3/16" WALL
SODIUM DEPOSIT
COOLING-AIR FLOW
19-7/8"
TEMP. = ~210°F
ANNULUS
3/4" x 1" x 24" SPACER; SWEEPS 2-1/4 WIDE RING ON ROTATION
TEMP. = ~230°F
DEPOSITS
33"
PRIMARY-TANK COVER
PRI.-TANK COVER GAS (ARGON WITH No AEROSOL, 1000-1400 WPPM.)
15"
TEMP. = ~625°F
PRI. SODIUM 700°F
CROSS-SECTION OF LARGE PLUG SEAL, SHOWING LOCATION OF DEPOSITS RELATED TO THE ARGON SIDE OF SEAL

the alloy was molten. In 1966, an 18-mm-dia hole was drilled through the steel and high-density concrete of each of the shield plugs to gain access to the seals for cleaning. During the early attempts to clean the seal alloy, it was discovered that the oxide would adhere to a stainless steel brush and that it could be removed from the molten alloy using brushes. Although time consuming, the brush cleaning technique was effective and provided relatively trouble free rotation. In 1972, a 76-mm-dia hole was drilled through the support structure and through the top part of the seal trough wall to provide increased access to the air side of the large plug seal. A similar hole was drilled for access to the air side of the small plug seal. The new access holes allowed a more direct means of cleaning the seal alloy using sieve-like dippers.

Even though the air side of the seals could be kept clean, periodic sticking of the large plug continued and became progessively worse. It was suspected that the location of the sticking was on the argon side of the seal. To investigate, a 76-mm-dia hole was drilled through the large plug directly above the argon side of the seal trough. Inspection through the new hole showed that the large plug sticking was due to a large accumulation of material in the annulus between the wall of the large plug and the seal trough support structure. Samples of the material showed it to be intermetallic compounds of sodium and tin ($Na_{15}Sn_4$) and sodium and bismuth ($Na_3Bi$). The melting point of the material was greater than 400°C; it was soft and gummy at temperatures as low as about 100°C. It is believed that the material is formed by the reaction of the tin-bismuth alloy and the sodium vapor/aerosol which is carried

to the annulus region by argon convection currents. About 148 kg of material was removed and free plug rotation was restored when the annulus area was cleaned in 1976. The cleaning was performed using clamshell-type digging tools and a glove box.

Since 1976, routine cleaning of the air side of the seal and periodic cleaning (i.e. seven times) of the annulus have been necessary to maintain rotation of the large plug. Although the cleaning actions mitigate the plug sticking problem, cleaning is not the final solution. At present an annulus purge system is being evaluated. It is hoped that by introducing a flow of clean argon into the annulus area the transport of sodium vapor can be substantially retarded or stopped completely.

### 4.1.2 Loss of Critical Instrumentation

Throughout the operating history of EBR-II, a number of nonreplaceable sensing devices for primary sodium flow, pressure, and temperature have failed. Of primary significance is the loss of primary flow sensors and critical temperature indications.

Originally there were both magnetic flowmeters and venturi flow tubes in each of the five flow paths of sodium (i.e. two flow paths to the core, two flow paths to the blanket region, and a single outlet flow path). At present all of the flow tubes except the one in the reactor outlet piping have failed. Also, three of the five magnetic flowmeters

have failed. The two remaining magnetic flowmeters monitor the discharge flow from one of the primary pumps to the core and to the blanket region.

A substitute for the failed magnetic flowmeter in the reactor outlet piping has been installed and incorporated into the reactor shutdown system. This new flow sensor is a pressure transmitter which senses the pressure in the outlet plenum of the reactor. It is installed on the end of a removable probe, which is mounted on the small rotating plug and which penetrates into the upper plenum of the reactor vessel. Because the outlet plenum piping has fixed geometry and discharges to the primary tank sodium at a constant pressure, the pressure drop in the outlet piping is porportional to the square of reactor outlet flow and can be processed to produce an outlet flow indication. This flow sensor's performance has been excellent since it was installed. It is currently planned to expand the new system by adding a second pressure sensor and then modifying the existing one-of-four flow trip logic in the reactor shutdown system to a two-of-five configuration for increased reliability.

Numerous thermocouples and ten resistance thermometers were installed in the primary system. Many of the thermocouples and all of the ten original resistance thermometers have failed. Because of the redundancy in the original design, however, these failures have caused a problem in monitoring only one parameter -- the temperature of the sodium exiting the reactor. Of the original five thermocouples and two resistance

thermometers in the outlet piping, only one thermocouple is still functioning and its operation has become erratic. Since this temperature is used to derive core $\Delta T$, and used by the reactor operators as the basis for power control, it was necessary to develop a replacement for this parameter before the final functioning thermocouple failed.

