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FAST REACTOR OPERATION IN THE UNITED STATES

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ABSTRACT

Of the many American facilities dedicated to fast reactor technology, six qualify as liquid-metal-cooled fast reactors. All of these satisfy the following criteria: an unmoderated neutron spectrum, highly enriched fuel material, substantial heat production, and the use of a liquid metal coolant. These include the following: EBR-I, Clementine, LAMPRE, EBR-II, EFFBR, and SEFOR. Collectively, these facilities encompassed all of the more important features of liquid-metal-cooled fast reactor technology. Coolant types ranged from mercury in Clementine, to NaK in EBR-I, and sodium in the others. Fuels included enriched-uranium metallic alloys in EBR-I, EBR-II, and EFFBR; metallic plutonium in Clementine; molten plutonium alloy in LAMPRE; and a mixed UO_2-PuO_2 ceramic in SEFOR. Heat removal technique ranged from air-blast cooling in LAMPRE and SEFOR; steam-electrical generation in EBR-I, EBR-II, and EFFBR; to a mercury-to-water heat dump in Clementine. Operational experience with such diverse systems has contributed heavily to the U.S. breeder reactor program.

Each of the six systems is described from the viewpoints of purpose, history, design, and operation. Attempts are made to limit descriptive material to the most important features and to refer the reader to a few select references if additional information is needed.

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1.0 Introduction

The birth of fast breeder reactor technology may be easily traced to the war years when intensive efforts were devoted to the subject of nuclear fission. Obvious, even then, was the possibility of fuel breeding but little of a tangible nature emerged during this period to demonstrate the validity of the concept.

In the period immediately following the war the interests of the scientific community turned to the peaceful applications of nuclear technology. To expedite the development of peaceful applications the President of the United States, on August 1, 1946, signed into law the Atomic Energy Act which effectively transferred nuclear matters from military to civilian control. Shortly thereafter, on October 28, 1946, the President announced the appointment of a five member commission to administer the nation's nuclear affairs. On January 2, 1947, the United States Atomic Energy Commission became the custodian for extensive research and production facilities at thirty-seven installations scattered among nineteen states.[1]

To hasten and promote the peaceful applications of nuclear technology a government-supported system of National Laboratories was developed in the late 1940's. Of these Los Alamos Scientific Laboratory, Los Alamos, New Mexico, and Argonne National Laboratory, Argonne, Illinois, were assigned the principal responsibilities for pioneering early fast reactor technologies. Of the many facilities built and operated by these laboratories, four qualify as liquid-metal-cooled fast reactors since these satisfied the following criteria: an unmoderated neutron spectrum, highly enriched fuel, sensible heat production and the use of a liquid metal coolant. The design, construction and operation of these, EBR-I and EBR-II (Argonne), and Clementine and LAMPRE (Los Alamos), are discussed below.

In the private sector two fast reactor facilities deserve particular recognition. One of these, the Enrico Fermi Fast Breeder Reactor (EFFBR), was designed, built, and operated at Laguna Beach, Michigan, by a consortium of two nonprofit organizations (Atomic Power Development Associates and the Power Reactor Development Company), and funded by the Detroit Edison Company, a private stockholder-owned utility.

The other venture in the private sector was SEFOR (Southwest Experimental Fast Oxide Reactor), a highly specialized experimental facility that was designed, built, and operated as a joint venture of the Southwest Atomic Energy Associates, the Karlsruhe Laboratory of West Germany, Euratom, and the General Electric Company. Research and development effort was supported by the United States Atomic Energy Commission. The design, operation, and construction of these facilities, i.e., EFFBR and SEFOR are also described below.

Collectively, the facilities described below encompassed many facets of liquid-metal-cooled fast reactor technology. Coolants ranged from mercury in Clementine, to NaK in EBR-I, and sodium in the others. Fuels included enriched uranium metallic alloy in EBR-I, EBR-II and

EFFBR; metallic plutonium in EBR-I and Clementine; molten plutonium alloy in LAMPRE; and a mixed UO_2-PuO_2 ceramic in SEFOR. Heat dissipation methods embraced air-blast cooling in SEFOR and, LAMPRE; steam-electrical generation in EBR-I, EBR-II, and EFFBR; and a mercury-to-water heat dump in Clementine. Clearly, operational experience with such diverse facilities guided the selection of fuels, coolants, and materials for follow-on systems.

