ANL-000

EBR-II

Sixteen Years Of Operation

ARGONNE
NATIONAL
LABORATORY
WEST

EBR-II

Sixteen Years of Operation

Argonne National Laboratory - West
P. O. Box 2528
Idaho Falls, Idaho 83401

May 1980

FOREWORD

Now in its sixteenth year of operation, EBR-II continues to serve the nation as a facility for testing fuels, materials, and instruments of interest to the breeder reactor community. Designed and built with the technologies of the 1950's and 1960's, EBR-II stands as a tribute to those scientists and engineers whose imagination, competence, and initiative have transformed a visionary concept into operational reality.

Born as a concept during the war years and nurtured under the "Atoms for Peace" era of the 1950's and 1960's, EBR-II was originally designed to establish the feasibility of metallic-fueled sodium-cooled breeder reactors for power-plant service and to demonstrate the feasibility of on-site fuel reprocessing techniques. By the mid-1960's all original objectives were met. As interest in fuel breeding increasingly developed, EBR-II became the nation's lead facility for the irradiation testing of fuels, materials, and instruments of interest from the viewpoint of more advanced systems.

During sixteen years of operation, numerous changes have been made to enhance the usefulness and flexibility of EBR-II as an irradiation-testing facility. Despite the diversity and complexity of its experimental program, EBR-II has operated over the past six years with plant capacity factors at least as high as those for commercial nuclear and fossil-fueled electrical generating plants.

Details that relate to the history of EBR-II, its construction and operation, and its many important contributions to the national breeder program are given in the pages that follow.

TABLE OF CONTENTS

TABLE OF CONTENTS (Contd.)

TABLE OF CONTENTS (Contd.)

LIST OF FIGURES

LIST OF TABLES

1.0 INTRODUCTION

The importance of nuclear fuel breeding is a matter of international significance. All major industrial nations are currently engaged in aggressive programs to develop and deploy fast breeder reactors as base-loaded electrical generation units and as a means for greatly enhancing the utilization of the world's uranium supplies. Of international prominence are operating systems such as PFR in the United Kingdom; BOR-60, BN-350, and BN-600 in the Soviet Union; Rapsodie and Phenix in France; and JOYO in Japan. More advanced systems in each of these countries are either under construction or in various stages of planning as, for example, the CFR in the United Kingdom, BN-1600 in the Soviet Union, Super-Phenix in France, and MONJU in Japan. West Germany and Italy are also actively engaged in the planning and construction of fast breeder reactors.

Efforts in the United States focus around three principal facilities: the FFTF (Fast Flux Test Facility) at Richland, Washington; the CRBR (Clinch River Breeder Reactor) at Clinch River, Tennessee; and the EBR-II (Experimental Breeder Reactor II) at Idaho Falls, Idaho. Of the three, only EBR-II is operational today. Startup of the FFTF in late 1979 or early 1980 seems likely. Although the design, engineering, and procurement of long lead-time components for the CRBR are well advanced, the fate of this facility remains clouded with uncertainty.

EBR-II is, by definition, a Liquid-Metal-Cooled Fast Breeder Reactor (LMFBR). It is cooled with molten sodium metal, its chain reaction is perpetuated with extremely energetic (fast) neutrons, and it was designed with the potential for breeding more fuel than it consumes. Today, EBR-II serves the nation as its lead facility for generating information needed for the design, construction, and operation of advanced breeder systems.

Among the many EBR-II achievements are the following: the generation of over one billion kilowatt-hours of electricity; the irradiation of over 10,000 specimens of fuel, structural, and absorber materials; the *in situ* testing of advanced instrumentation concepts; and the successful demonstration of an on-site, diversion-proof system for fuel reprocessing. The results of many tests and experiments coupled with over 15 years of operating and maintenance experience have contributed heavily to national and international FBR (Fast Breeder Reactor) technology. The information that follows is limited to that which provides the reader with descriptive and factual information concerned with the history, design, construction, operation, and accomplishments of one of the nation's most important reactor test facilities.

2.0 LOCATION

The EBR-II complex is located approximately 35 miles west of Idaho Falls, Idaho, at the southeastern corner of the Idaho National Engineering Laboratory (INEL). The INEL (formerly the National Reactor Testing Station) was selected in 1949 as a national site for centralizing the construction, operation, and testing of a wide variety of reactor concepts. Since the early 1950's, 52 individual reactor systems have been built and tested at the INEL. Of these, 17 are still operable, the remaining 35 have been phased out upon the completion of their missions. Major existing facilities are identified in a map of the INEL, Fig. 1.

EBR-II is operated by Argonne National Laboratory, Argonne, Illinois, for the USDOE under a contract among the USDOE Chicago Operations Office, the Argonne Universities Association, and the University of Chicago. Of approximately 5000 employees, 4200 are located at the main laboratory at Argonne, Illinois. The remaining 800 are located at the INEL site. Some 300 employees at the INEL and Illinois sites are assigned to the EBR-II project.

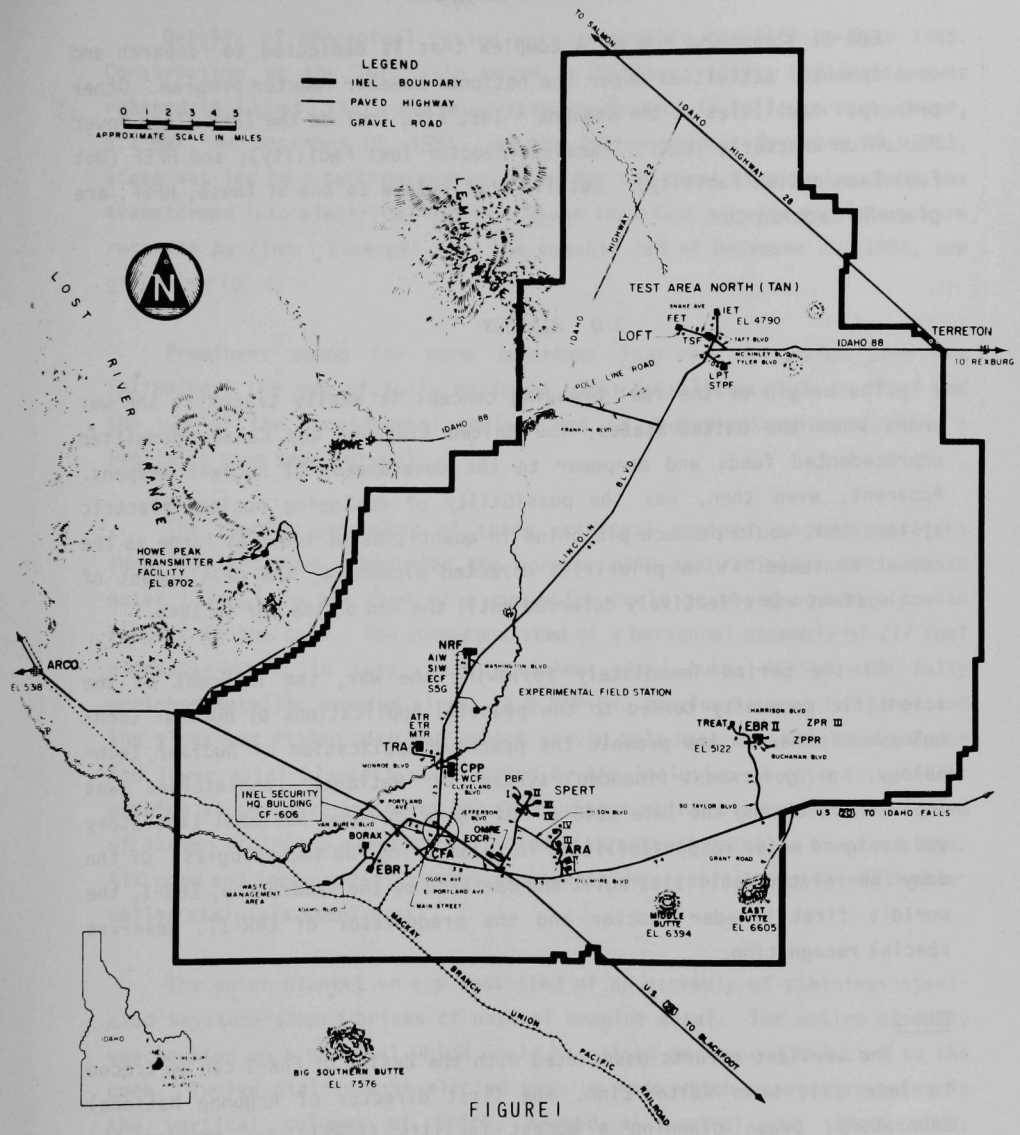


FIGURE I

Map of Idaho National Engineering Laboratory showing major existing facilities

EBR-II forms the hub of a complex that is dedicated to research and developmental activities under the national breeder reactor program. Other principal facilities at the Argonne - INEL site include the ZPPR (Zero Power Plutonium Reactor); TREAT (Transient Reactor Test Facility); and HFEF (Hot Fuel Examination Facility). Details that relate to one of these, HFEF, are given in Section 10.

3.0 HISTORY

The origin of the fuel breeding concept is easily traced to the war years when the United States, the United Kingdom, and Canada committed unprecedented funds and manpower to the development of nuclear weapons. Apparent, even then, was the possibility of designing nuclear-electric systems that would produce plutonium in quantities at least as large as the amount consumed. With priorities directed elsewhere, the development of such systems was effectively deferred until the end of the war in 1945.

In the period immediately following the war, the interest of the scientific community turned to the peaceful applications of nuclear technology. To hasten and promote the peaceful application of nuclear technology, a government-financed system of National Laboratories was implemented during the late 1940's. Of these, Argonne National Laboratory was assigned major responsibilities for pioneering FBR technologies. Of the many FBR-related facilities built and operated by the Laboratory, EBR-I, the world's first breeder reactor and the predecessor of EBR-II, deserves special recognition.

EBR-I

The earliest efforts associated with the design of EBR-I can be traced to late 1944 when Walter Zinn, the first director of Argonne National Laboratory, began planning a modest facility (EBR-I) for proving the validity of the breeding principle.

Details of conceptual design were reasonably complete by late 1945. Construction at the INEL site began in October, 1949. Criticality was reached in August, 1951, and the system was brought to full operating power, 1.1 MWt, on December 19, 1951. On the following day, December 20, 1951, steam was led to a turbine-generator and for the first time nuclear heat was transformed into electrical energy. Those important moments in history were recorded by Zinn. Excerpts from the console log of December 20, 1951, are given in Fig. 2.

Prominent among the more important features of design were the following: the use of fully enriched (93%) metallic uranium for fuel and the use of the liquid-metal coolant NaK (a sodium-potassium mixture, a liquid at room temperature).

The reactor consisted of three principal regions: a core, a light inner blanket that surrounded the core axially and radially, and a dense outer blanket in the form of a cup which could be moved vertically with respect to the core. The core consisted of a hexagonal assembly of 217 fuel elements which, in turn, were stainless steel tubes filled with fully enriched metallic uranium slugs. A 0.010-in. annulus between the tubes and the slugs was filled with NaK, which served as a heat-transfer bond. Upper and lower axial blankets were formed by the inclusion of natural metallic uranium slugs above and below the fuel. Inner radial blanket rods consisted of larger stainless steel tubes filled with natural metallic uranium slugs. All core and inner radial blanket components were contained within a double-walled stainless steel tank.

The outer blanket or cup consisted of an assembly of stainless-steel-clad keystone-shaped bricks of natural uranium metal. The entire assembly was mounted on a pedestal which could be raised or lowered relative to the core. Twelve stainless-steel-clad uranium rods which penetrated holes in the vertical columns of brick, served as control and safety rods. Controlled movement of the cup served as a coarse variation of reactivity and upon the receipt of a scram signal the cup dropped away from the core to reduce reactivity by approximately 6% $\Delta k/k$.

Dec 20th 9:50 a.m.
Reactor cooling system connected in

On vibrating reed
Open to close of hole goes
from 62 to 4½ ∓ 13.8 times

Pile levelled off at 250 kw
at about 10:40 a.m.
Steam turbine and Generator
started running 11 a.m.

At 11:30 am. the turbine was
shut off and power level raised
at 12:42 Turbine again turned on.
At 12:55 pm power raised to 410 kw

At 1:23 p.m. Load dissipators
from the Generator were
connected - Electricity flows from
atomic energy.

FIGURE 2

Excerpts from the console log of EBR–I on the day when nuclear power was born

A cutaway view of the reactor as it existed during the earlier loadings is given in Fig. 3. Coolant from an elevated supply tank flowed by gravity into an upper plenum, downward through the inner blanket region, and into a lower plenum, where the flow reversed and passed upward through the core and into an upper exit plenum.

A schematic layout of the plant is given in Fig 4. After leaving the outlet plenum, the coolant flowed by gravity to a primary-secondary (NaK to NaK) heat exchanger, then into a receiving tank in the basement. A pump, operated at a capacity slightly higher than the main coolant flow, returned the coolant to the gravity supply tank. An overflow system, connecting the supply tank to the receiving tank, guaranteed a constant delivery rate of coolant to the core and blanket.

Heat from the secondary NaK was removed in a steam generator which consisted of a water heater, boiler, and superheater. Water, in the form of a film, traveled downward through the heat exchanger tubes. Heat transferred by the counter-flowing secondary NaK generated steam which, after superheating, was used to drive a conventional turbine-generator. Approximately 200 kilowatts of electricity were generated -- enough to satisfy the needs of the EBR-I facilities.

In the course of its useful life (1951-1964), EBR-I operated with four different fuel loadings, each designed to establish specific bench marks of FBR technology. Descriptive information for each of the four loadings is given in Table I.

Among the most notable achievements of EBR-I were the following:

- The first generation of electricity from the fissioning of uranium.
- Proof that the operating characteristics of fast reactors are similar to those of thermal systems.
- Proof that a breeder reactor can produce more fuel than it consumes. (A conversion ratio of 1.01 ± 0.05 was established for the first uranium-fueled loading.)

- The first production of "super-pure" plutonium.
- The measurement of nuclear parameters needed for the design of more advanced LMFBR systems.
- A demonstration of the simplicity of working with liquid metal coolants.
- An evaluation of the effects of alloying materials on the behavior of metallic fuels.
- An evaluation of the effects of structural features on operational stability.
- The first plutonium-fueled reactor loading.
- Proof that a breeder reactor fueled with plutonium enjoys a higher breeding ratio than one fueled with uranium. (A breeding ratio of 1.27 ± 0.08 was established for the plutonium loading).

Following the completion of tests with the plutonium loading in 1964, EBR-I was shut down, placed in standby status, and in 1966 declared a National Historic Landmark under the stewardship of the United States Department of the Interior. The facility was opened to the public in June, 1975. Visitors may tour the facilities, located on U.S. Route 20 between Blackfoot and Arco, during the period from June 15 to September 15.

EBR-II

The success of EBR-I, coupled with the interest in fuel breeding under the "Atoms for Peace" program in the early 1950's, prompted studies of a larger metallic-fueled, sodium-cooled system which eventually became known as EBR-II. The initial objectives of EBR-II were two-fold: a feasibility demonstration of metallic-fueled, sodium-cooled, fast breeder reactors as power plants; and a demonstration of the applicability of pyrometallurgical techniques for the on-site reprocessing of spent fuel. In the early stages of planning, EBR-II was regarded as an evolutionary step in the ultimate development of commercial-size fast breeders.

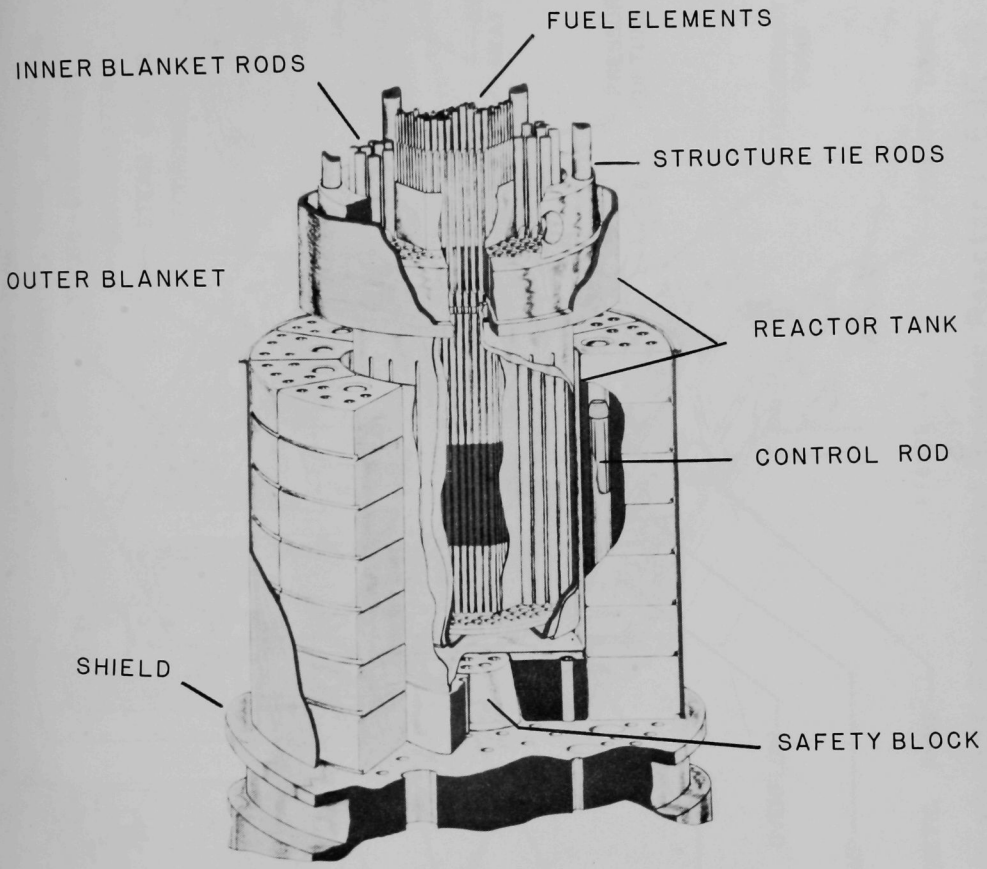


FIGURE 3

Cutaway view of

EBR–I, Mark I and Mark II loadings

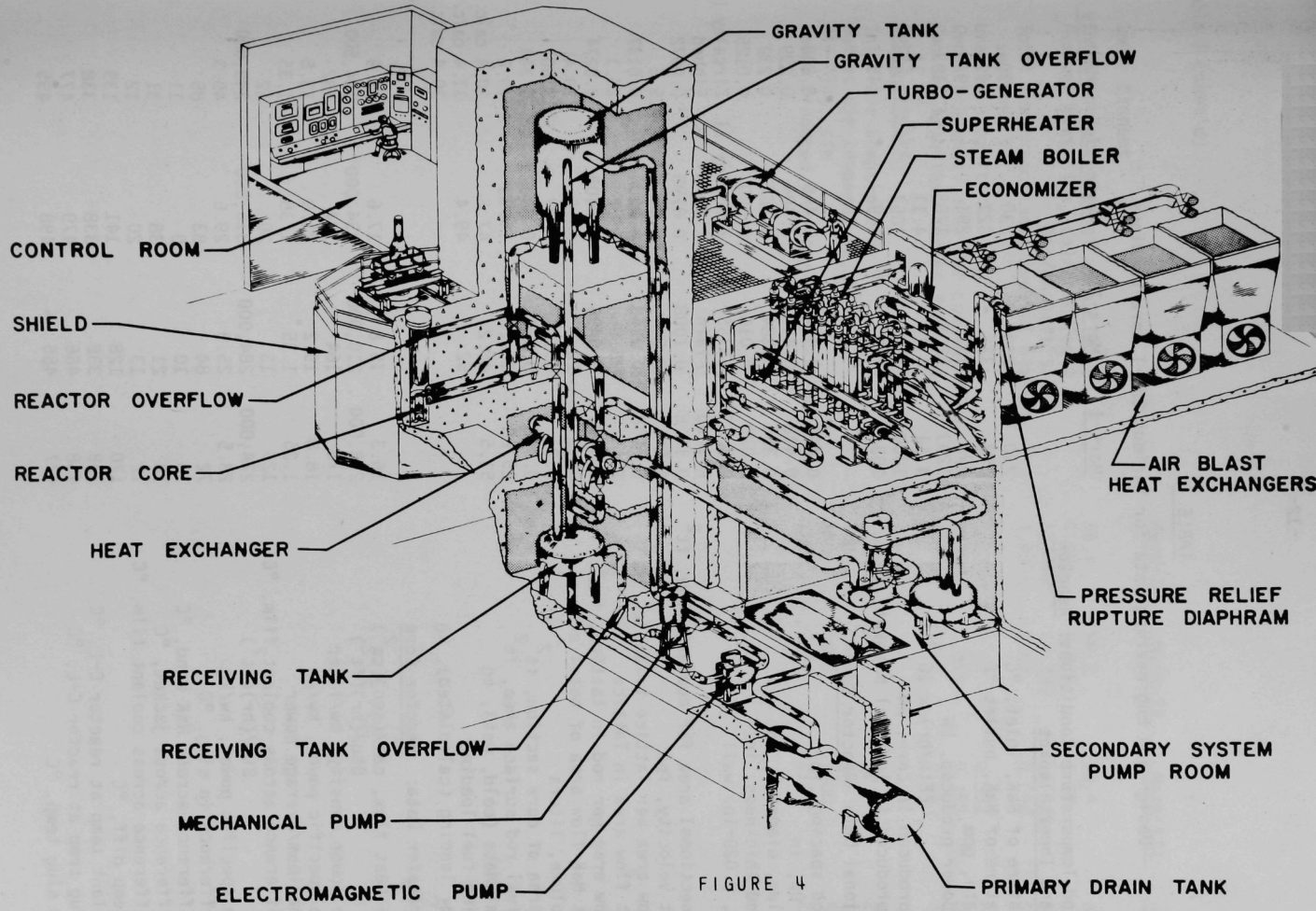


FIGURE 4

Schematic view of Experimental Breeder Reactor I (EBR-I) and its ancillary facilities