In 1984, two new systems were installed. The first is a derived reactor $\Delta T$ signal which is continuously recalculated by a microprocessor. The miscroprocessor performs a secondary system calormetric calculation and then develops the core $\Delta T$ based on existing primary system flow. The second system is an average of four subassembly outlet temperatures measured in the upper mixing plenum of the reactor. This average outlet temperature is a very close approximation of the mixed reactor outlet temperature and is used as a backup to the $\Delta T$ calculation. Both systems have worked very well since they were installed.

### 4.1.3 Performance Degradation in a Superheater

The evaporators and superheaters are duplex-tube, double-tube-sheet units. In some of the evaporators and one of the superheaters, the duplex tubes were bonded by mechanical means and in the others, a "soldering" process was used.

In 1974, the superheater containing mechanically bonded duplex tubes began showing sudden decreases in steam outlet temperature just as full power was approached. This behavior was believed to be caused by

an increased thermal resistance of the duplex tubes caused by a reduction in the contact pressure, and finally, actual separation. The superheater was removed from service in 1981 and replaced with one of the existing evaporators which had been converted to a superheater during the previous year. The converted evaporator has worked without problems since it was installed.

### 4.1.4 Failure of an Antimony-Beryllium Source

In February, 1984, the primary sodium plugging temperature exceeded 177°C, the technical specification limit for reactor operation. Subsequent analysis of sodium samples identified an unusually high concentration of antimony in the sodium. This was the first time that antimony had ever been identified as a significant sodium contaminate. Since the normal plugging species of oxygen, hydrogen, and bismuth were present in the sodium in their normal concentration levels, it was concluded that antimony was the cause of the higher-than-normal plugging temperature. The only sources of antimony in the core were three antimony-beryllium neutron sources. An antimony-beryllium source is a two-piece assembly comprising a source rod which contains antimony and a beryllium thimble into which the source rod is inserted once it has been activated.

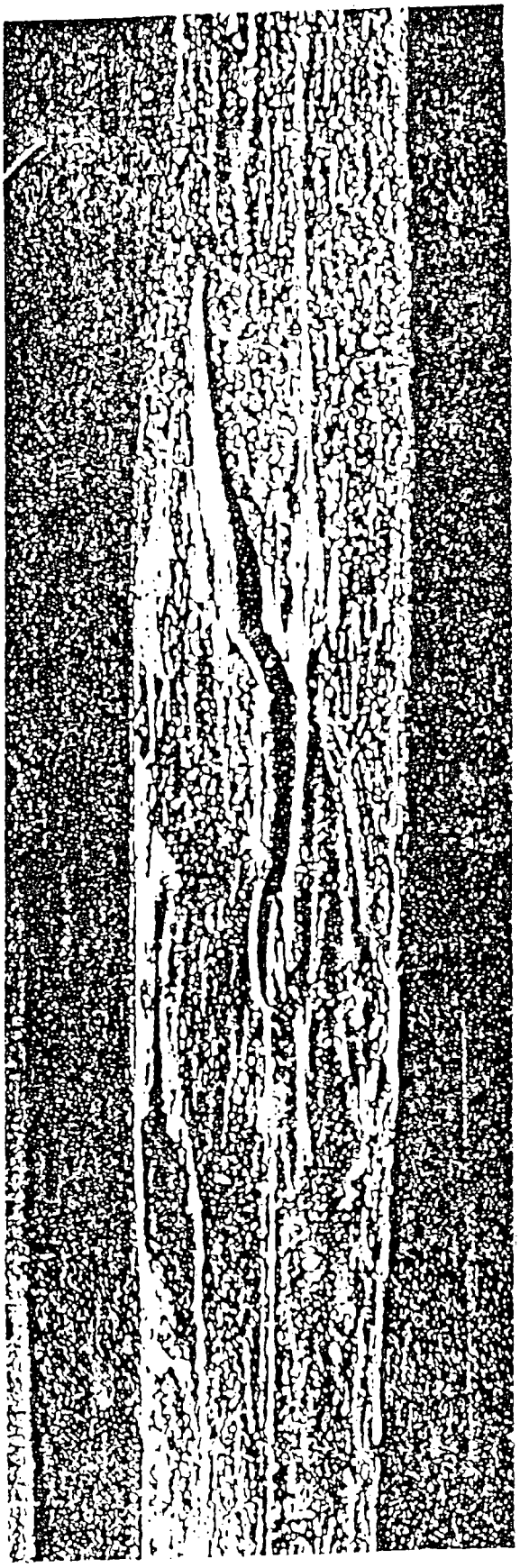
During removal of the source rods from the core for inspection, one of the rods was found to be protruding from its stainless steel storage thimble about 178 mm farther than normal. When this source rod was

removed from the core, only the top half of the rod came out of the thimble, the bottom half of the rod which contained the antimony remained in the thimble. After the stainless steel thimble was removed (with a great deal of difficulty) from the core for inspection, a visual examination revealed a 203-216-mm long crack in the 14.7-mm thick wall of the thimble (see Fig. 4). The remains of the source rod were removed by cutting the thimble open.