In compiling the information summarized below an attempt was made to limit descriptive effort to the most important features and to refer the reader to a few select references if additional information is needed. In general, the following matters are addressed for each of the six facilities: purpose, history, design, construction, and operation. Insofar as possible, each of the facilities is addressed in approximately the order of development.

2.0 EBR-I

2.1 Purpose

EBR-I was designed, built, and operated by Argonne National Laboratory to: prove the validity of the breeding principle, evaluate the feasibility of using liquid metal coolants, provide measurable quantities of high-purity ^{239}Pu, demonstrate the control characteristics of a fast-neutron system, accumulate operating experience with NaK-to-water heat exchangers, and establish the feasibility of transforming nuclear heat into electrical energy.

2.2 History

Prominent among those who envisioned fuel-breeding as a promising solution to future power shortages were E. Fermi and W. H. Zinn.[2] Their approach to the technology was simple: minimize the fraction of neutrons lost by parasitic capture to the coolant, moderator, and fuel; and maximize productive captures in massive uranium

blankets. Although sophisticated neutron cross-section data were lacking at that time (circa 1943) enough information did exist to support the conclusion that parasitic neutron losses could be sharply reduced if the average neutron energy remained high. Such a requirement imposed a variety of constraints, among the most important of which was the complete absence of conventional coolant-moderator materials. Attention was, accordingly, directed to liquid-metal coolants, in particular to NaK (the sodium-potassium eutectic mixture). NaK had the obvious advantages of being a liquid at room temperature and having excellent heat transfer properties. Furthermore NaK was considered nearly ideal from the viewpoint of neutron economy; it was both a poor moderating and a poor absorbing material.

At the urging of Fermi, Zinn, in late 1944, began planning a small-scale proof-test facility for proving the validity of the breeding principle and for evaluating the feasibility of using a liquid metal as a coolant. General plans for the facility, which eventually became known as Experimental Breeder Reactor I (EBR-I), were reasonably complete by late 1945.

The years 1945 to 1949 were spent in firming up matters of nuclear and engineering design. Many imponderables appeared. In addition to a paucity of fast neutron physics data little, if anything, was known about liquid-metal pumping, sodium and potassium corrosion, NaK-to-water heat exchangers, and the behavior of fuel, cladding and structural materials under the hostile environments of radiation and temperature. Nevertheless, design parameters were fixed, on the basis of the best information available, and sometimes as a matter of intuition and judgment.

Construction began at the National Reactor Testing Station (since renamed Idaho National Engineering Laboratory) in Idaho in October 1949. Criticality was achieved in August 1951 and full power operation at 1.1 MWt was reached on December 19, 1951. Shortly thereafter, steam

was led to a turbine-generator and for the first time, on December 20, 1951, nuclear heat was transformed into electrical energy.

2.2 Design[3]

The reactor consisted of three principal regions: a core, a light inner blanket that surrounded the core axially and radially, and a denser cup-shaped outer blanket.

Inner blanket rods consisted of cylindrical rods of natural uranium; fuel rods consisted of cylindrical rods of unalloyed, fully enriched, uranium metal. The outer reflector consisted of 84 one-hundred pound, keystone-shaped, steel-clad uranium bricks, arranged in the form of a cup, and mounted on a hydraulically driven pedestal. Separating the cup and the core was a double-walled tank system. Raising or lowering the cup provided coarse control of reactivity through the reflection of leakage neutrons. Under scram conditions the cup, worth approximately 5% $\Delta K/K$, was dropped under gravity by releasing hydraulic pressure to the pedestal ram.

Twelve two-inch vertical holes in the cup accommodated stainless-steel-clad natural uranium rods. Eight of the twelve were positioned fully-in during operation. These were rigged to fall under spring-assisted action in the case of scram. The remaining four rods were used for fine reactivity control.

Heat generated in the cup and control/safety rods was removed by the forced circulation of air through a series of vertical holes in the reflector pieces. As it turned out, reflector cooling proved to be the factor that limited reactor power, nominally designed for 1.2 MWt. Surrounding the cup were a 19-in.-thick graphite reflector and a concrete shield approximately 9 ft in thickness.

The core and inner blanket were cooled by NaK which flowed by gravity from an elevated supply tank, upwards through the reactor, through a primary-secondary heat exchanger, and into a receiving tank. A pump, operating at a slightly higher capacity than reactor coolant flow, returned the coolant to the gravity supply tank. An overflow system connected the gravity supply tank to the receiving tank. Such a feature was beneficial in two ways: by providing a constant delivery head, and by assuring 30 min of gravity-delivered flow under emergency conditions.