TABLE I

Design and Engineering Data for Various EBR-I Core Loadings

	Mark-I	Mark-II	Mark-III	Mark-IV
1. Flow and Temperature Conditions: Reactor Core and Inner Blanket				
Temperature of NaK, inlet, °C	230	230	230	230
Temperature of NaK, outlet °C	322	322	322	322
Flow rate, gpm	291	291	291	291
Total power produced, kW	1203	1203	1203	1203
(Btu/hr) x 10^{-6}	4.11	4.11	4.11	4.11
Power produced in core, kW	1000	1000	1054	1054
Power produced in internal blanket, kW	203	203	149	149
2. Dimensional Data: Reactor Core				
Fuel rod spacing, in.	0.494	0.494	0.450	0.348
Jacket OD, in.	0.448	0.448	0.404	0.299
Fuel slug, diameter, in.	0.364	0.384	0.364	0.232
NaK bond thickness, in.	0.020	0.010	0.000	0.0125
Jacket, 0.020-in. wall	347 SS Ribbed	347SS Plain	Zircaloy II Ribbed	Zircaloy II Ribbed
Cross-sectional area for coolant flow, ft^2	0.1008	0.1008	0.0795	0.0972
Coolant velocity, fps	6.5	6.5	6.85	5.6
NaK flow area per lattice triangle, $in.^2$	0.0248	0.0248	0.0204	0.0137
Percent flow area in lattice	23.5	23.5	23.2	26.1
NaK flow area per rod in lattice, $in.^2$	0.0496	0.0496	0.0408	0.0274
Percent NaK flow area of total area	28.2	28.2	26.5	29.4
Core Volume, liters	5.9	6.1	6.07	5.80
Total area of core section, ft^2	0.338	0.338	0.348	0.348
Total fuel rod surface area, ft^2	15.68	16.20	15.33	19.5
Critical Mass (cold, wet), kg	51.5	48.2	47.6	30.4 calc.
Operating fuel loading		52	49.4	31.4 calc.
Critical loading (calculated), kg				30.4 calc.
3. Heat Transfer Data: Reactor Core				
Average heat flux, cal/(sec)(cm^2)	16.3	15.8	17.6	13.9
Btu/(hr)(ft^2)	218,000	210,000	234,000	184,500
Average power density, kw/liter	170	164	174	182
Average specific power, kw/kg	18.8	19.2	21.9	33.5
Ratio maximum/average power	1.25	1.35	1.35	1.35
Temp difference across coolant film, °C	12	13	20	12
Btu/(hr)(ft^2)	274,000	284,000	316,000	249,000
Maximum specific power, kw/kg	23.5	25.9	29.6	45.3
Temp difference in slug, °C	72	84	83	85
Temp difference across NaK bond, °C	18	10	0	11
Temp difference across jacket, °C	18	21	38	31
Temp difference across coolant film, °C	12	13	20	12
Total Temp diff, °C	120	128	141	139
NaK coolant temp at reactor C-L, °C	338	338	338	338
Fuel slug temp at reactor C-L, °C	458	466	479	477
Maximum slug temp, °C	477	485	498	495

able I (cont'd)

	Mark-I	Mark-II	Mark-III	Mark-IV
Inner Blanket				
Total surface area of rods, ft^2	56.2	56.2	30.4	30.4
Average heat flux, cal/(sec)(cm^2)	0.928	0.928	1.25	1.25
Btu/(hr)(ft^2)	12,300	12,300	16,600	16,600
Total cross-sectional area of blanket section, ft^2	1.02	1.02	0.599	0.599
Uranium area ft^2	0.57	0.57	0.312	0.312
Cross-sectional area for coolant flow, ft^2	0.368	0.368	0.155	0.155
Coolant velocity, series flow, fps	1.77	1.77	3.51	3.51
Outer Blanket				
Inlet air temperature, °C	20	20	20	20
Outlet air temperature, °C	108	108	108	108
Air flow, cfm	5800	5800	5800	5800
Power produced in outer blanket, kw	222	222	222	222

Primary design features were initially addressed in a proposal to the USAEC in December, 1953. Also addressed at that time was the feasibility of building and operating on-site facilities for reprocessing discharged fuel. A more detailed document, one suitable for engineering evaluation, was issued in March, 1954. Federal funds were authorized in July, 1955. Site preparation began in October, 1957, and formal construction of the power plant, reactor plant, and cooling tower began in June 1958. Fabrication of the reactor vessel started in October, 1958 and in December, 1958 the containment building was completed. A rotary bridge crane was erected in the containment building in November, 1959, thus permitting the installation of the primary tank. Plans for the FCF (Fuel Cycle Facility) and Sodium Boiler Plant were finalized in March, 1959. Construction of these facilities began in July, 1959. By May, 1961, construction and installation work in the primary tank was essentially completed.

In late 1961, dry critical experiments were conducted. Other important activities conducted during this period included the prooftesting of various components and systems such as reactor instrumentation, fuel handling equipment, control rod drives, etc. During the tests, construction work continued on the secondary sodium and steam systems and the Fuel Cycle Facility.

In February, 1963 the tank was filled with 86,000 gal of sodium in preparation for startup. The approach to criticality began on October 30, 1963, and ended on November 11 with a fuel loading of 181.2 kg of U-235. Subsequent tests conducted under critical and subcritical conditions included the calibration of control and safety rods, and measurements of neutron source strength, subassembly worth, isothermal temperature coefficient, and neutron flux distribution in the core and blanket.

By June, 1964, all systems were ready for operation. The approach to power began on July 16, 1964, and a power level of 37.5 MWt was reached on October 13, 1964. Reactor power was later raised to 45 MWt on March 27, 1965, and to 50 MWt on August 26, 1969. A final increase in power to 62.5 MWt, the design level, was made on September 25, 1969. A more

comprehensive summary of construction highlights is given in Table II.

4.0 PLANT DESCRIPTION

Pool-Type Concept

A feature unique to the design of EBR-II is the pool-type concept, which is based on the complete submersion under molten sodium of the reactor core, reflector, blanket, neutron shield, primary pumps, primary piping, heat exchanger, and in-vessel fuel handling equipment. In neither of the two advanced American systems under construction or in planning status, i.e., the Fast Flux Test Facility and the Clinch River Breeder Reactor, respectively, is this feature repeated. The designs of these systems are based on the loop-type concept in which radioactive primary sodium is pumped from the core outlet to intermediate heat exchangers external to the primary tank. Information derived from the operation of EBR-II and pool-type systems in other countries such as the United Kingdom, France, and the Soviet Union, will provide a valuable base for comparing the relative merits of the two concepts.

Features that affected the decision to utilize the pool-type concept for EBR-II include the following:

- The regular conformation of the primary containment tank and the lack of nozzles and penetrations greatly simplify design, construction and inspection activities.
- All systems that contain primary (radioactive) sodium are located within the primary containment barrier. The effects of sodium leakage from primary components are effectively confined to the primary tank.
- The submersion of all primary components under constant temperature sodium greatly reduces problems of thermal stress.
- The effects of loss of pumped coolant flow are considerably mitigated, since the core and blanket will always remain covered with sodium.
- Loss of coolant is a virtual impossibility.

TABLE II

Chronology of EBR-II Construction

Event	Date
Completion of Preliminary Proposal and Feasibility Report	Dec. 3, 1953
Original Authorization of Funds	July 11, 1955
Award of Architect-Engineer Contract	Nov. 15, 1956
Completion of Safety Analysis Report	May 15, 1957
Procurement of Reactor Containment Vessel Authorized	Oct. 22, 1957
Site Preparation Begins	Oct. 23, 1957
Procurement of Long Lead-time Items Begins	April 1, 1958
Construction on Ancillary Facilities Begins	June 19, 1958
Fabrication of Primary Tank Cover Begins	Sept. 30, 1959
Fabrication of Reactor Vessel Begins	Oct. 17, 1958
Containment Vessel Completed	Dec. 1, 1958
Fabrication of Rotating Plugs Begins	May 1, 1959
Rotary Bridge Crane Installed	Nov. 10, 1959
Installation of Primary Tank Begins	Jan. 12, 1960
Fabrication of First Fuel Loading Begins	Apr. 4, 1960
Primary Tank Installation Completed	Apr. 15, 1960
Construction of Cooling Tower Completed	Aug. 23, 1960
Installation Effort in Reactor Building Completed	May 12, 1961
Fuel Arrives On Site	June 19, 1961
Sodium Arrives On Site (10 tank cars)	Aug. 11, 1961
Primary Heat Exchanger Arrives	Aug. 25, 1961
Approach to Dry Criticality Begins	Sept. 18, 1961
Dry Criticality Achieved 230.2 kg of ^{235}U	Sept. 30, 1961
Turbine-generator operated on 175 lb steam	Dec. 15, 1961
Turbine-generator Synchronized to INEL Loop	Mar. 7, 1962
Primary Tank High Temperature Dry Test Completed	April 11, 1962
Steam-generation system cleaned	Aug. 22, 1962
Sodium-boiler Plant Completed	Nov. 13, 1962
All Construction and Component Installation Completed	Dec. 26, 1962
Argon Cell, FCF, Purged with Nitrogen	Jan. 23, 1963
Primary Tank Filled with Sodium	Feb. 26, 1963

Table II cont'd

Event	Date
Initial Operation of Primary Pump	Apr. 17, 1963
Argon Cell, FCF, Filled with Argon	Aug. 6, 1963
Secondary System Filled With Sodium	Aug. 29, 1963
Precritical loading completed	Sept. 23, 1963
Wet Criticality Achieved (181.2 kg ^{235}U)	Nov. 11, 1963
Wet Critical Experiments Completed	Dec. 5, 1963
Combined Operation of Primary and Secondary Systems	Apr. 9, 1964
Low Power Experiments Begin	May 22, 1964
Approach to Power Begins	July 16, 1964
Reactor Operated at 30 MWt, Turbine Generator Synchronized With INEL Loop, 8 MWe	Aug. 13, 1964
First Increment of Spent Fuel Reprocessed in FCF	Sept. 1964
Reactor Power Increased to 45 MWt	Mar. 27, 1965
First Irradiation Experiments Installed in Core	May 1965
Reactor Power Increased to 50 MWt	Aug. 26, 1969
First Unencapsulated $(U\text{-}Pu)O_2$ Elements Installed	April 1969
First Xenon Tagged Irradiation Experiment Installed	April 1969
FCF Shut Down	April 1969
Reactor Power Increased to 62.5 MWt	Sept. 25, 1969

- The location of intermediate heat exchangers within the primary containment vessel eliminates the need for heavily shielded external facilities, e.g., pipe galleries, equipment cells, etc.

Plant Layout

The EBR-II plant consists of three principal facilities which house the reactor, steam generation system, and power plant. Each of these is depicted in Fig. 5. Heat generated in the core is transferred to the primary sodium coolant which is pumped through the core, into a primary-to-secondary heat exchanger, and discharged directly to the sodium-filled primary tank. A cutaway view that illustrates the relationship of primary tank components is given in Fig. 6. The heated secondary coolant is pumped from the heat exchanger to the sodium-boiler building where steam is produced and superheated. The superheated steam, in turn, is used to drive a turbine-generator located in the power plant building. Condensate along with makeup water is returned to the sodium boiler building for steam generation. After losing its heat to the steam generation system, secondary coolant is pumped back to the heat exchanger in the reactor tank to complete the cycle. In effect, the reactor is a 62.5-MW source of heat which is used to provide steam for electrical power generation.

A central control room, located on the upper floor of the power plant building serves as a "nerve center" for monitoring and controlling the reactor plant and its ancillary systems. Additional facilities, not shown in Fig. 5, consist of HFEF/S and HFEF/N where discharged fuel and irradiation experiments are processed for ultimate disposition. The relationship of EBR-II plant facilities with other ANL-W facilities is illustrated photographically in Fig. 7. A more detailed and functional description of the EBR-II plant is given below. For convenience the discription begins with the reactor itself. Subsequent descriptions follow a format in which components and systems are addressed in the approximate order of their locations with respect to the core.