It has been postulated that a small defect had developed in the tantalum clad surrounding the antimony, causing a very energetic reaction with the antimony. This reaction pushed the top of the source rod upward 178-mm, split the clad around the source rod, and ultimately split the stainless steel storage thimble. When the source rod cladding was split, the primary sodium was allowed to flow past the antimony and the reaction continued until the antimony was consumed. It is believed that virtually all of the antimony went into solution in the primary sodium.

After the visual examination of the failed source rod and thimble, the two remaining source rods were removed from the reactor until a more satisfactory neutron source could be designed. In the interim, a temporary source made of solid beryllium was installed in the reactor. The main disadvantage of this source is that is has a half life to about 12.8 days because it depends on fission product decay gammas instead of the 60-d gamma decay half life of the antimony source.

Figure 4



CRACKED STORAGE THIMBLE

## 4.2 EBR-II Achievements

To balance the previous paragraphs on plant and operational problems, it is appropriate to mention some of the many significant achievements which have occurred at EBR-II. The following is a brief listing of the more significant achievements.

### 4.2.1 Plant Capacity Factor

Since 1975, EBR-II's annual plant capacity factor has averaged 70.5%. The best was 77.1%, achieved in 1980. Annual capacity factors could probably be increased by 5-10% if EBR-II were not operated as an experimental facility. See Table I for annual capacity factors.

### 4.2.2 Metal Fuel Experience

Twenty years of experience with metallic fuel have shown that this fuel performs exceptionally well. Burnup limits have been increased incrementally from an initial value of 1.0 at.% in 1964 to 8.0 at.% today. Experimental fuel specimens have been successfully irradiated to 18.5 at.% and preparations are being made to increase the existing burnup limit to 10 at.%. EBR-II's present fuel design has also demonstrated its resistance to failure under repeated, severe power transients by the successful completion of a qualification program which imposed 56 slow transients from 24 MWt to 62.5 MWt at a rate of 0.4 MW/s and 13 fast transients from 26 MWt to 62.5 MWt at a rate of 4 MW/s.

Also, in recent tests on two experimental subassemblies, fuel pins were operated at peak fuel/clad interface temperatures above 750°C for twelve hours without a cladding breach occurring (at 715°C eutectic formation begins between 316 stainless steel and uranium).

### 4.2.3 Experimental Irradiation Experience

Since EBR-II assumed its mission as an irradiation facility, it has been used to irradiate more than 13,000 specimens of various fuel, structural, and adsorber materials in 830 different experimental subassemblies.

### 4.2.4 Personnel Exposure to Radiation

Exposure of the operating and maintenance personnel has historically been kept at very low levels. This is due primarily to the fact that the major radiation concern in a sodium cooled reactor is $^{24}Na$ which has a 15 h half life. This relatively short half life allows ample time for decay before performing maintenance without greatly impacting plant factor. Over the last ten years, the average annual exposure to an operator or instrument technician has been about 90-100 mRem and to a maintenance technician, about 500 mRem.

### 4.2.5 Fission Product Release to the Environment

Radioisotope releases to the environment over the last seven years have averaged about 130 Ci/yr. This has been due primarily to fission gases which are released to the primary cover gas from breached fuel pins irradiated to and beyond cladding breach as part of the irradiation program. Xenon and krypton are removed from the cover gas by a cryogenic gas cleanup system which stores the fission gases for decay. If it were not for the fact that the cryogenic sump must be periodically dumped to a charcoal absorber, EBR-II would be a near-zero release facility. However, when the sump is dumped, the charcoal absorber bed releases long-lived $^{85}Kr$ to the atmosphere.

## 5.0 THE FUTURE

### 5.1 Programs

The present program of fuel irradiations and operational reliability testing is scheduled to continue through 1986. In addition to these programs, EBR-II is actively involved in testing of new components and materials is support of advanced liquid metal fast reactor designs and is also being considered for testing of major system components such as advanced-design steam generators.