Heat from the secondary side of the heat exchanger was removed in two ways: through the generation of superheated steam or through a fan-cooled NaK-air heat exchanger. Under nominal full power operating conditions enough electrical power was generated (approximately 200 kWe) to satisfy the building demand.

2.3 Operation

2.3.1 Mark-I Loading

In the course of its useful life, EBR-I operated with four different fuel loadings. The first of these, Mark-I, was fueled with cylindrical slugs of fully enriched uranium metal contained in Type 347 stainless steel tubes.[3] NaK in the annuli between the slugs and the tubes provided an excellent heat transfer medium. One of the principal features associated with the Mark-I loading was a gradual loss of reactivity (beyond that expected for fuel burnup). The loss was correctly attributed to axial fuel growth and after 3.5×10^6 kWt-hr of energy production, operations with this loading were terminated.

Valuable information was derived from the Mark-I loading. Measurements of the breeding ratio demonstrated conclusively the feasibility of breeding.[4] Although a value of only 1.01 ± 0.05 was established it was clear that higher values could be achieved by reducing neutron leakage and by fueling the core with ^{239}Pu.

The operation of the Mark-I loading also confirmed the theoretical prediction that the neutronic behavior of both fast and thermal systems below prompt critical should be identical. Other important contributions appeared in the forms of realistic tests of fast reactor instrumentation, the production of "super-pure" plutonium in the outer blanket, and the demonstration that NaK, a liquid metal, posed no problems for pumping both centrifugally and electromagnetically.

2.3.2 Mark-II Loading

To study the effects of alloying on radiation resistance, a second loading, Mark-II, consisting of uranium-2% zirconium metallic alloy was installed in February 1954. As expected the inclusion of zirconium significantly enhanced the radiation resistance of the fuel. Reactivity losses were found to be consistent with burnup considerations; no anomalies were encountered as with the Mark-I loading.

A peculiarity shared by the Mark-I and Mark-II loadings was a tendency for reactor power to oscillate whenever reactivity or coolant flow was varied rapidly. Although the origin of the instability was obscure, circumstantial evidence pointed to the complex coupling of two dominant feedback effects: one prompt and positive; and the other strongly delayed and negative. To investigate these matters the final phase of the experimental program for Mark-II was devoted to an analysis of reactor performance under various conditions of power and flow. Upon completion of these experiments, operation of the reactor was to be terminated and the plant placed in standby status. A planned transient test with the main coolant flow stopped demonstrated the existence of a prompt positive power coefficient and led to an unintentional partial meltdown of the core.[5]

As a consequence of the melt-down incident considerable concern was expressed for the safe operation of future fast-breeder reactors. To prove there was nothing intrinsically unsafe in the operation of a fast reactor the damaged core was replaced with one (Mark-III)

specifically designed and sufficiently versatile to study in detail feedbacks originating from fuel, coolant and structural expansions.[6] As the result of a comprehensive experimental test program it was concluded that those features responsible for the instability noted in earlier loadings could be completely eliminated by rather elementary changes in mechanical design.[7 8]

2.3.3 Mark-IV Loading

The fourth and final loading in EBR-I was fueled entirely with metallic plutonium, with a small (1.25 wt %) inclusion of aluminum. The use of plutonium introduced a variety of problems that required scrutiny prior to loading and operation.[9] Although many problems were of a conventional nature others were complicated by physical, chemical and neutronic properties peculiar to plutonium. The low melting point of the fuel, its tendency to deform under stress, its toxicity, and its small value of β constituted sources of potential hazard not encountered in ^{235}U fueled systems.

Benefits derived from the operation of EBR-I with plutonium fuel included the following: the assurance that there is nothing inherently hazardous in the operation of a plutonium fueled system,[10] and proof that the breeding ratio of a plutonium-fueled system can significantly exceed that of a system fueled with ^{235}U. In a series of intensive foil-activation experiments a value of 1.27 ± 0.08 was measured.[11 12]

Following completion of the Mark-IV tests in 1964, EBR-I was shut down, decommissioned, and in 1966 declared a national historic landmark under the stewardship of the U. S. Department of the Interior.

3.0 CLEMENTINE

3.1 Purpose

Clementine was designed, built, and operated by Los Alamos Scientific Laboratory to: serve as a critical facility for investigating the effects of fuel configuration on criticality, provide a copious source of unmoderated neutrons for physics studies, study and evaluate the feasibility of controlling fast neutron systems, and provide information relevant to the use of plutonium as a fuel for follow-on fast breeder reactors.