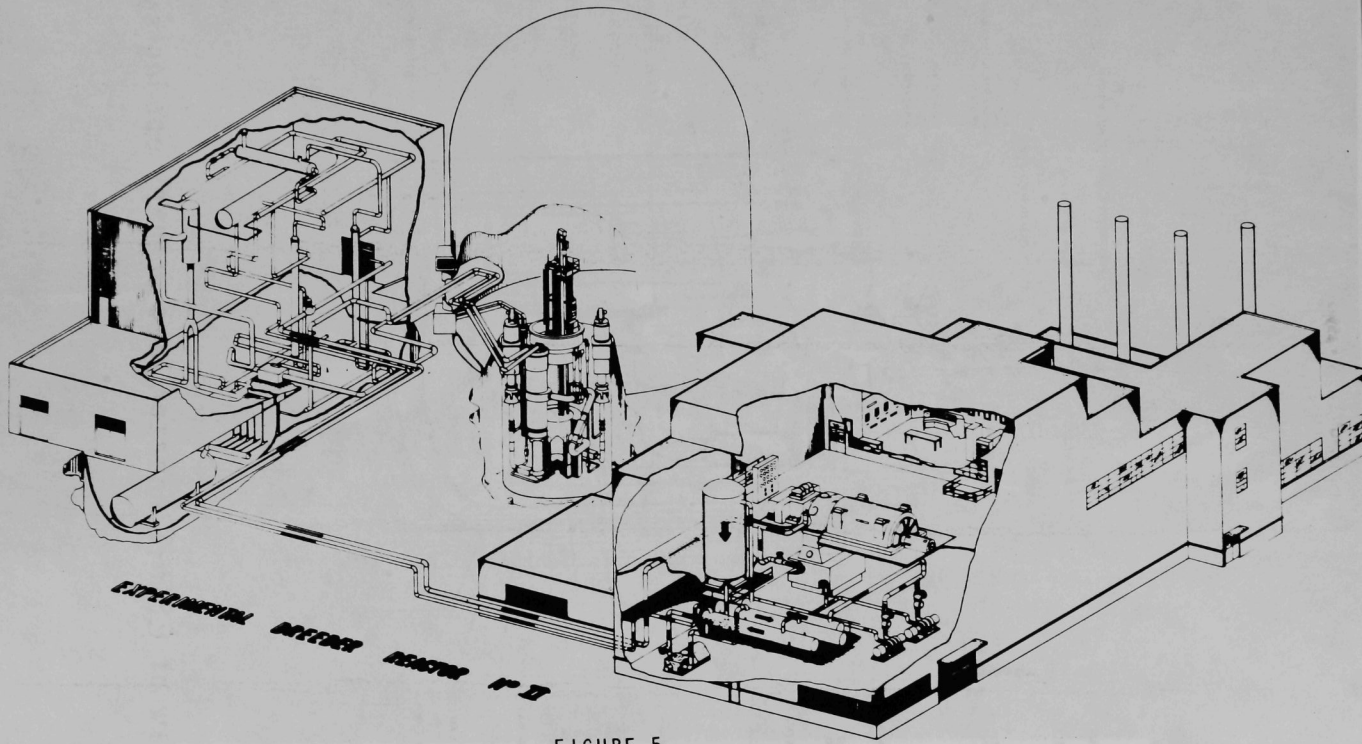


FIGURE 5

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

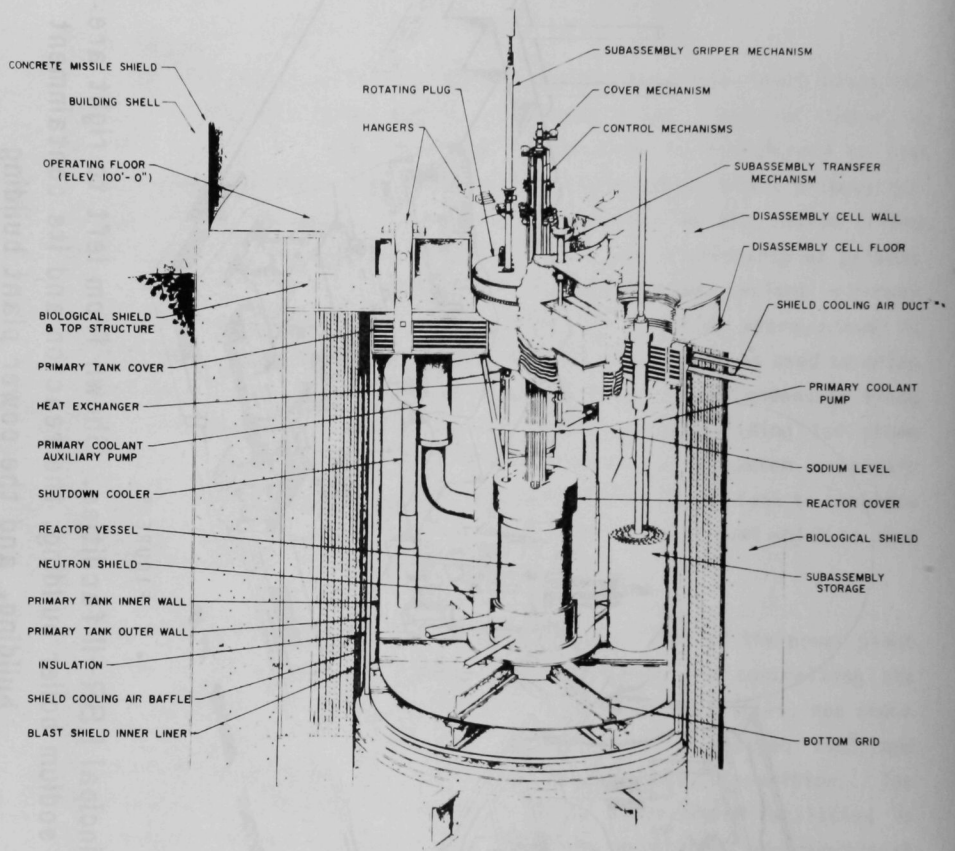


FIGURE 6

Cutaway view of EBR–II reactor and primary cooling system

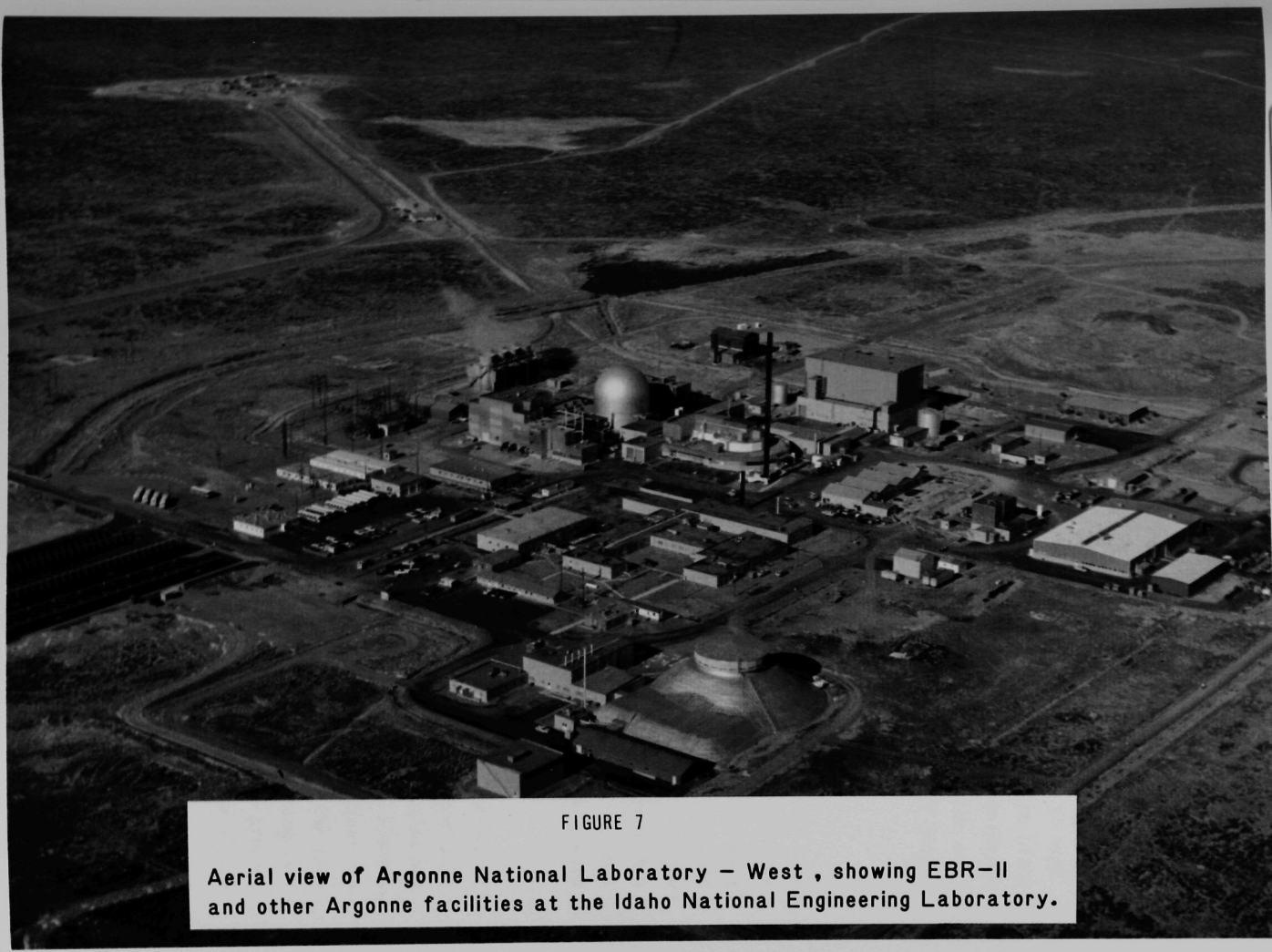


FIGURE 7

Aerial view of Argonne National Laboratory – West , showing EBR–II and other Argonne facilities at the Idaho National Engineering Laboratory.

Reactor

The reactor portion of the EBR-II plant consists of four principal subsystems: the grid plenum assembly, the reactor vessel, the reactor vessel cover, and the reactor core (see Figs. 8 and 9). Collectively, these systems form a right cylinder approximately 8 ft in diameter and 13 ft high. The entire assembly is bolted and welded to a 1.5-in. thick leveling plate which, in turn, is attached to reinforcing structural members at the bottom of the primary tank. The assembly is located at the bottom center of a double-walled tank 26 ft high and 26 ft in diameter. Sodium at 700°F covers the assembly to a depth of 10 ft.

Grid Plenum Assembly

The grid plenum assembly consists of upper and lower grid plates that are interconnected through a system of stainless steel tubes. A cylinder of stainless steel, which surrounds the grid plenum assembly, serves as a support for the grid plates and as a seal between the pumped sodium flow and the bulk primary sodium. By means of a system of high pressure and low pressure inlet ports and appropriate baffles inlet, coolant is divided into two streams: one which flows upward under high pressure through the core and inner blanket and the other, under lower pressure, which flows upward through the outer blanket.

The central portion of the lower grid plate consists of a series of concentric hexagonal steps which cover coolant inlet holes in the pole pieces of core subassemblies. Near the center of the core, fewer holes are covered. Conversely, near the outer edge of the core and inner blanket region, more holes are covered. This arrangement serves two purposes: to maintain a relatively uniform axial temperature gradient across the subassemblies and to conserve pumped coolant flow. A system of orienting bars at the underside of the lower grid and different-sized grid holes prevent the inadvertent insertion of a blanket subassembly in the core or a core subassembly in the blanket region.

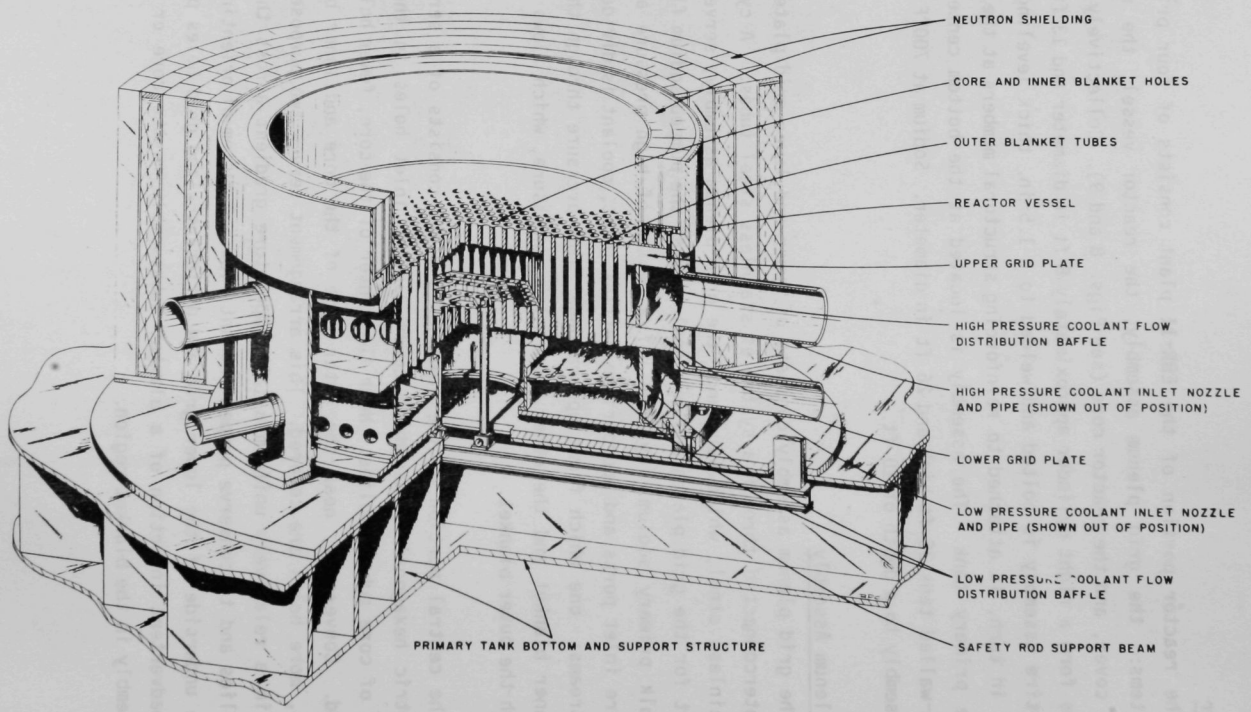


FIGURE 8

EBR–II grid plenum assembly

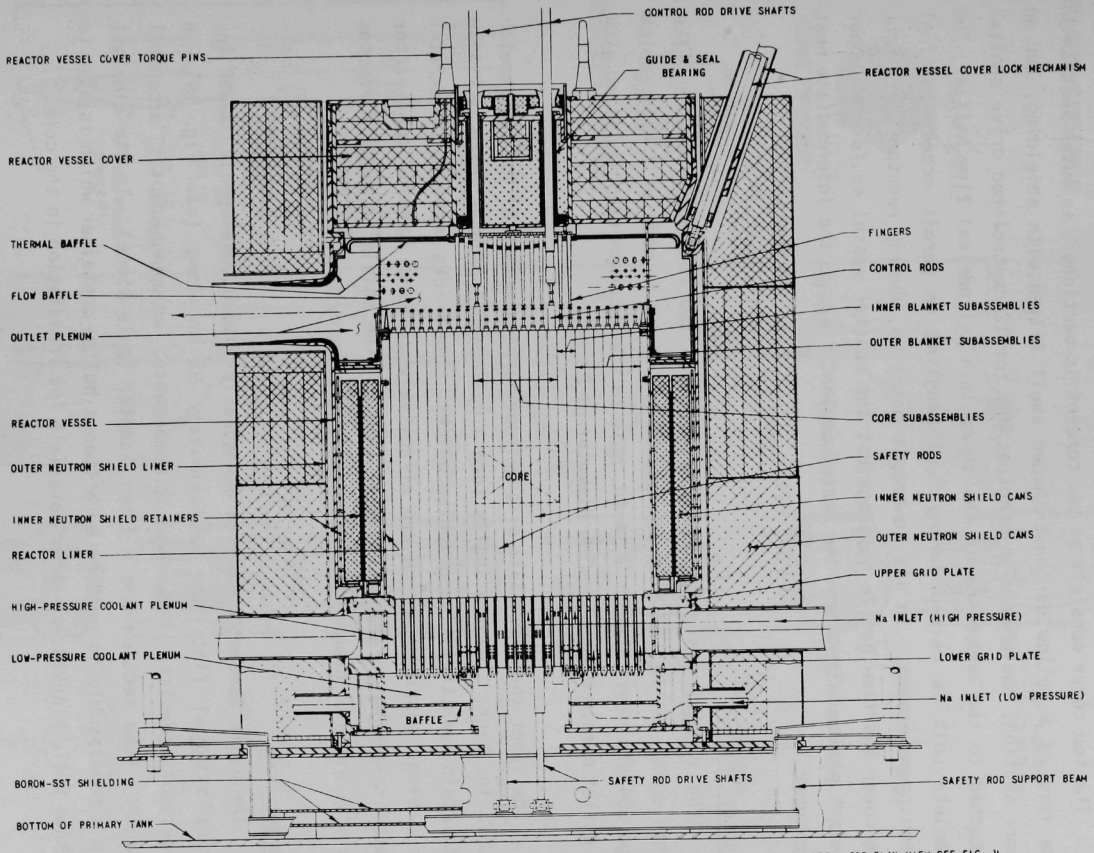


FIGURE 9

NOTE: FOR PLAN VIEW SEE FIG. 4

EBR–II reactor vessel assembly

Reactor Vessel Top Cover

The top cover consists of two concentric sections: an outer section in the form of a hollow doughnut-shaped shell filled with shielding, and an inner section, equipped with penetrations for 12 control-rod drive shafts. Attached to the underside of the cover is a cylindrical flow baffle. The complete unit is raised during fuel handling to permit access for fuel handling equipment. In its lowermost position, during reactor operation, the cover and flow baffle nest against the reactor vessel to form an upper coolant plenum. From here the heated coolant flows to the intermediate heat exchanger.

Reactor Vessel

The reactor vessel assembly consists of two concentric shells. The outer shell, 91-in. in diameter and 90-in. high, is made of 3/4-in.-thick type 304 stainless steel. A 14-in. hole in the upper portion is fitted with an outlet nozzle with an inside diameter of 13 in.

The inner shell consists of a cylinder approximately 67 in. in diameter and 71 in. high. The two shells form a cylindrical annulus which is filled with stainless-steel-clad graphite bricks. The bricks serve as a moderator for leakage neutrons, which are effectively captured outside the reactor vessel in a borated-graphite radial shield. The shielding greatly reduces the activation of components located in the primary tank.

Reactor Core

The basic loading unit is a hexagonally shaped subassembly, 2.290 in. across external flats and approximately 92 in. long (see Fig. 10). A subassembly consists of three basic components: an upper adapter, a central hexagonal tube section and a lower adapter (pole piece). In its simplest form a subassembly is nothing more than a hollow container which is used to package fuel, blanket, or other materials for irradiation in the core.

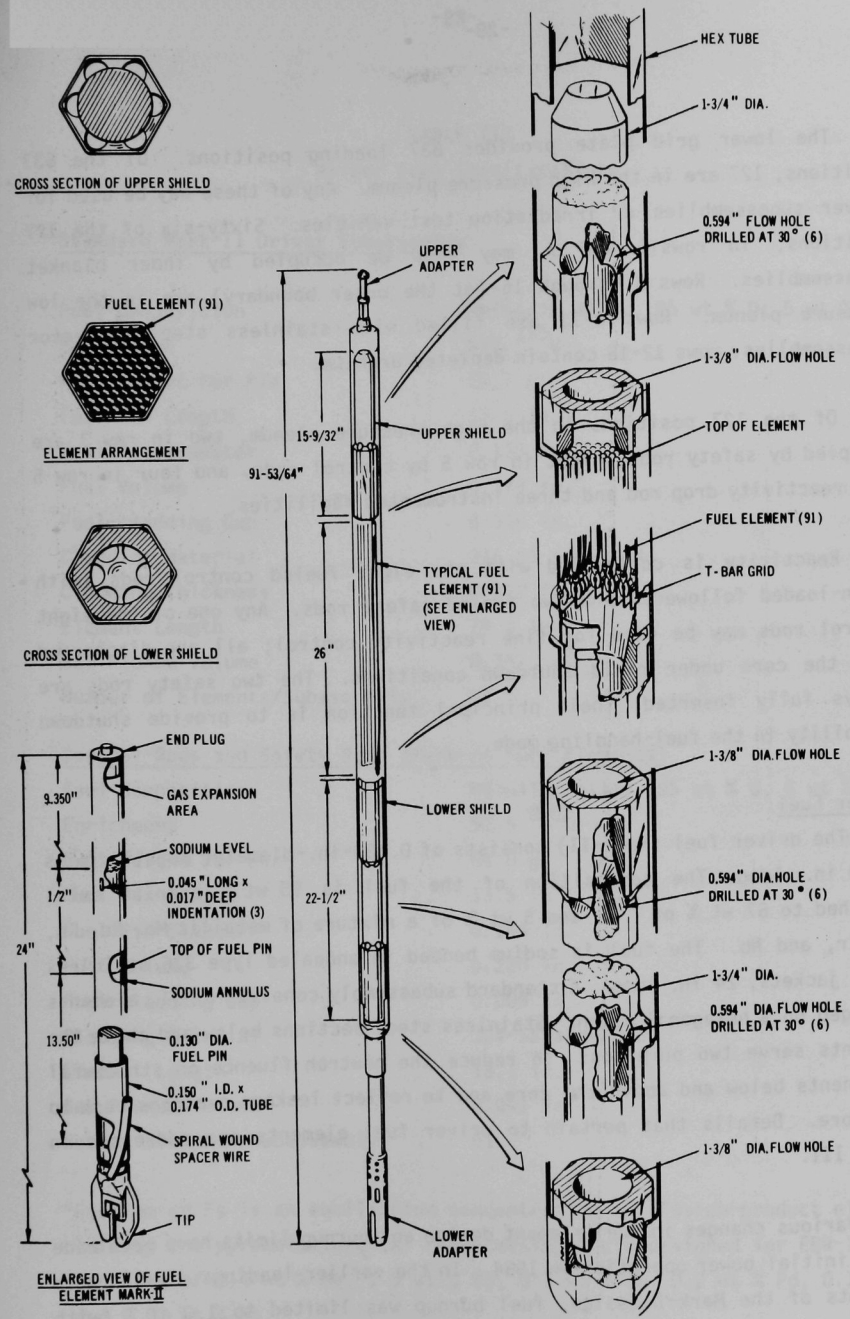


FIGURE 10

Components of an EBR–II Mark–II driver fuel subassembly

The lower grid plate provides 637 loading positions. Of the 637 positions, 127 are in the high pressure plenum. Any of these may be used for driver subassemblies or irradiation test vehicles. Sixty-six of the 127 positions, in rows 6 and 7, may also be occupied by inner blanket subassemblies. Rows 8 through 16 (at the outer boundary) are in the low pressure plenum. Rows 8-11 are filled with stainless steel reflector subassemblies, rows 12-16 contain depleted uranium.

Of the 127 positions in the high pressure plenum, two in row 3 are occupied by safety rods, eight in row 5 by control rods, and four in row 5 by a reactivity drop rod and three instrumented facilities.

Reactivity is controlled with the eight fueled control rods (with boron-loaded followers) and two fueled safety rods. Any one of the eight control rods may be used for fine reactivity control; all are discharged from the core under rapid shutdown conditions. The two safety rods are always fully inserted; their principal function is to provide shutdown capability in the fuel-handling mode.

Driver Fuel

The driver fuel (Mark-II) consists of 0.130-in.-diameter metallic pins 13.50 in. long. The composition of the fuel is 95 wt % uranium metal enriched to 67 wt % of ^{235}U and 5 wt % of a mixture of metallic Mo, Ru, Rh, Pd, Zr, and Nb. The fuel is sodium bonded in annealed Type 316 stainless steel jackets, 24 in. long. 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. Details that pertain to driver fuel elements are addressed in Table III.