## 5.2 Metal Fuels - A New Initiative

On the eve of EBR-II's twentieth anniversary, Argonne National Laboratory was developing an advanced Liquid Metal Reactor (LMR) concept that could promise improved, inherent safety characteristics, low capital cost, and economic fuel cycle closure. The concept, titled the Integral Fast Reactor (IFR), is based on four basic technical features. These features are: (1) liquid sodium cooling, (2) pool-type reactor configuration, (3) metallic-fuel core design, and (4) an integral fuel cycle with an on-site fuel cycle facility utilizing pyrometallurgical reprocessing and injection-cast fuel fabrication.

The IFR concept has a number of specific technical advantages that collectively address the potential difficulties facing the LMR today. The three main advantages are in the areas of fuel performance, reprocessing and fabrication, and safety.

The work scope and progress in the IFR concept are summarized below for each of the three main technology areas. This work involves not only the utilization of EBR-II for fuel development and demonstration, but also related development efforts by other ANL reactor program organizations.

### 5.2.1 Fuel Performance Demonstration

The basic physical properties of the IFR fuel and the fuel/cladding interactions over a range of conditions, compositions, and temperatures are to be established. This work is proceeding at the present time. Out-of-reactor experiments to establish the compatibility of the IFR fuel with advanced cladding materials, to characterize the distribution of the alloying elements within the fuel, to measure the thermal and physical properties of the fuel, and to establish calculational methods of modeling the fuel behavior, are all underway.

A second objective is to expand the IFR U-Pu-Zr fuel irradiation data base to provide the technical bridge between this alloy and the extensive EBR-II data base already in hand on a related metal fuel. The initial task was to quickly establish the capability for IFR U-Pu-Zr fuel fabrication at ANL-West, then to fabricate prototypical fuel subassemblies for irradiation in EBR-II and the Fast Flux Test Facility (FFTF).

The fuel fabrication capability, termed the Experimental Fuels Laboratory (EFL) is now fully operational. All three planned EBR-II lead test subassemblies have been fabricated. These lead assemblies are undergoing irradiation in EBR-II at the present time. The FFTF test planning and preparation is also progressing on schedule.

### 5.2.2 Pyrometallurgical Process Development

The concept uses a reprocessing technique that promises substantial advantages in cost, simplicity, and deployment flexibility. The objective of the task is to establish the chemical feasibility of the proposed processes for recycle of discharged core and blanket materials and for disposal of the fission-product waste. The major process steps are electrorefining, which will be used for the core material, and halide slagging, which will be used for the blanket. The work is to establish that product yields will be adequate, that fission-product removal will be sufficient, that container materials and process reagents can be specified and will perform as expected, and that the processes are adaptable to remote operations.

The first electrorefining experiments on a mixture of uranium and simulated fission products have already been successfully completed. Plutonium-based electrorefining and halide slagging work will start soon, as the construction of the necessary glove box facilities is nearing completion. Finally, demonstration of waste treatment for the resulting wastes, including metallic, chloride salt, and gaseous volatile waste is planned. At the end of this development, the process would be a reality, on very nearly practical scale.

### 5.2.3 Demonstration of Safety Characteristics

The new metallic fuel promises improved, inherent safety characteristics due to the properties of the fuel. Its high thermal conductivity results in very favorable reactivity feedback characteristics under loss-of-flow accident conditions so that the severity of such events is significantly reduced. Further, fission gases entrapped within the fuel alloy matrix itself may provide a self-dispersive mechanism that plays an important role in early termination of transient overpower accidents. The overall objective of this task is to provide the experimental data to validate unique inherent safety features and fully characterize the totality of safety features associated with metal fuel through detailed analysis, modeling, safety-related in-reactor tests, and out-of-reactor experiments.

## 6.0 SUMMARY

In addition to the invaluable contributions EBR-II has made in the area of steady-state and transient fuels testing, the safe and efficient operation of EBR-II for the last 20 years represents a significant portion of the existing U.S. LMFBR operating experience. It is expected that EBR-II will continue to contribute to this experience base for at least another 10 years since there is at present no apparent limitation due to any single component, plant system, or any identified situations that would limit the plant life to less than 30 years.

TABLE I

EBR-II PLANT CAPACITY FACTOR

| YEAR | CAPACITY FACTOR |
|---|---|
| 1970 | 58.0 |
| 1971 | 39.1 |
| 1972 | 46.9 |
| 1973 | 49.9 |
| 1974 | 58.7 |
| 1975 | 66.1 |
| 1976 | 76.9 |
| 1977 | 71.5 |
| 1978 | 72.8 |
| 1979 | 71.1 |
| 1980 | 77.1 |
| 1981 | 73.0 |
| 1982 | 62.3 |
| 1983 | 65.5 |
| 1984 | 69.1 |
| Average | 70.5 |