3.2 History[13 14]

The concept of Clementine, like EBR-I, originated during the war years. As the war years came to a close, physicists, realizing the potential of breeding systems, initiated conceptual design studies on a modest facility that would demonstrate the feasibility of operating a power-producing, plutonium-fueled system under fast neutron conditions.

The construction of Clementine was formally proposed in November 1945, approval was given the following month, and design work began immediately. Construction began in September 1946, and although technically still under construction, criticality experiments were performed as early as November 21, 1946. Construction effort resumed during February 1947, and continued intermittently until January 1949. In March 1949, the system was brought to full operating power at 25 kWt. For historical accuracy Clementine enjoys the distinction of being the world's first liquid-metal-cooled fast reactor.

3.3 Design

The fuel material consisted of delta-phase plutonium machined into rods 0.647 in. in diameter and 5.5 in. long. Tubes of Type 1020 steel, 0.020 in. in diameter and 5.5 in. long were used for cladding. Reflector rods containing natural uranium were fabricated to the same dimensions and had the same cladding. Fuel and reflector rods were

fixed on a hexagonal pitch between a system of upper and lower positioning plates which permitted the flow of coolant. When fully assembled for operation the core consisted of 35 fuel and 20 reflector rods arranged in a cylindrical bundle approximately 5.9 in. in diameter. The core assembly, i.e., the fuel and reflector rods, was contained in a 46-in. long, 6.2 in. outside diameter, mild-steel cylinder, 0.10 in. thick. Immediately above the core, and extending to the top of the containment cylinder was a removable reflector and shield plug. The containment cylinder, or pot, was surrounded by a 6-in. uranium reflector and 4 in. of lead. Additional shielding consisted principally of layers of borated plastic, steel and, finally, concrete.

Shutdown reactivity was effected by a large block of uranium located directly under the core. Raising the uranium block increased reactivity through increased reflectivity. Dropping the block under gravity provided scram capability. Fine control of reactivity was provided by two uranium rods which moved vertically outside the pot in the radial reflector. Two safety or shim rods, also located in the radial reflector, were composed of sections of uranium and ^{10}B. When operating, the uranium section was inserted in the reflector. During shutdown the rods were lowered and reduced reactivity even further by the insertion of the boron.

3.4 Operation

Mercury at a rate of 2.4 gal/min and a velocity of 0.16 ft/sec was pumped by an eddy-current type of electromagnetic pump to a plenum beneath the core. The coolant flowed upward around the fuel rods into an upper plenum and then to a mercury-water heat exchanger. Mercury entered the core at 100°F and exited at 250°F. Central fuel temperatures amounted to approximately 275°F.

Criticality at 1 W was reached on November 21, 1946. Several low-power tests (at 1 W) were completed in the months immediately following. These included a measurement of critical mass, an evaluation

of control problems, and measurements of the isothermal temperature coefficient, neutron spectrum and delayed neutron fraction.

During the period December 1946 to December 1947, work on the biological shield was completed. Additional measurements made with the shield installed included the effects of alpha-phase plutonium on reactivity, neutron distributions in the core and reflector, danger coefficients, and activation cross sections.

Final assembly and closure with the mercury system operating was achieved in February 1949. Shortly thereafter, in March 1949, the system was brought to full power at 25 kWt. From March 1949 until December 1952, Clementine served as a highly intense source of unmoderated fission neutrons for a variety of physics experiments. Its operation during this period was highly successful since it led to refinements in cross-section data and neutron parameters. In addition, the operation of Clementine at power demonstrated the ease of control action for plutonium-fueled systems.

In December 1952, a fuel rod ruptured and plutonium entered the coolant system. Since all original objectives had been met, Clementine was shut down and dismantled.

4.0 LAMPRE

Among the early fast reactors, the most visionary was LAMPRE, a system characterized by the use of plutonium fuel in the molten state.[15 16] By operating with the fuel in the molten state a core fueled with plutonium could be operated at higher temperatures. Plutonium alloyed with 10 at.% iron melts at a relatively low temperature, i.e., approximately 800°F. Another obvious advantage includes the elimination of structural damage in the fuel under intense radiation conditions.

4.1 Purpose

LAMPRE was designed, built, and operated by the Los Alamos Scientific Laboratory to: demonstrate the feasibility of using molten plutonium alloys as a fuel material for fast reactors, evaluate the relative merits of various binary and ternary plutonium alloys, and investigate the compatibility of molten plutonium alloys with various containment materials.