Various changes in fuel element design and burnup limits have been made since initial power operation in 1964. In the earlier loadings, fueled with elements of the Mark-I design, fuel burnup was limited to 1.0 at.% (with respect to heavy atoms). Basic changes were made in 1966. The fuel column

TABLE III

Driver Fuel Specifications

Standard Mark-II Driver Subassembly

Fuel Composition	Metallic Alloy, 95 wt % U, 5 wt % Fs*
Enrichment	67% ^{235}U
Fuel Weight Per Pin	51.7 g
Fuel Pin Length	13.5 in.
Fuel Pin Diameter	0.130 in.
Fuel Volume	0.179 $in.^3$
Fuel-cladding Gap	0.010 in.
Cladding Material	316 SS annealed
Cladding Thickness	0.012 in.
Element Length	24.1 in.
Gas Plenum Volume	0.147 in^3
Number of Elements/Subassembly	91

Control-Rods and Safety Rods (Mark-IA Fuel Elements)

Fuel Composition	Metallic Alloy, 95 wt % U, 5 wt % Fs
Enrichment	52.5 ^{235}U
Fuel Weight Per Pin	64.0 g
Fuel Pin Length	13.5 in.
Fuel Pin Diameter	0.144 in.
Fuel Volume	0.220 in.
Fuel-cladding Gap	0.006 in.
Cladding Material	304-SS annealed
Element Length	18.1 in.
Gas Plenum Volume	0.041 in^3
No. of Elements/Subassembly	61

*Fissium or Fs is an equilibrium concentration of fission-product elements left by the pyrometallurgical reprocessing cycle designed for EBR-II. It consists of 2.4 wt % Mo, 1.9 wt % Ru, 0.3 wt % Rh, 0.2 wt % Pd, 0.1 wt % Zr, and 0.01 wt % Nb.

was shortened, the ^{235}U enrichment was increased, the gas plenum was enlarged and the pin restrainers were modified. At that time, burnup limits were increased to 1.2 at.%. Subsequent increases were made in the burnup limits, to 1.5 at.% and later to 1.8 at.% in 1969, and finally to 2.6 at.% in 1975. At the present time the use of Mark-IA elements is limited to control and safety rods. All other fuel subassemblies contain elements of an improved design designated as the Mark-II. Burnup for Mark-II elements is limited to 8.0 at.%. Additional information relating to Mark-II fuel elements is given below.

A listing of typical (average) temperatures for fuel, cladding, and coolant at the top of the core is given in Table IV. Fission rates, neutron flux, and other related data are summarized in Table V.

Other Subassemblies

Under typical operating conditions approximately 60 of the 127 positions in the high pressure (core) region are occupied by driver-fuel subassemblies. The remaining positions are filled with reflector, control-rod, safety-rod, and experimental irradiation subassemblies. The resulting complex configuration is illustrated in Fig. 11, which identifies various subassembly types for Run 89A (June, 1977). Information that pertains to the various subassemblies located in the core, rows 1-7, and in the first row of the stainless steel reflector, row 8, is summarized in Table VI. In general, positions in rows 8-11 are normally occupied by stainless-steel-filled reflector subassemblies, whereas rows 12-16 are occupied by blanket subassemblies. The latter contain 19 stainless-steel-clad, sodium-bonded, depleted uranium elements 55 in. long and 0.433 in. in diameter.

Heat Removal

Essentially all power production takes place in the core and inner blanket region. A small fraction of the total power, approximately 5%, is generated in the outer blanket. Although blanket power tends to increase with operating time, as plutonium accumulates, the periodic replacement of spent blanket subassemblies tends to keep the core/blanket power ratio constant. Each region is cooled separately by upward parallel flows of

TABLE IV

Average Temperatures for Mark-II Fuel and Cladding at Top of Fuel Column*

Row Number	Fuel Pin Center	Fuel Pin Edge	Inner Cladding	Outer Cladding	Coolant Temp.
1	1052°F	927°F	903°F	880°F	849°F
2	1038	919	896	874	844
3	1090	964	940	916	883
4	1079	972	952	832	903
5	1094	993	974	954	926
6	1100	1010	993	975	950
7	1060	977	961	945	922

*Values calculated for a typical run in the period, 1976-1979.

TABLE V

Physics Data

Run 89A, Fig. 11

Fission Rate, fissions/g-sec

	^{235}U	^{238}U	^{239}Pu	^{10}B (captures)
Row 1	9.6×10^{12}	0.61×10^{12}	11.1×10^{12}	1.8×10^{14} capt/g-sec
Row 2	9.4	0.57	10.8	1.7
Row 3	9.2	0.57	10.5	1.7
Row 4	8.7	0.53	10.0	1.6
Row 5	8.2	0.52	9.4	1.5
Row 6	7.3	0.47	8.4	1.4
Row 7	6.3	0.33	7.0	1.4
Row 8	5.6	0.15	5.8	1.6

Neutron Flux, neutrons/cm^2-sec

	Total Neutron Flux	Neutron Flux >0.111 MeV
Row 1	2.7×10^{15}	2.3×10^{15}
Row 2	2.6	2.3
Row 3	2.5	2.2
Row 4	2.4	2.1
Row 5	2.3	1.9
Row 6	2.0	1.7
Row 7	1.6	1.4
Row 8	1.3	0.9

Other	Power Density in Driver Fuel	%Power by Rows	Gamma Heating Rate in Iron
Row 1	0.18 w/g of U	1.3%	5.9 w/g
Row 2	0.18	4.1	5.4
Row 3	0.18	10.8	5.3
Row 4	0.17	11.1	5.0
Row 5	0.16	19.0	5.0
Row 6	0.14	26.7	4.7
Row 7	0.13	23.0	3.4

Table V (cont'd)

Maximum Power Density in Driver Fuel	854 kW/SA
Average Power Density in Driver Fuel	470 kW/SA
Total Power from Driver Fuel	41.5 MWt
Total Power from Irradiation Experiments	18.5 MWt
Power from Reflector and Blanket	2.5 MWt

Fissile - Fertile Inventory

^{235}U content of core (Rows 1-7)	244 kg
^{238}U and ^{240}Pu content of Core (Rows 1-7)	139 kg
^{239}Pu and ^{241}Pu content of core (Rows 1-7)	13.9 kg
^{235}U content of blanket (Rows 11-16)	35.8 kg
^{238}U content of blanket (Rows 11-16)	167,000 kg
^{239}Pu content of blanket (Rows 11-16)	68.4 kg

Maximum Heat Flux

Driver Fuel	5.5×10^5 Btu/hr-ft^2
Experimental Mixed (U-Pu)C Element	12.0×10^5 Btu/hr-ft^2
Experimental Mixed (U-Pu)O_2 Element	7.9×10^5 Btu/hr-ft^2

Maximum Irradiation Exposures

Experimental (U-Pu)O_2 Element	20.5 at.% burnup
Experimental (U-Pu)C Element	12.3 at.% burnup
Experimental (U-Pu)N Element	9.5 at.% burnup
Mark-II Driver Fuel Element (Encapsulated)	16.4 at.% burnup
Structural Materials	1.7×10^{23} nvt fluence
Control Materials (absorbers)	7×10^{21} captures/cm^3

Other

Prompt Neutron Lifetime	7×10^{-6} sec
Delayed Neutron Fraction	0.00682
Power Coefficient of Reactivity, Prompt	28×10^{-6} $\Delta k/k/MW$
Power Coefficient of Reactivity, Delayed	25×10^{-6} $\Delta k/k/MW$

Table VI

Subassembly Inventory

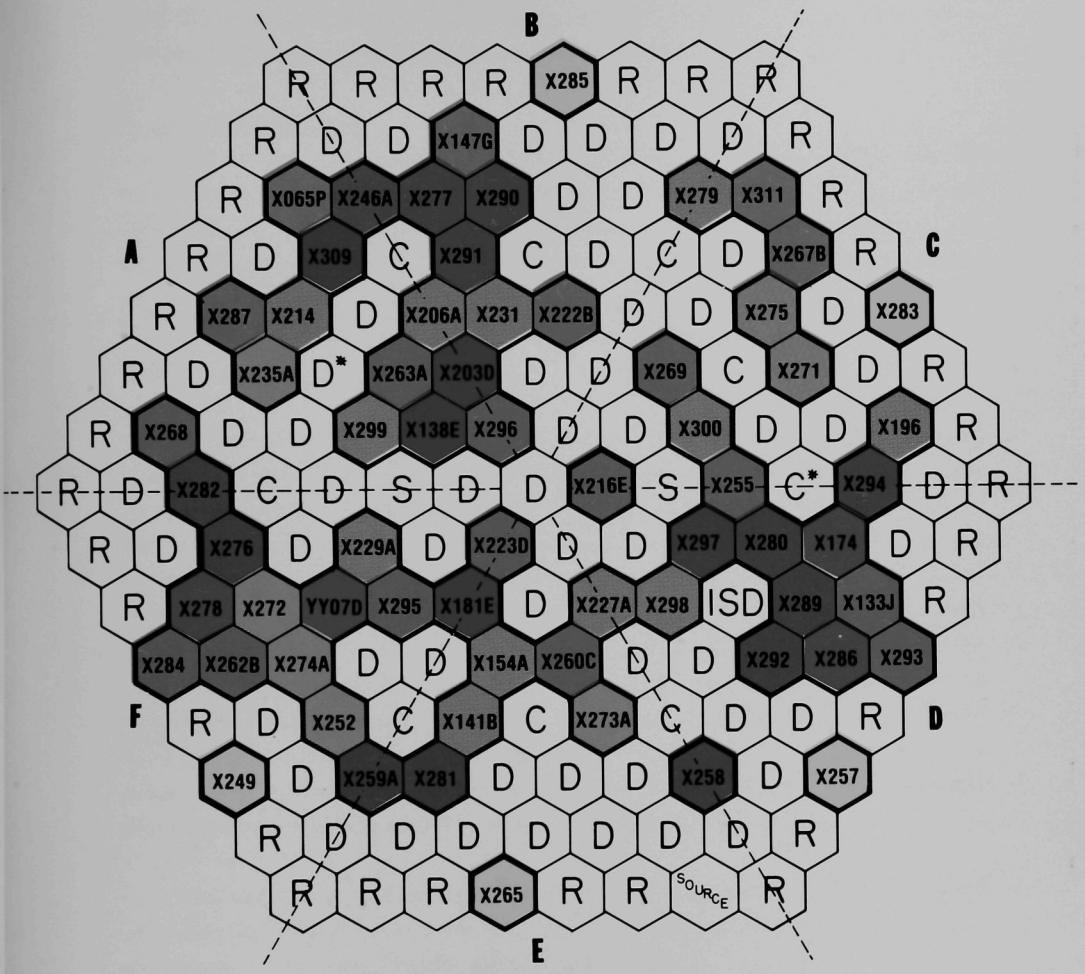
Rows 1-8, Run 89A, June 1977

		Positions
Standard Mark-II Driver Subassemblies		57
Safety Rods (Mark-IA Fuel Elements)		2
Control Rods (Mark-IA Fuel Elements)		8
Other Control Rod Positions		4
INCOT Driver Fuel Assembly		
Stainless Steel Drop Rod for Kinetics Monitoring		
INSAT Dummy Subassembly		
INCOT Experiment	32 creep specimens	
Irradiation Experiments Row 1-8,		64
Mixed (U-Pu)O2 Fuel Elements	17 Subassemblies, 647 specimens	
Mixed (U-Pu)N Fuel Elements	2 Subassemblies, 25 specimens	
Mixed (U-Pu)C Fuel Elements	11 Subassemblies, 222 specimens	
Metallic Fuel Elements	10 Subassemblies, 910 specimens	
Structural Materials	19 Subassemblies, 151 specimens	
Absorber Materials	5 Subassemblies, 20 specimens	
Positions in Row 8 (Exclusive of Irradiation Experiments)		
Stainless-steel Reflector Subassemblies		33
Antimony-beryllium Neutron Source		1

EBR-II REACTOR -- GRID LOADING DIAGRAM

RUN NO. 89

MAXIMUM EXPERIMENTAL LOAD TO DATE



START OF RUN - JUNE 19, 1977

- STRUCTURAL
- ADVANCED FUELS
- MIXED OXIDE FUELS
- METAL FUELS
- ABSORBER
- RBCB
- MISC.

D* - INCOT DRIVER
D - STANDARD DRIVER
C - CONTROL ROD
S - SAFETY ROD
C* - STAINLESS STEEL ROD
ISD- INSAT DUMMY
R - STAINLESS STEEL REFLECTOR

primary coolant. The two streams merge and mix in a plenum immediately above the reactor and beneath the reactor vessel cover. From this point on, heat dissipation proceeds via three principal systems: the primary sodium system, the secondary sodium system, and the steam system.

Primary Sodium System

Two centrifugal pumps, each rated at 4500 gpm, take suction from the bulk sodium in the primary tank and discharge sodium to the lower plenum (see Fig 12). Flow from the pump outlets is split into two streams: one which enters the high pressure plenum at approximately 50 psi, and the other which enters the low pressure plenum at approximately 17 psi. Throttle valves in the two low pressure delivery systems are used for pressure reduction.

Sodium flows upward through the core and blanket to the upper plenum, where it leaves the reactor through a single 13-in.-ID outlet pipe and flows through an auxiliary dc electromagnetic pump. From here the stream flows to the intermediate heat exchanger, where the heat from the thermally hot and radioactive primary system is transferred to the secondary sodium system. Primary coolant enters the heat exchanger at 883°F and is discharged directly to the bulk sodium at 695°F. Principal radioactive species in the primary sodium systems are ^{24}Na and ^{22}Na. Under sustained operating conditions, the activity levels of these species approach 3.0 mCi/g and 0.13 μCi/g, respectively. The secondary sodium, on the other hand, is essentially radioactively inert. Na-22 is missing and the ^{24}Na component may approach a maximum activity level of 40 nanocuries per gram.

The auxiliary pump operates continuously. It serves solely as a means of removing decay heat from the core in the event of a primary pump coastdown. The pump, rated at 500 gpm, operates on a rectifier-battery system which guarantees flow even if all electrical power should be lost.

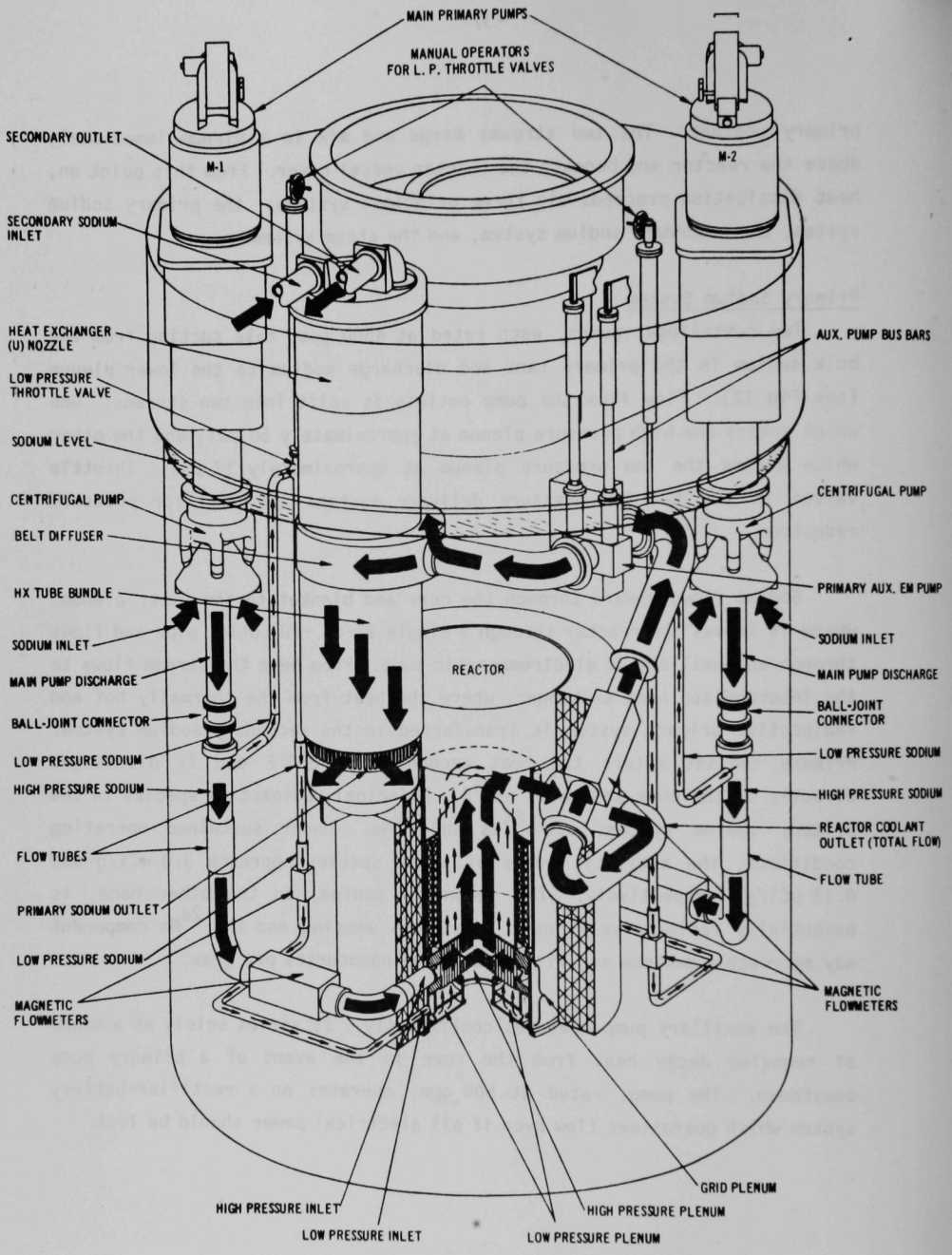


FIGURE 12

EBR–II primary cooling system

Reactor Containment Building

The reactor containment building is a gas-tight shell which houses the reactor, primary tank, shielding, and all ancillary primary components. The shell, usually referred to as the reactor containment building, consists of a carbon steel cylinder, 80 ft in diameter, up to one inch thick, with a hemispherical top and a semi-ellipsoidal bottom (see Fig. 13). Reinforced concrete lines the inside of the cylindrical portion of the shell to a depth of 12 in. Five inches of reinforced concrete line the inside of the hemisperical dome.

The shell was designed to withstand an internal pressure of 24 psig and a maximum leak rate of 1000 ft^3/day under a pressure differential of 20 psig. Three airlocks provide the means for moving equipment and personnel into and out of the reactor building without violating building containment. All areas of the reactor building are accessible during normal reactor operation.

Secondary Sodium System

The nonradioactive secondary sodium system is isolated from the radioactive primary sodium at the intermediate heat exchanger (IHX). Secondary sodium under pumped flow of 5000 gpm enters the IHX at 588°F and leaves at 875°F. It then flows to the sodium boiler building (see Fig. 14) where it flows upward through two superheaters and downward through eight water evaporators. A schematic view of an evaporator-superheater pair is given in Fig. 15. The streams from the lower portions of the evaporators then flow to a surge tank located at the highest point in the system. Sodium is pumped from the surge tank back to the IHX to complete the loop. All portions of the secondary sodium system are located at elevations higher than the level of primary sodium. This feature prevents the entrance of primary sodium into the secondary system in the event of an IHX leak between the two systems and provides the capability for natural circulation in the secondary system.

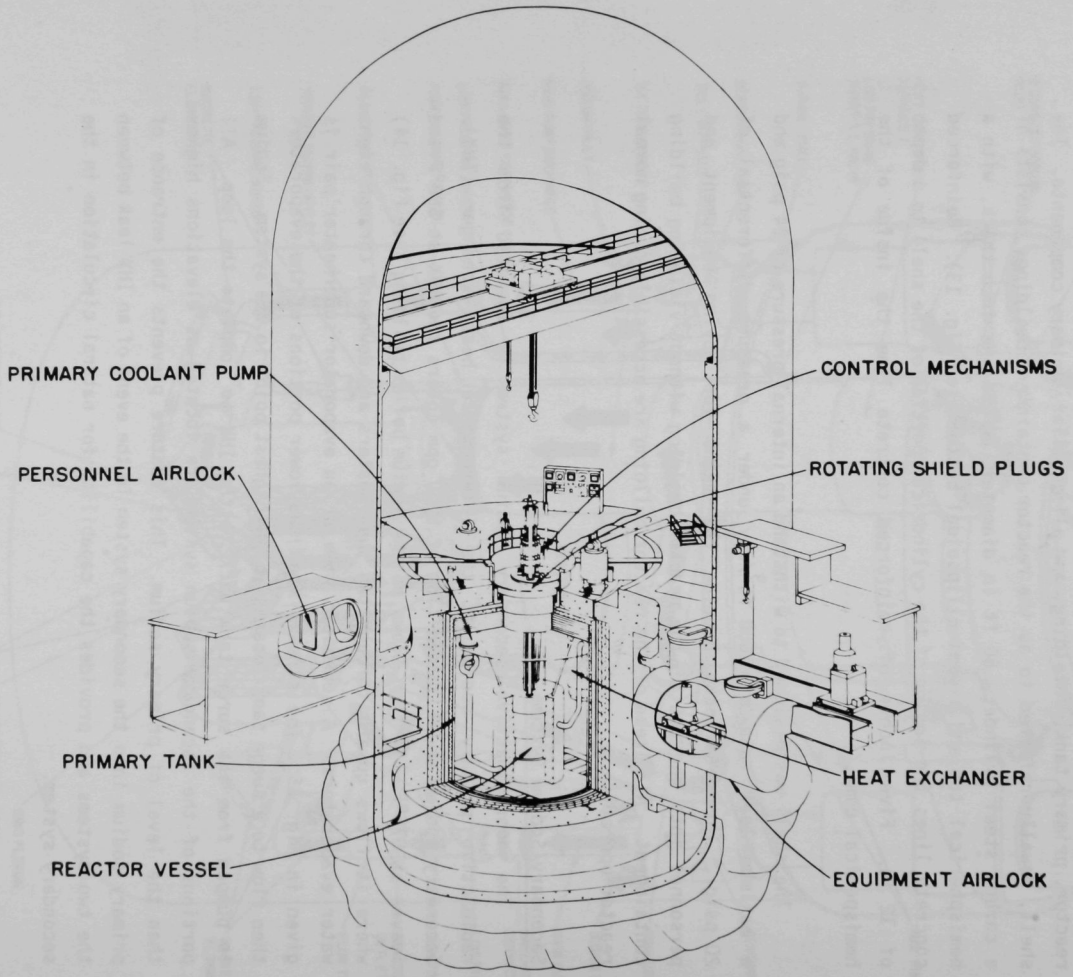


FIGURE 13

EBR–II reactor containment building

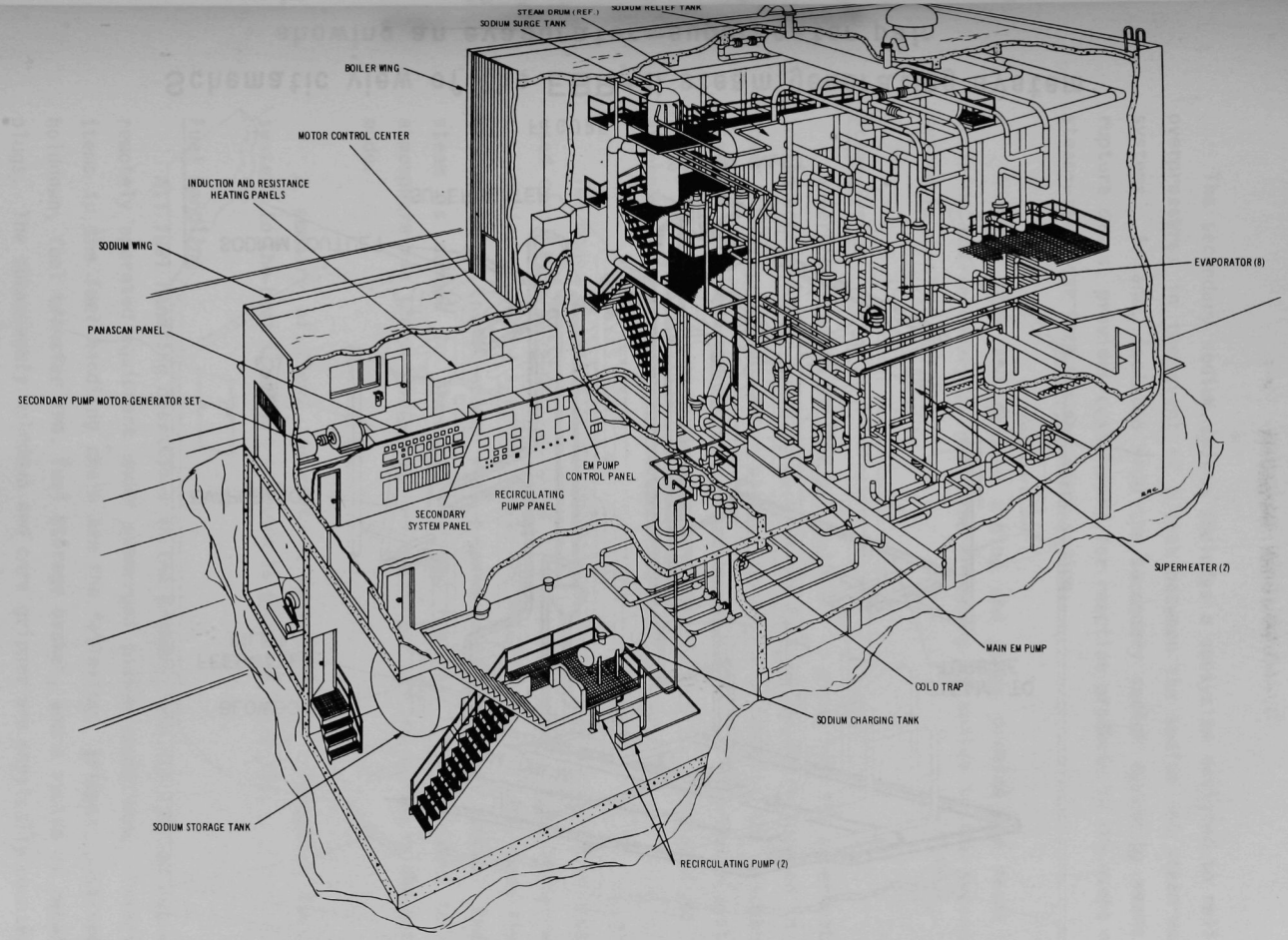


FIGURE 14

EBR–II sodium boiler building showing location of secondary sodium components

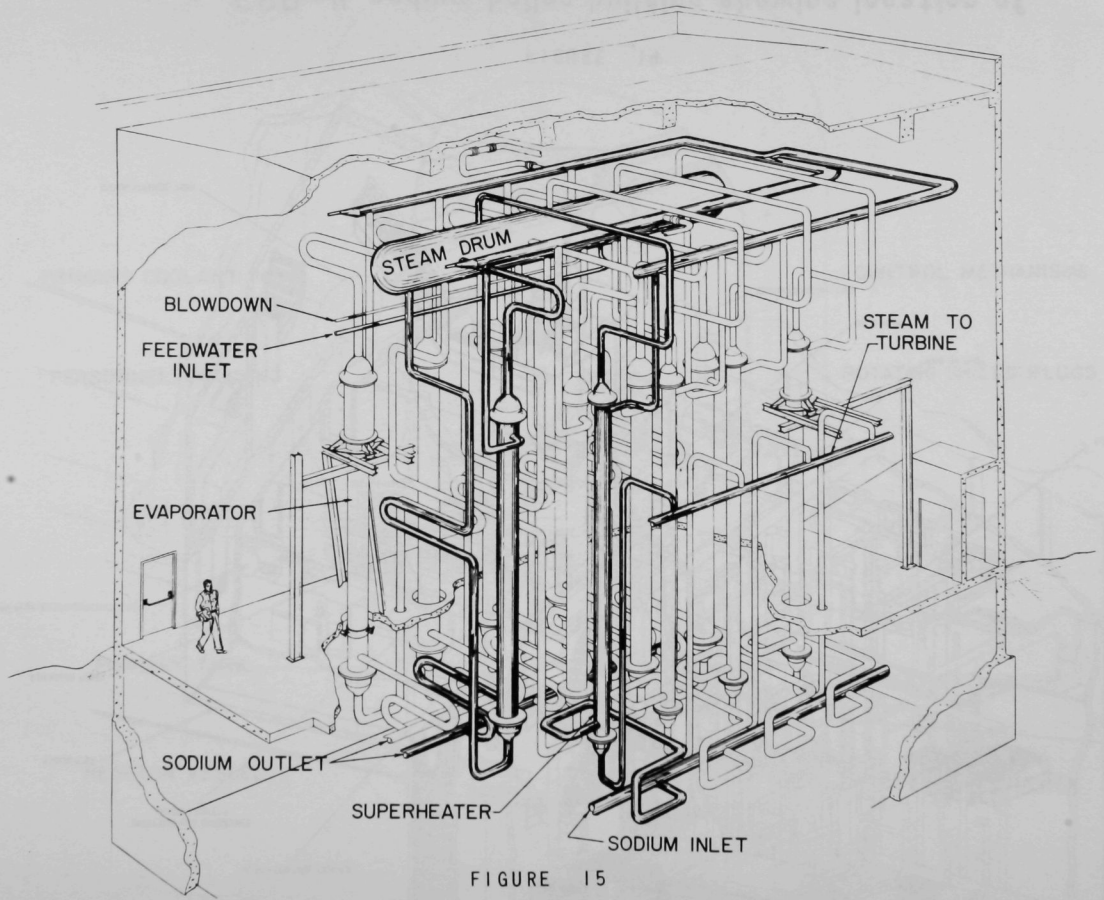


FIGURE 15

Schematic view of the EBR–II steam generating system showing an evaporator–superheater pair

The secondary sodium system includes a subsystem designed to relieve overpressures in the event of a leak between the sodium and steam-water systems. A tank, isolated from the secondary sodium system by means of rupture discs, provides containment for reaction products in the event of a steam-water leak into the secondary sodium.

A drain tank and related piping and pumps provide the means for draining the secondary system. The inventory of sodium in the secondary system is about 12,000 gal.

Steam System

Superheated steam at 1250 psi and 820°F is piped from the superheaters in the sodium boiler building to a 20-MWe turbine generator located in the power plant building. After passage through the turbine, the discharged steam is condensed and recycled through a condensate and feedwater system. Low grade heat in the condenser cooling water is dissipated to the atmosphere via a forced draft evaporative cooling tower.

Two modes of generator control are available. The mode that is usually used consists of controlling steam pressure in the main steam header with the turbine-generator synchronized to the power grid. The other mode limits the output of the generator with a speed governor control system. Excess steam is dumped to the condenser and the heat is dissipated to the atmosphere via the cooling tower and condenser cooling system in the second mode.

A summary of flow, temperature, and pressure information for all systems in the heat transport sequence is given in Table VII.

Fuel Handling

All fuel handling operations in the primary tank are carried out with remotely operated equipment under submerged sodium conditions. Principal items in the fuel handling chain are the following: gripper, subassembly holddown, fuel transfer arm, fuel storage basket, and a system of rotating plugs. The subassembly holddown and core gripper are physically mounted on the small rotating plug which, in turn, is eccentrically mounted in the

Table VII
Design and Operating Data
Heat Transfer Systems

Total Heat Output	62.5 MWt
Gross Electrical Output	19.5 MWe
Primary Sodium Temperatures	700°F
Outlet Sodium Temperature (from core)	883°F
Primary Sodium Flow Rate (through reactor)	9000 gpm
Maximum Velocity of Primary Sodium Coolant	23.8 ft/sec
Primary System Sodium Capacity	89,000 gal
Secondary Sodium Temperature, to IHX	588°F
Secondary Sodium Temperature, from IHX	875°F
Secondary Sodium Flow Rate	5890 gpm
Steam Output	250,000 lb/hr
Feedwater Temperature	550°F
Steam Flow to Turbine	195,300 lb/hr
Steam Temperature at Turbine	820°F
Steam Pressure at Turbine	1250 psig

large rotating plug. Through the precision rotation and indexing of the two plugs, the holddown mechanism and subassembly gripper can be placed over any of the 637 core and blanket subassemblies. All principal components in the fuel handling sequence are illustrated in Fig. 16.

The fuel handling system is specifically designed to permit two modes of operation. In the unrestricted fuel handling mode, the reactor is shut down and subassemblies may be inserted into or removed from the core. In the restricted mode, subassemblies may be transferred to or from the primary tank with the reactor operating.

The unrestricted fuel handling mode involves the following sequence of events. The reactor is shut down with all control rods driven to their lowermost configurations, the control rod drives are disconnected and raised, and the reactor cover is raised to its uppermost position. The two rotating plugs are then rotated to place the holddown mechanism and subassembly gripper over a specific core position. The holddown mechanism and core gripper perform the following respective functions: to secure the six neighboring subassemblies and to engage the lifting adapter on the subassembly of choice. The gripper, along with its engaged subassembly, is raised to the elevation of the transfer arm. The subassembly, now completely clear of the core, is transferred to the transfer arm. At this point the storage basket is raised, rotated to its proper position, and the subassembly is placed in a specific location of the storage basket. The insertion of a subassembly into a core position follows the reverse order.

The restricted fuel handling mode involves the transfer of a subassembly from the storage basket to the transfer arm or the reverse. With the subassembly firmly engaged, the transfer arm is rotated to a position under the fuel transfer port. Located above the fuel transfer port is a track-mounted shielded cask called the FUM (fuel unloading machine). A gripper in the FUM is lowered to engage the lifting adapter. When free of the transfer arm, the subassembly is lifted into the cask with the FUM

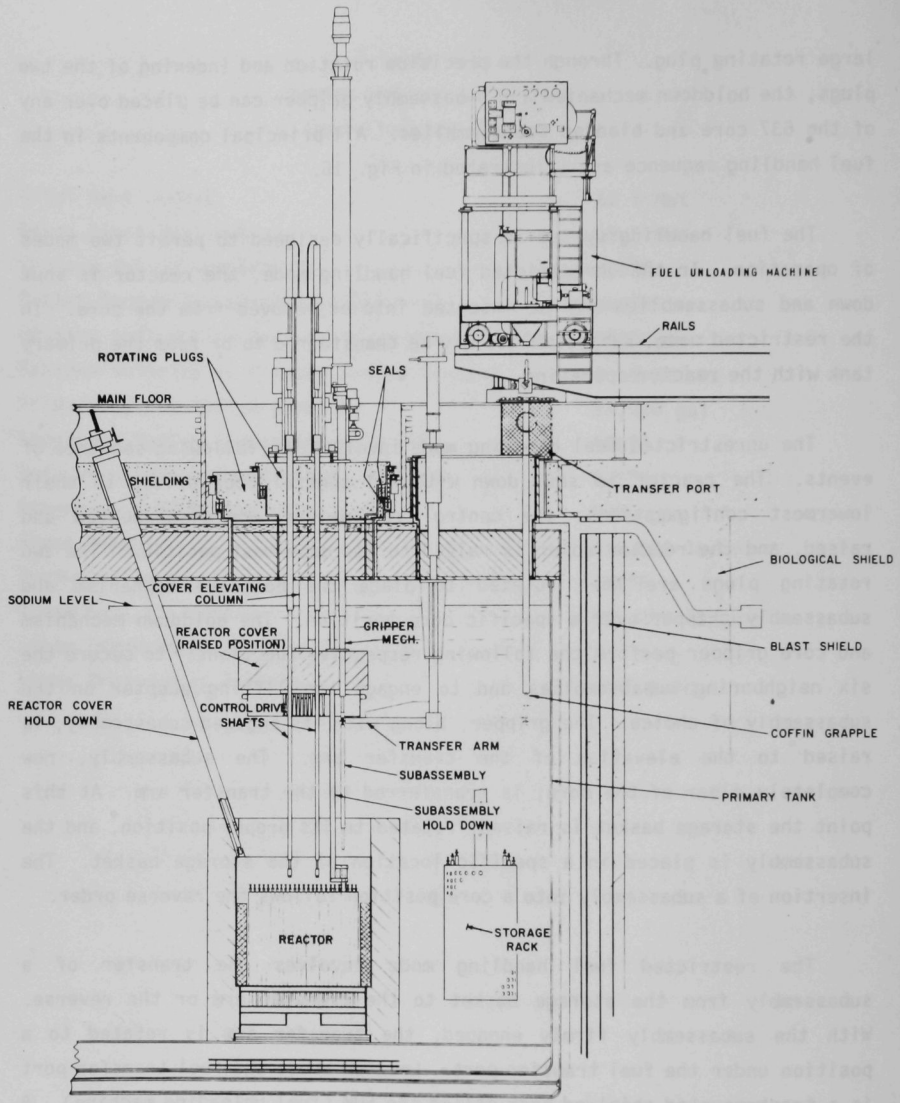


FIGURE 16

Principal components used in the EBR–II fuel–handling sequence

gripper. The FUM is then moved along a set of rails to a position which permits the subassembly to be lowered into a portable shielded cask called the IBC (interbuilding coffin). Following this transfer the FUM is backed away from the IBC and the IBC is lifted with a polar crane and lowered through an air lock onto a railed cart. The air lock is closed, and the cart is driven to a second air lock which opens into a corridor in the HFEF/S Complex. From here the IBC with the discharged subassembly may be moved to facilities in either HFEF/S or HFEF/N. In general, spent fuel assemblies are handled in the air cell of HFEF/S, whereas irradiation subassemblies are processed in HFEF/N.

Irradiation subassemblies containing irradiated materials may be reinserted in the core via the reverse route. Fresh fuel subassemblies, on the other hand, may be loaded directly into the FUM. Since initial reactor operation in 1964, over 6000 round-trip transfers have been made.

Primary Sodium Purification System

Sodium in the primary system is maintained under high-purity conditions through the periodic "trapping" of impurities in an externally located purification facility. Impurity removal or "cold-trapping" is based on a temperature-sensitive solubility effect. When impure sodium is cooled below saturation solubilities, the impurities precipitate as solids in a wire mesh-filled trap (or tank).

The primary sodium purification systems takes suction from the primary tank, routes the stream through a surge tank, through a dc electromagnetic pump, to a heat economizer, through the cold trap, back through the economizer, and ultimately back to the primary tank. The cold trap is cooled by the circulation of NaK through a jacket which surrounds the trap. The NaK system is cooled by a silicone-oil system which, in turn, is cooled by a cold water stream.

The principal impurities removed during cold-trapping operations are sodium oxides, sodium hydrides, various fission products (probably in the oxide form), and tritium. Impurities in the primary coolant are controlled

at levels that correspond to saturation temperatures in the region of 208°F (the melting point of sodium) to approximately 300°F. Samples of primary sodium are periodically taken from a pumped loop that delivers primary sodium to the externally located cold trap.

Important Subsystems

Many of the EBR-II subsystems that are operating today were designed with the technologies of the early 1950's. Despite the unavailability of important design information at that time, the systems have performed remarkably well. Some of the more important systems that were designed on a first-of-a-kind basis are discussed below.

Steam Generator

The building of reliable steam generators for sodium-cooled reactors is a matter of worldwide interest. Experience with the EBR-II steam generator has shown that reliability and efficiency are the logical consequences of sound design and construction practices. The EBR-II steam generator has operated reliably since 1964. Since that time, the only difficulty consisted of a faulty but easily repairable weld.

The steam generator consists of eight evaporators connected in parallel, a steam drum, and two superheaters also connected in parallel. The evaporators are supplied with water from the steam drum, which is located at the high point in the system. Water flows from the steam drum through the evaporators. The water-steam mixture returns to the steam drum under natural circulation.

Construction details of the superheaters and evaporators are similar; both are vertical, straight-tube, counter-flow heat exchangers. (See Fig. 15 for details.) The tubes in each superheater and evaporator consist of two bonded concentric tubes. Sodium flows on the shell side of the exchangers. Steam and water, on the other hand, flow in the tube side of the superheaters and evaporators.