4.2 History

Research and developmental activities including criticality studies began in 1957. Installation effort was completed in 1959. Wet criticality was reached during March 1961 and the system was operated at its design power level of 1 MWt shortly thereafter. After approximately 60 hr of full-power operation a pronounced accumulated reactivity loss was noted. This was promptly identified as the result of a fission-product gas accumulation that tended to form bubbles with carbon and plutonium-carbide additives. The system was shut down in January 1962 and the core was unloaded.

A second loading, without the additives, reached criticality in April 1962. Operation with the second core continued until July 1963. The second core was unloaded in April 1964. Subsequent experiments with ternary plutonium alloys were conducted. Operations continued until program cancellation in 1965.

4.3 Design

The principal problem in the development of LAMPRE was the design and fabrication of capsules that could provide adequate containment. Material choices were eventually narrowed to various tantalum alloys or to "super-pure" tantalum. The fuel used in the first loading consisted of a mixture, 90 at.% plutonium and 10 at.% iron. Excess carbon in the form of elemental carbon and carbides was added to inhibit corrosive attack of the tantalum. Capsules used to contain the fuel

mixture were 8.312 in. long, 0.376 in. inside diameter, and 0.025 in. thick. Filling the capsules was a simple operation. Slugs of the fuel material were machined to fit the capsules and were inserted into the capsules immediately after machining.

The core consisted of approximately 140 capsules filled with fuel mixture and 60 dummy stainless steel capsules (for reflection and shielding). Capsules were fixed to complex handles which served several functions: as an upper reflector, as a flow collector, as a locking device, and finally as a device for removing and inserting the capsules. All capsules, both reflector and fuel, were fixed at the lower ends by a locator plate.

Sodium, used as a coolant, was pumped by two parallel ac electromagnetic pumps downward through an annulus between the primary vessel and a flow separator. The stream reversed at the bottom, passed upward through the locator plate, around the capsules, through the top reflector region (rod handles), and into an outlet plenum. After leaving the reactor the coolant flowed through a finned air-cooled heat exchanger. Coolant entered the lower plenum at 842°F and exited at 1045°F. The maximum temperature in the fuel mixture was 1598°F.

Control of the system was achieved by reflector motion. Coarse control was effected by the vertical motion of an annular reflector, 20 in. outside diameter and 10.74 in. inside diameter. Fine control relied on four nickel rods 16 in. long and 3.8 in. in diameter which moved individually in the annular reflector.

4.4 Operation

Aside from problems of fuel-frothing, bubble accumulation, reactivity loss, and capsule failure the operation of LAMPRE was characterized by extreme stability. At no time was there any evidence of control problems originating from frothing and bubble accumulation.

In the second loading carbon and carbide additives were omitted. As a result fission product gases tended to disengage from the molten fuel much more cleanly. Reactivity losses noted with the second core were, for a given power accumulation, reduced by a factor of approximately three.

Operation with the second core continued uneventfully until September 1962 when a capsule failure released approximately 75 g of fuel material into the coolant. A few months later two additional releases were noted. In all three cases fuel was released rapidly under pressure (145 to 370 psi). Although the coolant system was contaminated no evidence of gaseous fission products in the operating area was indicated. In all cases defective capsules were identified and removed.

LAMPRE continued to operate successfully until its decommissioning in 1965.

5.0 EBR-II (Experimental Breeder Reactor II)

The initial success of EBR-I in the early 1950's encouraged more visionary plans for the exploitation of the fast breeding principle. In response to a request from the USAEC, ANL in 1953 submitted a proposal for a second and larger liquid-metal-cooled fast reactor.[17] This system, later known as EBR-II,[18 19] was conceived as a pilot plant and as an intermediate step between EBR-I and a full-scale commercial fast breeder. Although much useful information was derived from EBR-I additional information was needed for the design of much larger systems.

5.1 Purpose

EBR-II was designed and operated by Argonne National Laboratory to: demonstrate the feasibility of a sodium-cooled fast reactor in a power-generation system; establish the feasibility of "closing the fuel cycle" through on-site fuel reprocessing; operate such a reactor under prototypal conditions of power density; accumulate operating experience

with sodium system components, viz., pumps, heat exchangers, valves, flowmeters, etc.; and demonstrate the inherent safety characteristics of metal-fueled fast reactors.