The bonds between the concentric tubes consist of two types. The tubes in four evaporators and one superheater are mechanically bonded by drawing the concentric tubes through a die and over a mandrel and then expanding the inner diameter with a subsequent drawing operation. This procedure leaves the tubes in a prestressed condition with the outer tube in tension and the inner tube in compression. The tubes in the remaining four evaporators and one superheater are metallurgically bonded by the predrawing treatment of mating surfaces with a nickel, nickel phosphorus alloy.

Sodium Pumps

The two primary sodium pumps are single stage centrifugal pumps with hydrostatic sodium lower bearings. The pump motors are sealed to the pumps and the motors are filled with argon gas. Aside from early difficulties, operational experience with the pumps has been excellent. On three occasions, twice in 1963 and once in 1971, pumps were removed to correct problems caused by shaft bowing and galling.

The secondary system utilizes a single ac three-phase linear electromagnetic pump for the main system loop. The pump not only provides normal flow, but is also used to restrict natural convective flow in the system when the reactor is not operating. Reversing the pumping direction and "bucking" the convective flow allows control of sodium flow to within 1/10 of 1% in the secondary system during shutdown. Only one repair, the welding of a crack in the pump tube, has been necessary since 1964.

Pump operation experience in EBR-II has shown that properly designed pumps can operate satisfactorily in a sodium environment.

Intermediate Heat Exchanger (IHX)

The primary sodium to secondary sodium heat exchanger (IHX) is another major component at EBR-II that has demonstrated the high operational reliability of components submersed under sodium.

The IHX is a counter-flow heat exchanger that has 3026 stainless steel tubes, each of which is 5/8 in. in diameter. Secondary sodium flows through the tubes and primary sodium flows around the tubes on the shell side. Except for a broken pipe clamp on a drain tube inside the IHX, no operational difficulty has been experienced. Important details are illustrated in Fig. 17.

The Use of Metallic Fuels in EBR-II

EBR-II is the only operating FBR in the world with metallic driver fuel. All but one of the others, either planned, under construction, or operating, are fueled with a mixture of PuO_2 and UO_2.*

Several considerations affected the initial design of the EBR-II driver fuel during the early and mid-1950's. These were: the relative ease of fabrication, the success of four metallic fueled loadings in EBR-I, the excellent heat transfer properties of metals, the superior breeding characteristics of metallic fuels, and the inherent promise of a simple and rapid on-site turnaround of discharged fuel.

From the start, operation with metallic driver fuel was successful. As operating experience accumulated, burnup limits were incrementally increased from the original value of 1.0 at.% (established in 1961) to 1.2 at.% in 1966, to 1.8 at.% in 1969, and to 2.6 at.% in 1975. Beyond 3.0 at.%, fuel swelling became a problem. Pressures exerted at the fuel cladding interface were so large that cladding strain and subsequent rupture seemed likely. Such limitations were always evident. Apparent, even in the early days of design, were the inevitable penalties: short operating cycles and inefficient fuel utilization. However, the impact of the penalties was softened by the rapid on-site reprocessing of discharged fuel. On certain occasions discharged fuel was reprocessed and returned to the core within 29 days.

* BOR-60, in the USSR, is an exception; BOR-60 is fueled with enriched UO_2.

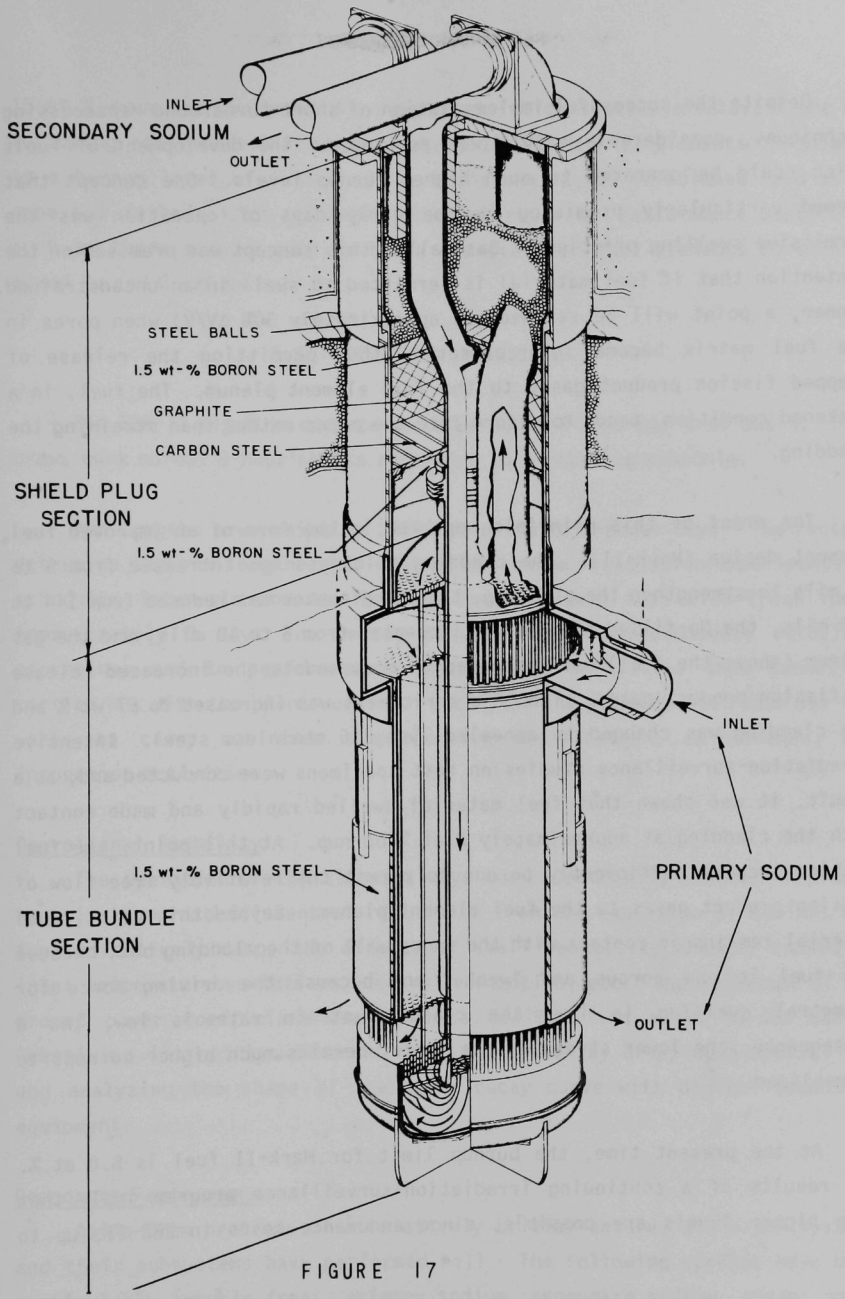


FIGURE 17

Major components of the EBR–II intermediate heat exchanger

Despite the successful implementation of short turnaround reprocessing techniques, considerable effort was devoted to the development of fuels which could be operated to much higher burnup levels. One concept that seemed particularly promising in the early days of operation was the permissive swelling principle. Basically, this concept was premised on the contention that if fuel material is permitted to swell in an unconstrained manner, a point will be reached (at approximately 30% $\Delta V/V$) when pores in the fuel matrix become interconnected, thus permitting the release of trapped fission product gases to the fuel element plenum. The fuel, in a weakened condition, tends to deform into the pores rather than straining the cladding.

The proof of this principle appeared in the form of an improved fuel element design (Mark-II). The cladding thickness was increased from 9 to 12 mils to strengthen the cladding, the pin diameter was reduced from 144 to 130 mils, the Na-filled annulus was increased from 6 to 10 mils, and the gas plenum (above the fuel) was increased to accommodate the increased release of fission product gases. The ^{235}U enrichment was increased to 67 wt % and the cladding was changed to annealed Type 316 stainless steel. Intensive irradiation-surveillance studies on test specimens were conducted and, as a result, it was shown that fuel material swelled rapidly and made contact with the cladding at approximately 2 at.% burnup. At this point, the fuel lattice becomes sufficiently porous to permit the relatively free flow of fission product gases to the fuel element plenum. Beyond this point, fuel material remains in contact with the inner wall of the cladding but, because the fuel is now porous and "weak," and because the driving force for diametral swelling is low, the cladding strain rate is low. As a consequence, the lower stress on the jacket permits much higher burnups to be realized.

At the present time, the burnup limit for Mark-II fuel is 8.0 at.%. The results of a continuing irradiation-surveillance program imply that even higher levels are possible, since endurance tests in EBR-II, up to

16 at.%, have been successfully completed. Of considerable interest are the following statistics: Of approximately 10,000 Mark-II elements irradiated to a burnup level of 6 at.% and an additional 2,000 irradiated to 8 at.%, none has failed, i.e., suffered cladding rupture. Performance statistics such as these are as good as those expected for fuel elements containing mixed Pu and U oxides.

5.0 OPERATING EXPERIENCE

The EBR-II plant is operated on a 24-hour, 7-days-per-week basis. Four crews work normal 8-hour shifts according to a rotating schedule.

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 7 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.

Operational Stability

The operation of EBR-II has always been kinetically stable. A prompt negative power coefficient of reactivity from the expansion of coolant and fuel effectively damps the effects of small reactivity changes caused by inlet temperature variations, control rod motion, etc. The amplitude of the prompt power coefficient component is monitored on a run-to-run basis. Reactivity is rapidly withdrawn from the system by dropping a special rod and analyzing the shape of the power decay curve with digital computing equipment.

Component Performance

Aside from problems of a relatively trivial nature, all major systems and their subsystems have performed well. The following systems have been particularly trouble free: primary sodium, secondary sodium, steam, power

plant, and plant instrumentation (flow, temperature, pressure, etc.). Many difficulties have been experienced with the fuel handling and rotating plug systems, but even these were correctable by relatively simple modifications.

Power Output

EBR-II is normally operated at its full power rating of 62.5 MWt. Under full power conditions, 19.5 MW of electrical energy is generated. Approximately 5.5 MW is used to satisfy the ANL-W demand; the remaining 14 MW is fed through a 13.8 kV loop to the INEL distribution grid. In addition, approximately 12,000 lb. of saturated steam per hour is utilized for local space heating. The use of plant steam results in an annual savings of approximately 400,000 gal of oil. The savings of electricity and fuel oil costs and revenues from the sale of electricity amounts to approximately $1,000,000/year.

Plant Availability

During the period 1976-1978, EBR-II operated with plant availability factors greater than 70%. A peak value of 76.9% was reached during 1976. Plant availability factors in this range compare favorably with those for commercial nuclear and conventional fossil-fueled power plants. If down-time required by the experimental program is discounted, the actual plant availability factor during the 1976-1978 period exceeded 80%.

Refueling

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 fuel storage feature is beneficial in another important respect. After fulfilling minimum cooling requirements,

spent subassemblies may be transferred into and out of the storage basket while the reactor is running.

Fission Product Releases

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 results of endurance tests in which fuel elements are intentionally irradiated to failure and beyond. On other occasions, the failure may be the result of birth defects in the element (e.g. a faulty weld) or by premature failure of cladding. More recently, the effects of sustained operation under breached cladding conditions have been investigated. Fuel elements which have failed under irradiation are permitted to remain in the core for periods up to a few weeks in order to evaluate the effects of sustained operation on the fuel element and the consequences of releasing fission products to the primary coolant and cover-gas systems.

A portion of the program is concerned with the effects of fission products on the primary sodium and cover-gas systems. Species released to the coolant are removed, in part, by means of an externally located cold-trap. Gaseous fission products released to the cover-gas system are removed by a combination of cryogenic trapping and absorption by liquid-nitrogen-cooled charcoal beds.

Fission-Product Detection Techniques

Several on-line techniques are used to detect, annunciate, and measure releases of fission products from failed fuel elements to the primary sodium and cover-gas systems. One, the FERD (Fuel Element Rupture Detector), is used to monitor a small by-pass stream of primary coolant for the presence of delayed neutron emitters in the exit coolant. Another device, the GLASS (Germanium Lithium Argon Scanning System) analyzes a flowing cover-gas stream by means of gamma pulse height spectrometry with a Ge-Li crystal detector. An older system, the FGM (Fission Gas Monitor) analyzes a flowing cover-gas sample for the presence of rare-gas fission products by

When components in this category are being removed, the system must be "cooled" for approximately 5 days to permit the decay of ^{24}Na. The component is then pulled into a shielded pipe that is handled by the building crane. As for all activities that involve access to the primary tank, precautionary measures must be taken to prevent the inleakage of air to the cover gas system and the leakage of cover gas to the reactor building.

As the result of over 15 years of successful experience with removing, cleaning, maintaining, repairing, and disposing of primary tank components, it is reasonable to conclude that with ingenuity and forethought, FBR's can be designed to permit maintenance-repair activities on in-tank components.

7.0 IRRADIATION PROGRAM

Since 1965, EBR-II has served as the nation's only facility for the irradiation-testing of fuels, structural materials, absorbers, and sensors under conditions similar to those expected for more advanced FBR's.

The irradiation program in EBR-II is predicated on the need to learn more about the chemical and physical behaviors of various fuel and structural materials under high-flux, high-temperature, flowing sodium conditions. Without definitive behavioral information, the designers of more advanced systems would be limited in their ability to arrive at the best possible design from the viewpoints of plant safety and economics. Many factors and variables must be considered. Among these are the following: fuel type, method of fabrication, fuel column thickness and length, annular gas gap, fuel-swelling, fuel cracking, central void formation, cladding type, cladding thickness, cladding swelling, cladding strain, cladding strength and ductility, etc. Also important are matters such as power density, heat transfer, pumping power, breeding gain, doubling time, fuel utilization, etc. Usually if the designer changes one variable he affects others, some beneficially and some adversely. It is his task, then, to understand how such variables interact and ultimately arrive at a design that is based on the best possible tradeoff. Although sophisticated

Maintenance activities conducted on EBR-II and its peripheral plant systems may be assigned to three general categories that reflect accessibility and hazards concerned with radioactivity, viz, work on conventional, readily accessible and nonradioactive components; work on unconventional, relatively accessible but moderately radioactive components; and work on conventional, relatively inaccessible, and highly radioactive components.

Activities that fall under the first category are classified as preventive maintenance. Typical systems subject to preventive maintenance are those located outside the primary system. Preventive maintenance consists essentially of periodic inspection, repair, modification, and replacement, if necessary. Depending on the component and its function, preventive maintenance activities may or may not be conducted with the plant in operation. If the plant must be shut down, work activities are scheduled to coincide with the next scheduled shutdown.

Considerable maintenance experience with unconventional, relatively accessible, and moderately radioactive components has been achieved. A typical component falling in this category is the transfer arm (part of the fuel handling system). Precautionary measures associated with such activities are those needed to maintain an argon atmosphere around the component, to prevent the inleakage of air to the cover-gas system, and to prevent the leakage of cover gas to the reactor building. All such activities are based on pulling the component into a rubberized-nylon bag which is sealed by a flange to a penetration in the primary containment tank.

The most difficult components to maintain are those that are unconventional, relatively inaccessible, intrinsically radioactive, and radioactively contaminated. Examples of such components are control-rod drives, the main core gripper, and the subassembly holddown fixture. Such components penetrate the reactor vessel cover and, as a result, became highly radioactive through neutron activation. The problems of intrinsic radioactivity, sodium contamination, and fission-product plateout make direct repair virtually impossible. In situations such as these, e.g., a malfunctioning control rod drive, the entire unit is removed and replaced.

When components in this category are being removed, the system must be "cooled" for approximately 5 days to permit the decay of ^{24}Na. The component is then pulled into a shielded pipe that is handled by the building crane. As for all activities that involve access to the primary tank, precautionary measures must be taken to prevent the inleakage of air to the cover gas system and the leakage of cover gas to the reactor building.

As the result of over 15 years of successful experience with removing, cleaning, maintaining, repairing, and disposing of primary tank components, it is reasonable to conclude that with ingenuity and forethought, FBR's can be designed to permit maintenance-repair activities on in-tank components.

7.0 IRRADIATION PROGRAM

Since 1965, EBR-II has served as the nation's only facility for the irradiation-testing of fuels, structural materials, absorbers, and sensors under conditions similar to those expected for more advanced FBR's.

The irradiation program in EBR-II is predicated on the need to learn more about the chemical and physical behaviors of various fuel and structural materials under high-flux, high-temperature, flowing sodium conditions. Without definitive behavioral information, the designers of more advanced systems would be limited in their ability to arrive at the best possible design from the viewpoints of plant safety and economics. Many factors and variables must be considered. Among these are the following: fuel type, method of fabrication, fuel column thickness and length, annular gas gap, fuel-swelling, fuel cracking, central void formation, cladding type, cladding thickness, cladding swelling, cladding strain, cladding strength and ductility, etc. Also important are matters such as power density, heat transfer, pumping power, breeding gain, doubling time, fuel utilization, etc. Usually if the designer changes one variable he affects others, some beneficially and some adversely. It is his task, then, to understand how such variables interact and ultimately arrive at a design that is based on the best possible tradeoff. Although sophisticated

computational techniques are available to assist the designer; the success of his efforts must still rely on experimentally-based information.

EBR-II As An Irradiation Facility

Although EBR-II's use as an irradiation facility was not initially considered, the accelerated interest in fuel breeding during the 1960's prompted a redefinition of goals. Its conversion to a test facility was gradual and relatively trouble free; major modifications were not needed.

The vehicle originally used for irradiating specimens was and still is, in its simplest form, a standard subassembly that has been internally modified to accommodate one or more irradiation specimens. As the irradiation program developed, a wide variety of vehicles was designed. Today more than 50 different vehicles are available. These can accommodate from one to 91 individual fuel or structural specimens. Although internal designs vary widely, all irradiation vehicles are superficially identical. Since all are uninstrumented (therefore no leads), the existing fuel handling system is used for insertion and removal activities.

The first use of EBR-II as an irradiation facility 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 increased to a record of 65 during June, 1977. A diagram that illustrates this particular core loading is given in Fig. 11.

Of a total of 127 core positions available, 64 were occupied by irradiation vehicles, 8 by control rods, 2 by safety rods, 3 by special instrumented facilities, one by a drop-rod and 57 by subassemblies filled with driver fuel. Of the 64 irradiation vehicles, 17 contained mixed U-Pu oxide fuels, 19 contained structural specimens, 13 contained mixed U-Pu nitride and carbide fuels, 5 contained absorber materials, and 10 contained driver-fuel elements irradiated under run-to-failure conditions. A breakdown of the experimental complement into individual specimens reveals the following: metallic fuel elements, 910; mixed U-Pu oxide fuels, 647;

mixed U-Pu nitride and carbide fuels, 247; structural specimens, 151; and absorber materials, 20.

On a cumulative basis, as of December 31, 1978, the complement of elements, capsules, and other specimens may be broken down as follows: mixed U-Pu oxide fuels, 2918; driver-fuel elements (surveillance and run-to-failure program, 4528; other LMFBR fuels (principally mixed U-Pu nitrides and carbides), 623; cladding and structural specimens, 1188; absorber materials, 212; and miscellaneous, 313. As a result of the irradiation program, many milestones have been reached. Mixed U-Pu oxide fuels have been irradiated to a heavy-atom burnup of 20.5%, cladding temperatures of 1500°F have been achieved, and neutron fluences of 1.7×10^{23} nvt have been accumulated in structural materials. Absorber materials have been irradiated to 7×10^{21} neutron captures per cc and EBR-II driver fuel has been irradiated to a heavy atom burnup of 16.4%.

The irradiation program currently in progress addresses the irradiation behavior of the following classes of specimens:

- Structural Materials. Various alloys that indicate promise for use as fuel element cladding, subassembly wrapper tubes, wrapper wires, core structural materials, etc.
- Reference Fuels. Principally mixed U-Pu oxides.
- Advanced Fuels. Advanced mixed U-Pu oxide fuels, (thin cladding, higher densities), mixed U-Pu nitrides and carbides.
- Absorber Materials. Principally B_4C, tantalum and europia (mixed rare-earth oxides).

Instrumented In-Core Facilities

In the late 1960's and early 1970's flexibility was added to the irradiation program in the form of instrumented in-core irradiation facilities. Such facilities permit the continuous monitoring of physical data for specimens being irradiated under typical operating conditions.

Parameters measurable in these facilities include the following: temperature, pressure, coolant-flow, creep-rate, neutron flux, etc.

Two types of in-core instrumented facilities are now available. One, the INSAT (Instrumented Subassembly Test Facility), is used primarily for the irradiation of fuel materials. The other, INCOT (In-Core Test Facility), is used for the irradiation of structural materials, absorbers, sensors, etc. Both differ substantially in design and operation.

INSAT Facilities

The needs for gathering physical data from subassemblies under irradiation were not considered in the original design of EBR-II. As the irradiation program accelerated during the late 1960's, increased interest was expressed in an in-core irradiation facility which would permit cabling connections between sensors in the fuel bundle and out-of-core recording instruments. The design of the facility was constrained by two principal considerations: the need to run instrument leads through a control rod penetration in the reactor vessel cover; and the need to keep the discharged fuel bundle under the primary sodium at all times during its removal from the core. These constraints were eventually satisfied by replacing a control rod (and its associated drive) by a single unit that can be loaded from the reactor floor into the reactor grid and which permits cable communication between the in-core sensors and out-of-core recorders. The unit is, quite simply, a fuel bundle physically similar to a control rod and a double-walled extension tube which is mechanically latched to the bundle. Cables are led from the sensors, upward through the double-walled tube, and into a terminal box located outside the primary tank. During routine operation the fuel bundle occupies the same position as that normally occupied by a control rod. During fuel handling operations the entire in-core assembly is raised to an elevation that permits the normal functioning of fuel handling equipment. Following fuel handling operations, the assembly is lowered back into the core for subsequent irradiation.

When test bundles reach their target burnup and are ready for removal from the core, a special cutting tool is lowered through the extension tube. Rotation of the tool cuts the instrument leads. The extension tube and cutting tool are then drawn upward into a pulling pipe, thus leaving the fuel bundle ready for transfer with the standard fuel handling equipment.

Depending on the needs of the experimenter, as few as one and as many as 61 fuel or other specimens may be accommodated in the test-vehicle. As of June 1, 1979, eight individual sets of instrumented fueled specimens have been irradiated in the INSAT facilities. Typical information derived from the tests include temperature, coolant flowrate, gas pressure, creep rate, and neutron flux. Such information is useful to designers and analysts in two principal ways: as empirical input for design studies, and as "bench mark" data for establishing and validating the accuracy of computer programs.

INCOT FACILITIES

INCOT and INSAT facilities differ mainly in the manner in which the test section is removed from the core. Whereas INSAT facilities rely on the normal fuel handling system for discharge operations, test specimens in INCOT facilities may be inserted or removed from the reactor floor area. The distinction between the two facilities is one of use. Because of decay-heat considerations, fueled specimens must be kept submersed in the primary sodium following irradiation. Such a constraint is more easily satisfied by treating the test section during its discharge as a spent fuel subassembly. Furthermore, attempts to discharge irradiated fuel specimens via the reactor floor would pose major shielding problems.

For nonfueled test specimens, decay-heat problems are essentially nonexistent. Associated radiation fields are such that shielding requirements during discharge through the reactor top can be satisfied.

INCOT facilities are, in their simplest form, a thimble assembly that extends upward from the reactor grid (below the core), through the core, reactor vessel cover, and small rotating plug to reactor floor level.

Instrument leads run directly from test specimens or sensors up the thimble to a top-mounted terminal box. Special handling systems permit the removal of irradiated test bundles from the facility, the reinsertion of reworked irradiated test bundles into the facility, or the removal of the facility itself. Removal operations involve the withdrawal of the test bundle into a 37-ft-long handling container which is suspended from the building crane. Operation of the container is remotely controlled from a shielded booth. As for INSAT facilities, the entire assembly is lifted prior to fuel handling operations.

Since 1972, INCOT facilities have been used to irradiate six principal series of instrumented experiments. Among these were the irradiation of absorber materials (B_4, C and Eu-O_2), self-powered neutron detectors, eddy current flow sensors, acoustical monitors, and biaxial creep specimens.

NITF (Nuclear Instrument Test Facility)

Facilities are also available for the in situ performance testing of nuclear instruments under LMFBR conditions. Two of eight existing instrument thimbles are used for this purpose. In the original configuration, EBR-II was fitted with four "J" type nuclear instrument thimbles which extend downward from the upper shielding to the primary tank, and into the neutron shield around the reactor vessel. Four other "O" type thimbles similarly extend downward from the upper shielding but terminate outside the neutron shield. One "O" thimble and one "J" thimble have been converted to instrumented test facilities.

The NITF thimbles are 28 ft long and 15 in. in diameter at their lower ends. Instruments are usually located at core mid-plane elevation. Leads from the instruments are led upward through the thimble to the reactor operating floor.

Test temperatures ranging from 150° to 700°F are made possible by varying the amount of cooling air circulated through the facility. If higher temperatures are needed, thermostatically controlled ovens may be used for instrument heating. Data taken under operating conditions enable

the experimenter to evaluate the effects of high temperature and intense radiation fields on the performance of various neutron detectors and instrument cables.

RSCL (Radioactive Sodium Chemistry Loop)

The RSCL is a facility which permits the development and testing of techniques for measuring impurity levels in primary sodium. The facility consists essentially of five shielded cells, a 2-in.-diameter main loop, and smaller branch lines which deliver primary sodium to and from the various cells. Coolant is pumped through the main loop by a dc electromagnetic pump at flow rates and pressures variable up to 30 gpm and 16 psig, respectively. Each cell may be isolated from the main loop and, after a suitable decay period, physical access is permitted for the installation and maintenance of equipment. Extensive precautions have been taken to prevent, annunciate, and minimize the impact of sodium leakage throughout the entire facility.

The facilities have been extensively used as a "test bed" for proving the application of various on-line impurity measuring devices. Prominent among those devices which have either been tested, are under test, or are about to be tested in the RSCL are the following: plugging temperature indicator, tritium meter, vacuum distillation samples, oxygen and hydrogen meters, segregated iodine sampler, graphite-cesium trap, and an equilibrium module for carbon analyses. The development and use of such devices are important in two respects: as a means for monitoring sodium quality in EBR-II, and as a base technology for more advanced LMFBR applications.

Disposition of Irradiation Subassemblies

Spent fuel subassemblies enter the HFEF/S air cell via the Inter-building Coffin (IBC). Here they are processed for shipment to the reprocessing plant. Discharged irradiation vehicles, on the other hand, are transferred via the IBC to HFEF/N for postirradiation disassembly and inspection activities. One of the first operations consists of disassembling the irradiation vehicle in the VAD (Vertical Assembler-dismantler). Some specimens may be returned without examination to their

sponsor, some may be examined and then returned to their sponsor, whereas others may be examined and returned to the reactor for additional irradiation. Interim examinations permit the experimenter to monitor and examine the effects of irradiation exposure on fuels and structural materials on a periodic basis.

A wide variety of equipment and techniques for postirradiation inspection activities is available at HFEF/N.

Examples of activities that can be carried out include the following:

- visual and photographic inspection of specimens and subassemblies
- removal of surface sodium
- dimensional examination of subassemblies and irradiation vehicles
- periscopic inspection (magnification)
- gamma scanning
- profilometry (diametral-change measurements)
- precision weighing
- eddy-current cladding testing
- spacer-wire removal and rewrapping
- sodium bond-testing
- neutron radiography
- machining
- puncturing and gas collecting
- sectioning, mounting, and conducting metallographic examination
- assembling, welding, and leak-testing capsules
- reassembling irradiated specimens into vehicles for additional irradiation

All in-cell equipment is designed for remote operation. Essentially all component parts are modular and removable via the manipulation of out-of-cell controls. In-cell equipment is designed to permit the handling and examination of specimens in the vertical attitude. This feature is necessary to preserve the integrity of sodium bonds and to effect a more efficient utilization of floor space. All equipment used for the puncturing

or removal of cladding is designed to retain debris that could contaminate the cell atmosphere.

8.0 FUTURE PROGRAM

The EBR-II will continue to be used as an irradiation facility for FBR fuels, materials and sensors, at least until the FFTF at Richland, Washington becomes fully operational. Although the complement of uninstrumented irradiation experiments will decrease during the interim period, approximately 20 irradiation experiments are programmed into 1983. A similar winddown is expected in the complement of instrumented in-core fuels and materials tests. At some time during the early 1980's, the national FBR irradiation-testing effort will be assumed by the FFTF. In the interim period and after the FFTF becomes fully operational, EBR-II will become increasingly available for a variety of missions that have been held in abeyance because of higher priority commitments to the national FBR testing program. In a true sense, the transfer of responsibility for irradiation testing to the FFTF will release EBR-II for a new but equally important role in national FBR programs.

Technological areas in which EBR-II will contribute during the interim period and after FTR becomes fully operational include the following:

- Continued demonstration of the ability to operate and maintain a FBR over a period of many years.
- Endurance testing of FBR components, as for example, heat exchangers, evaporators, superheaters, centrifugal and electromagnetic sodium pumps, cold traps, valves, flowmeters, control rod drives, fuel handling equipment, etc.
- The retention of a highly skilled cadre of personnel who have had extensive experience in nearly every area of FBR engineering, operation and maintenance.

- The testing of FBR peripherals such as fission counters, ion chambers, oxygen-hydrogen analyzers, fission-product monitors, tritium meters, acoustical sensors, flowmeters, water-to-sodium leak detectors, etc.

Current programs that could be continued and expanded include the following:

- Run-beyond-cladding Breach (RBCB). Substantial efforts are being made to define the consequences of operating with one or more defective fuel elements. Phenomena of particular interest include the loss and disposition of fission products from defective fuel, the effects of coolant interaction with various fuel materials, the development of techniques for assessing the upper limits of defective fuel operation, and the practical constraints of operating and maintaining plant systems under fission-product release conditions.
- Thermal-hydraulic tests. EBR-II has been used and can continue to be used as a facility for verifying the reliability of current and future thermal-hydraulic modeling codes. Such information will be needed in matters pertaining to the safety and licensing of future FBR's.
- Breached-fuel Test Facility. A facility for investigating the plateout of fission products on "hot-leg" components is currently under design. In addition to studying plateout effects, the facility will be used to study the transport of delayed-neutron emitters in the exit coolant stream and to monitor the extent of fuel washout from defective fuel elements.

Other programs which can be carried out include the following:

- Operational Safety Testing. A wide variety of mild undercooling and overpower tests can be safely conducted at EBR-II. Such information will complement that derived from more extreme tests being conducted in TREAT and the Sodium Loop Safety Facility in the ETR.

Closing the Fuel Cycle

EBR-II and its companion facility, the argon cell of HFEF/S, could be used for prooftesting concepts in which spent reactor fuel is reprocessed on site in a contiguous facility. The concept is appealing from two important viewpoints: those concerned with the diversion of fissionable material from fuel fabrication streams, and the economic incentives of rapid fuel turnaround and lower out-of-core inventories.

Fuel processing schemes in which the fuel material never leaves the site and which keeps the fuel material in a highly radioactive condition effectively prevent the illicit diversion of fuel material for clandestine purposes. Although these features were impressively demonstrated at EBR-II during the period 1964-1969, additional demonstrations with more prototypal fuels are needed to evaluate the usefulness of the concept for more advanced fast breeder systems. The original reprocessing facility, the argon cell of HFEF/S, could very likely be refurbished to accept subassembly size batches of irradiated fuel for the prooftesting of on-site reprocessing schemes.

The on-site reprocessing of spent fuel is attractive from the view point of economics. By avoiding long cooling periods, extensive preshipment packaging, and long over-the-road hauling, the fuel inventory charged to a given plant can be substantially reduced. The reduction in total plant fuel inventory should be reflected by lower capital and operating costs.

9.0 ON-SITE FUEL REPROCESSING

Concern over the diversion of fissionable material from fuel recycle operations has focused attention on a recycle process developed and implemented at EBR-II during the early 1960's and used for approximately five years as the sole source of EBR-II driver fuel. Throughout the entire sequence, from spent fuel discharge to core recharge, fuel material remained in a highly radioactive condition. Physical access to fuel material at all

stages of reprocessing was completely denied. Since this feature appears likely to satisfy stringent antidiversionary criteria, renewed interest in the closed on-site recycling concept may be anticipated.

Background Information

During the early design stages of EBR-II, circa 1953, considerable effort was devoted to the selection of driver fuel. Many factors soon narrowed the selection of fuel to metallic alloys since these satisfied the following criteria: high fissile density, excellent heat transfer properties, ease of fabrication, low neutron moderation, high breeding ratio, and high specific power density. Oxide fuels were considered and rejected because of their lower breeding ratios and the absence of an established technology for fuel fabrication. An additional consideration that influenced the early selection of a metallic fuel loading was the limited but satisfactory experience achieved with metallic fuels in EBR-I.

Prevalent at that time was the belief that metallic fuels would suffer from a serious disadvantage, viz, their tendency to swell and strain the cladding at relatively low levels of fuel burnup. Obvious, even then, was the adverse impact of low fuel burnup on run length and fuel utilization efficiency. Equally obvious was the need to develop a technology for the rapid "turnaround" of spent fuel in the reprocessing cycle. Since fuel material in the storage-reprocessing-refabrication cycle must be charged to total fuel inventory, a rapid turnaround of spent fuel was considered an economic necessity.

Clearly needed was a process that satisfied the following criteria:

- Short "turn-around" for recycled fuel
- Low fuel inventory
- Low capital plant costs
- Capability for handling "short-cooled" fuels
- On-site reprocessing
- Minimum solid wastes

The process that satisfied these criteria and which was implemented from the beginning was based on a pyrometallurgical technique in which spent fuel was melted, separated from bulk fission products, and recast into fuel pins for return to the reactor.

The FCF

The first loading of EBR-II fuel was manufactured from "cold" material, using prototypal fuel fabrication equipment under "cold" laboratory conditions. All subsequent reprocessing operations were carried out in a complex called the FCF (Fuel Cycle Facility). In its simplest form the FCF consisted of two heavily shielded and remotely operated facilities: one, the air cell, in which spent fuel subassemblies were disassembled and fresh fuel elements reassembled into subassembly form; and the other, the argon cell, in which all fuel element reprocessing and fabrication operations were performed. (See Fig. 18.) All operations in both the air cell and the argon cell were performed with the aid of bridge cranes, electromagnetic bridge manipulators, and master-slave manipulators. Specialized equipment contained in the cell was actuated from the various operating corridors via electrical, electromechanical, and pneumatic means. All in-cell equipment was designed for either remote repair or replacement on a modular basis.

The first reprocessing of irradiated fuel began in September 1964 and continued until April 1969. During this period approximately 6000 kg of irradiated fuel material was processed. This amount was equivalent to 353 subassemblies or approximately five full core loadings. Turnaround times for fuel during this period averaged approximately 45 days after arrival at the FCF. On certain occasions, however, irradiated fuel material was returned to the reactor within 29 days.

Until the FCF shutdown in April 1969, over 40,000 fuel elements (including the initial cold charge) were used to fuel the reactor. Of these, only one failed in service. The average availability of remote handling and process equipment during the active lifetime of the FCF was approximately 90%.

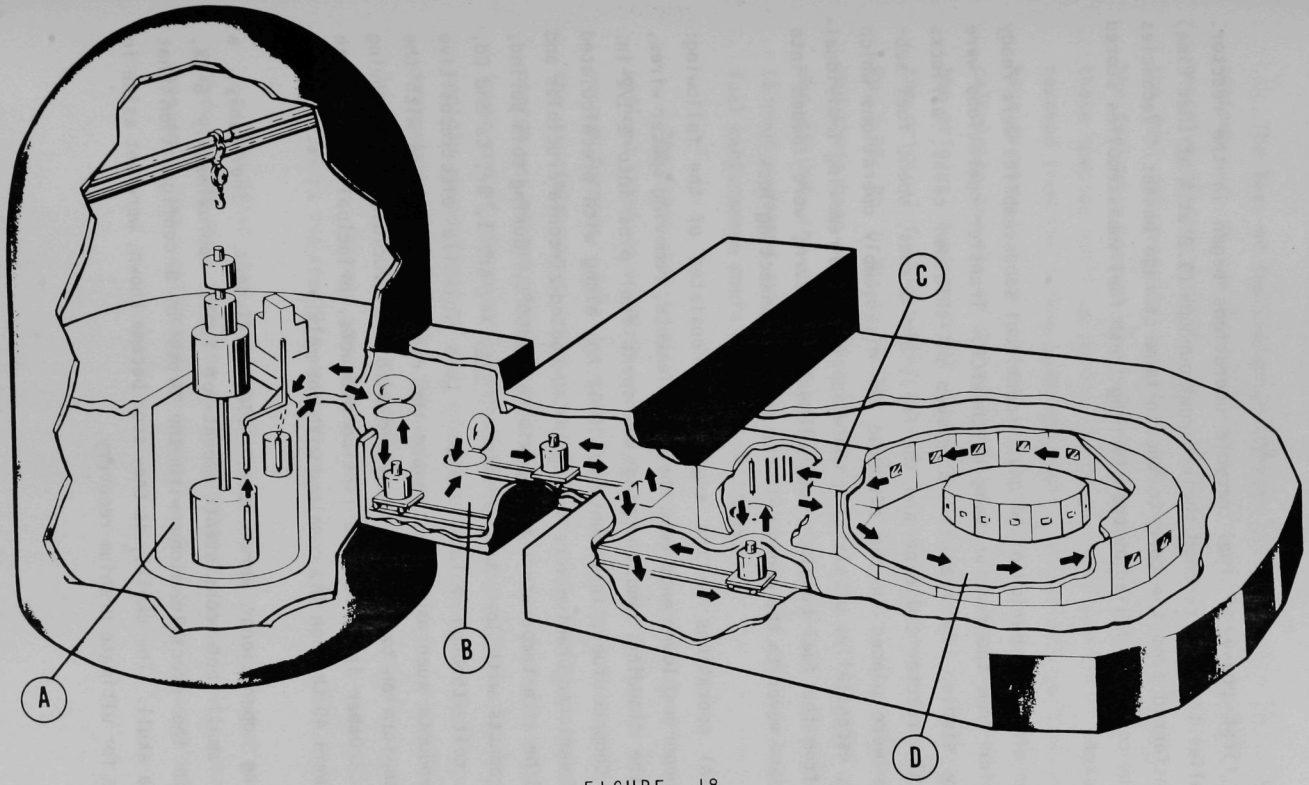


FIGURE 18

EBR–II reactor and Fuel Cycle Facility (FCF) showing (A) reactor vessel, (B) transfer of fuel, (C) air cell, and (D) argon cell

Fuel Flow

The first phase of fuel recycle operations began in the reactor. Subassemblies that had reached their target burnups (1.2 at.% at that time) were transferred from the core to the in-tank storage basket. Vacancies left in the core were filled by transferring fresh fuel subassemblies stored in the basket.

After a cooling period of 14 days, spent fuel subassemblies were ready for transfer from the reactor tank to the FCF. Transfer operations were relatively simple. By means of a system of shielded casks, airlocks (between the reactor building and the FCF), and cranes, spent fuel subassemblies were placed in the air cell for disassembly operations which consisted, essentially, of slitting the wrapper can and removing individual elements from the fuel bundle. Fuel elements, in turn, were loaded into magazines and moved to the argon cell via an interconnecting lock.

Initial operations in the argon cell consisted of the following: shearing upper and lower ends from the fuel elements, removing spacer wires, removing the cladding, and chopping the spent fuel pins into 1-1/2-in. lengths. Chopped fuel, in amounts of 10-12 kg, along with precalculated amounts of enriched uranium were melted in CaO-coated zirconia crucibles and held for 3 hr at a temperature of appoximately 1400°C. During this period, fission products with low vapor pressures, viz, Xe, Kr, I, Br, Cs and Cd, left the melt through volatilization. Less volatile electropositive fission products such as Y, Ba, Sr and the rare earths reacted with the zirconia to form an oxide which remained with the crucible after pouring operations. Other less reactive fission products, principally those with atomic numbers 40 through 46, remained with the melt.

Pouring operations were never 100% efficient. Inevitably, a relatively small but consistent amount of fuel, approximately 6-8%, remained in the crucible along with the dross in a configuration that resembled a skull. The skulls, as they soon became known, were set aside in hot storage for ultimate uranium recovery.

The loss of fuel material through skull formation was not necessarily serious and, in one important respect, was actually beneficial. The less reactive fission products, viz., Zr, Ru, Rh, Pd, and Nb, which remained with the fuel enhanced the metallurgical and radiation-resistant properties of the fuel when present to a collective extent of approximately 5 wt %. Since these species built up during the irradiation portion of the cycle and were removed from the stream with the skulls, an equilibrium concentration would have eventually been reached. To achieve an earlier equilibrium, carefully calculated amounts of the noble metals (atomic numbers 40-46) were added to the original fuel material. This material, known as fissium, was present in the original fuel material in the following amounts: Zr, 0.1%; Mo, 2.4%; Ru, 1.9%; Rh, 0.3%; Pd, 0.2%; and Nb, 0.01% (all weight-percentage values). Had there been no loss through skull formation, or "dragout" as the process was called, the concentration of the noble metals would have increased through repeated cycling to a point where their effects on reactivity and fuel performance would have been noticeable.

The first step in pin-casting operations consisted of remelting a melt-refined ingot in a thoria-coated graphite crucible with an induction-heated vacuum furnace. Also located in the furnace in a vertical attitude above the crucible was a cluster of approximately 100 Vycor molds. After the charge was melted, the crucible was raised to a position that immersed the lower end of the mold cluster to a depth of 1-1/2-in. The furnace was then rapidly pressurized to 1.7 atmospheres to drive the melt upward into the evacuated molds. After a few seconds, the melt froze and the crucible was lowered to its original position. Following a 4-hr programmed cooling period, the furnace was opened and the molds were removed. Unused fuel material in the crucible was broken up and returned to the fuel stream.

The first step in pin-processing operations consisted of breaking the Vycor molds away from the castings with a pneumatically actuated crushing system. Pins, collected in trays, were fed by gravity to a shear that

cropped the pins to required lengths. Following cropping operations, the pins were subjected to nondestructive tests which consisted principally of measurements for weight, length, diameter, and porosity. Rejects were fed back to the melt refining operation.

Acceptable pins were loaded into stainless steel tubes containing solid sodium wire (approximately 0.65 to 0.85 g). The sodium was melted and the fuel pins settled by gravity. End-plugs were inserted into the jackets and peripherally welded to the jackets with a remotely operated capacitor-discharge welder. To ensure void-free sodium bonds, the finished elements, clustered 50 to a magazine, were heated in a furnace for one hour at 500°C. The elements were then subjected to approximately 1000 vertical impacts (under 500°C conditions) at the rate of 100 impacts per minute. The finished elements were then inspected for bond-flaws and sodium level. The final step consisted of mounting the elements on a grid-system and incorporating the resultant fuel bundle in a fresh hexagonal wrapper can for ultimate return to the reactor.

After nearly five years of highly successful operation, the FCF was shut down (1969). Subsequent core charges were made either by vendors in the commercial sector or by on-site personnel in a companion "cold" line facility. The current status of the argon cell is one of a standby nature. The cell has been stripped of equipment and is in the process of decontamination. If interest in on-site reprocessing techniques revives, the cell could very likely be refurbished to serve as a proof-type facility for testing attractive reprocessing schemes.

10.0 MAJOR SUPPORT FACILITIES

Essentially all facilities at the ANL-W sites are dedicated to research and development activities under the national fast breeder reactor program. Some, in particular the HFEF complex, directly support the basic irradiation mission.

Hot Fuel Examination Facility (HFEF)

The HFEF complex provides the capability for assembling, disassembling, and examining specimens that have been irradiated in EBR-II, TREAT, and the SLSF loop in the ETR. The complex consists of two principal facilities: HFEF/S (formerly the FCF), and HFEF/N, a larger, more recent, and more flexible version of HFEF/S. Schematic views of HFEF/S and HFEF/N are presented in Figs. 19 and 20, respectively. The HFEF/S facility consists of a single heavily shielded air-filled cell. Its original companion, the toroidal argon-filled cell used for reprocessing EBR-II driver fuel during the period 1965-1969, has been stripped of equipment and is in the process of decontamination. The HFEF/N facility consists of two heavily shielded cells, one air filled and the other argon filled. All operations on components and specimens that are adversely affected by oxygen, e.g., exposed fuel and elemental sodium, are carried out in the argon-filled main cell.

All operations in the three operational cells are carried out remotely by means of electrically, electro-mechanically, and pneumatically controlled manipulators and cranes. All cells are equipped with viewing windows, approximately 4 ft thick, made of high density leaded glass. All in-cell equipment is specifically designed for remote maintenance and repair and for removal and replacement, if necessary.

In general, operations on smaller components, e.g., fuel and irradiation subassemblies, are conducted in the HFEF/S facilities. Here subassemblies may be disassembled, cleaned, examined, and either packaged for shipment or reassembled for additional irradiation in EBR-II. Discharged driver fuel subassemblies are disassembled and the spent fuel elements are packed in canisters for shipment to the Idaho Chemical Processing Plant for fuel recovery. Capabilities exist for remote milling operations. In this way specimens of various materials may be prepared for additional study.

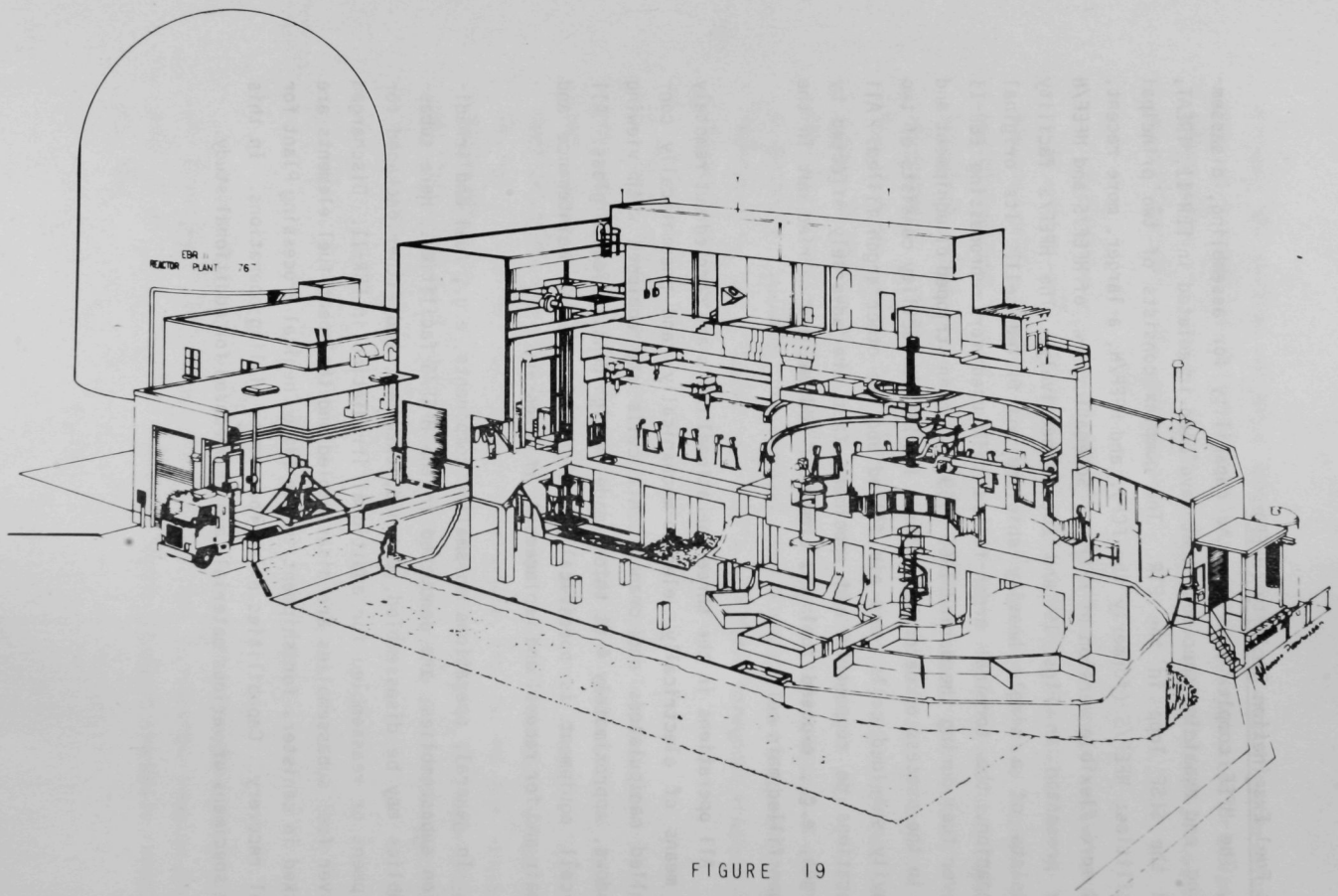


FIGURE 19

Cutaway view of the Hot Fuel Examination Facility/South
(formerly the Fuel Cycle Facility)

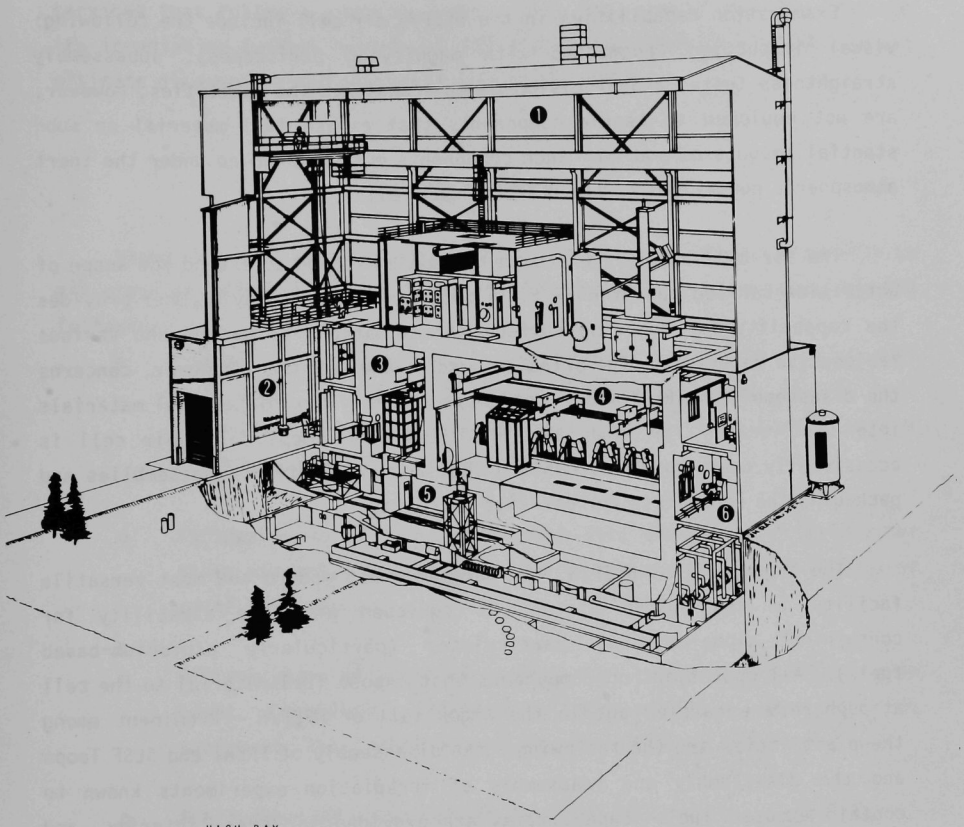


FIGURE 20

Cutaway veiw of the Hot Fuel Examination Facility/North, a larger more recent version of the Hot Fuel Examination Facility/South

Examination capabilities in the HFEF/S air cell include the following: visual inspection (augmented with magnifying periscopes), subassembly straightness tests, and precision gamma scanning. The facilities, however, are not equipped to handle components that expose fuel material or substantial amounts of sodium. Such components must be handled under the inert atmospheric conditions of the HFEF/N argon cell.

The air cell at HFEF/N is used to supplement and to extend the scope of activities carried out at HFEF/S. A decontamination-spray chamber provides the capability for removing surface contaminations of sodium and various radioactive materials. The principal use of the facility, however, concerns the disassembly of TREAT loops and the assembly of irradiated fuel materials into test trains for both the TREAT and SLSF loops. The air cell is occasionally used for disassembling fuel and irradiation subassemblies and packaging the components for off-site shipment.

The argon cell at HFEF/N, the largest, most modern and most versatile facility in the United States, is equipped with the capability for containing alpha-emitting particulates (particularly plutonium-based fuels). All operations on components that expose fuel material to the cell atmosphere are carried out in the argon-cell of HFEF/N. Prominent among these activities are the following: the disassembly of TREAT and SLSF loops and the disassembly and reassembly of irradiation experiments known to contain exposed fuel. Capabilities are provided for pre-, interim-, and postirradiation examination of fuel specimens irradiated in EBR-II. Similar capabilities are provided for the pre- and postirradiation examination of fueled materials in TREAT and SLSF loops. Examination capabilities include the following: neutron radiography in the NRAD facility (a General Atomics TRIGA reactor), precision gamma scanning, visual inspection (augmented with magnifying periscopes), profilometry, eddy-current testing (for cladding flaws), precision weighing, subassembly straightness testing, precision milling, specimen preparation, and specimen sampling. Collectively, facilities that form the HFEF complex provide

services that follow a given experiment from its initial assembly, through its irradiation (either in EBR-II, TREAT or in the SLSF in the ETR), to its ultimate disassembly and postirradiation examination.

11.0 ACHIEVEMENTS

Since its initial operation in 1964, EBR-II has contributed heavily to all areas of FBR technology. Among EBR-II's principal achievements are the following:

- Remote reprocessing and refabrication of EBR-II driver fuel equivalent to about four core loadings, some 40,000 fuel elements, with an average turnaround time of 45 days and a minimum of 29 days (September 1964 - April 1969).

- Fifteen years of successful experience with the sodium-to-water steam generator (evaporators and superheaters) with no leakage of water into sodium.

- Fifteen years of experience with under-sodium components such as pumps, flowmeters, intermediate heat exchanger, fuel handling equipment, etc., with only minor and repairable problems.

- Achievement of annual plant capacity factors that compare favorably with the best performance of commerical power plants - while operating as the Nation's fast-flux irradiation facility. Plant capacity factors as high as 76.9% have been achieved.

- Use of EBR-II plant steam for domestic heating at the ANL-W site since September 1974.

- Through December 1978, over one billion kilowatt-hours of electrical generation. EBR-II supplies approximately 14 MWe of electrical power to the INEL grid.

- Extensive experience in maintenance of sodium components; practical demonstration that maintenance and repair of such components can be accomplished by straightforward techniques, with relatively simple equipment, and without undue hazard to personnel.

- Operation of the plant with minimal release of radioactivity to the environment.

- Replacement of a portion of the EBR-II radial uranium blanket with a stainless steel reflector to enhance the core environment for experimental irradiations.

- Irradiation of 571 experimental subassemblies through December, 1978. These contained 9782 individual experiments, including 2918 mixed-oxide fuels, 623 mixed-carbide or nitride fuels, 4528 metal driver fuels, 1188 cladding and structural materials, 212 absorber materials, and 313 miscellaneous specimens.

- Improvement of EBR-II metallic driver fuel element design and increase of fuel burnup capability to 10 at.%; increase of fuel burnup limit for ANL-produced fuel (Mark-II) from 1 at.% to 8 at.%.

- Irradiation of mixed-oxide fuel to a heavy atom burnup of 20.5 at.%. Attainment of fuel element cladding temperature of 1500°F (815°C), accumulated neutron fluence of 1.7×10^{23} nvt on structural material specimens, and 7×10^{21} captures/cm^3 for absorber materials.

- Design, installation, and operation of the following experimental facilities: Instrumented Subassemblies (INSAT); Instrumented In-Core Facilities (INCOT); the Radioactive Sodium Chemistry

Loop (RSCL) for testing sodium-quality-monitoring instruments in primary sodium; the Nuclear Instrument Test Facility (NITF) for testing nuclear instrument sensors and cabling; and systems for testing fission-product detection instruments.

- Irradiation of over 200 individual experiments (capsules, creep specimens, fuel elements, sensors, etc.) in instrumented in-core facilities.

- Development and demonstration of a system of hydrogen-meter leak detectors to monitor for leakage between water and sodium systems in the EBR-II steam generator.

- Development, demonstration, and beneficial utilization of failed-fuel element detection and location (FEDAL) techniques and equipment, including delayed-neutron monitoring in the primary sodium coolant.

- Development, demonstration, and beneficial utilization of a failed-fuel location technique based on unique mixtures of xenon isotopes ("xenon tags") for rapidly identifying the sources of fission-product releases.

- Design, construction, and operation of the Cover Gas Cleanup System (CGCS), incorporating cryogenic distillation to remove and concentrate rare-gas fission products from the cover gas.

- Development, prove-out, and routine use of a versatile system of computer programs for thermal-hydraulic-nuclear considerations; modeling of whole-plant thermal-hydraulic behavior; conduct of convective-flow tests in EBR-II, utilizing INSAT and INCOT facilities.

- Training and information services for personnel of other national breeder program organizations and of the academic nuclear community.

- Fast reactor familiarization training for FTR operating personnel.

- Investigation, monitoring, and explanation of thermal- and fluence-related effects in fast reactor core hardware; e.g. subassembly bowing and swelling.

- Development and application of techniques and equipment to characterize and monitor fast reactor kinetic behavior.

- Identification and investigation of major safety and availability issues surrounding sodium-cooled fast reactor operation with breached fuel elements; conduct of a vigorous run-beyond-cladding-breach testing program.

- Development and defense of safety philosophy and documentation supporting operation of sodium-cooled fast reactors, resulting from safety-related experimental programs and modifications in EBR-II; development of the first set of technical specifications for a fast reactor power plant in the U.S.

- Development, testing, and application of novel sodium-sampling and impurity-measuring equipment to monitor and control sodium purity in the EBR-II systems; characterization of cold-trap performance in the practical plant environment; monitoring of fission-product tritium transport; development of a trap for ^{137}Cs released to the sodium coolant from breached fuel elements.

- Development, testing, and application of diagnostic techniques and instrumentation for characterization of sodium-cooled fast reactor operating conditions (reactivity meters, thermal-expansion difference thermometers, gamma expansion difference monitors, self-powered neutron detectors, pulsed-neutron activation flow measurement, etc.).