5.2 History

Site preparation began in October 1957 and construction started in December 1957. Plans for support facilities such as laboratory facilities, guard house, fire house, pump house, etc. were firmed up by April 1958 and construction began in June 1958. By October 1958, construction of the power plant and reactor plant was underway. Plans for the fuel cycle facility and sodium boiler plant were firmed up by March 1959 and construction began in July 1959. By May 1961, installation and construction effort was sufficiently near completion to permit the planning of dry critical experiments (without sodium coolant).

In late 1961, dry critical experiments were conducted. The principal objectives of these included the following: a measurement of the dry critical mass which, when compared with the wet critical mass, would lead to an evaluation of the overall sodium coefficient; verification of reactor instrumentation, fuel handling equipment, control and safety rod drives, shielding, etc.; and the training of operating personnel. During the tests construction work continued on unaffected systems such as the secondary sodium and steam systems and the fuel cycle facility.

Following the completion of the dry critical experiments in November 1961, the plant was readied for sodium loading and wet criticality measurements. Dry heating tests were made in March and April 1962 at 350°F to test the alignment of fuel handling equipment with respect to the core. In February 1963, the seal troughs of the rotating plugs were filled with tin-bismuth alloy. The primary tank was then purged with nitrogen to reduce the oxygen concentration to approximately 0.5%. Sodium filling (86 000 gal) was completed in March 1963 and the temperature of the sodium was raised to 650°F. The approach to wet criticality

began on October 30, 1963, with a fuel loading of 47.9 kg ^{235}U.[20] Criticality was reached on November 11, 1963, with a fuel loading of 181.2 kg ^{235}U.

By June 1964, all systems including the secondary sodium, steam, and turbine generator were ready for operation. The approach to power was begun on July 16, 1964, and a power level of 37.5 MWt was reached on October 13, 1964. Reactor power was raised to 45 MWt on March 27, 1965, to 50 MWt on August 26, 1968, and to 62.5 MWt in September 1969.

5.3 Design

The design of EBR-II is based on the pot-type concept, i.e., all major primary components are immersed in a large double-walled tank that contains 86 000 gal of sodium at 700°F.

Two centrifugal pumps, each rated at 4500 gal/min, take suction from the bulk sodium. The pumps discharge sodium under a pressure of 45 lb/in.^{2} to a plenum under a grid plate that serves two principal purposes: to provide support for core and blanket subassemblies and to regulate the upward flow of coolant. Coolant flows upward through the core and surrounding blanket, into an upper plenum, through an outlet pipe, and into the intermediate heat exchanger. From here the primary coolant flows directly to the bulk sodium. Coolant enters the heat exchanger at 883°F and leaves at 695°F. A dc electromagnetic pump, rated at 500 gal/min, is located in the outlet pipe. This pump operates continuously with the specific purpose of removing decay heat in the event of a primary pump coastdown.

The secondary sodium cooling system is isolated from the primary system at the heat exchanger. Secondary sodium at 586°F is pumped through the heat exchanger at a rate of 5000 gal/min. The exit coolant, at a temperature of 872°F, is pumped by a water-cooled ac linear-induction pump to the sodium-boiler plant. Here the thermally hot but

essentially nonradioactive secondary sodium is passed through two superheaters, eight evaporators, and into a surge tank from which the coolant is pumped back to the intermediate heat exchanger to complete the cycle.

Superheated steam at a temperature of 815°F and a pressure of 1250 lb/in.[2] is piped to a 20-MW turbine-generator located in the power plant building. Under full-power operating conditions, 19.5 MW of electrical power is generated.

Reactivity is controlled by eight fueled control rods with boron-loaded followers and two fueled safety rods. Four other control rod locations are currently occupied by a single nonfueled drop rod for kinetic tests and three instrumented in-core test facilities. The reactivity worths of control and safety rods are respectively, 0.53 and 0.40% Δk/k. Any one of the eight control rods may be used for fine reactivity control; all are discharged from the core under scram conditions. The two safety rods are always fully inserted; their principal purpose is to provide shutdown capability in the fuel-handling mode.

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

5.4 Operation

The operation of EBR-II has always been smooth and relatively trouble-free. A relatively prompt negative power coefficient of reactivity, i.e., 3.6×10^{-5} Δk/k/MW, effectively damps the effects of small

reactivity changes caused by inlet temperature variations, control rod motion, etc.

EBR-II, now in its fourteenth year of operation, continues to serve as the nation's facility for testing fuels and materials under fast reactor conditions. The use of EBR-II as an irradiation facility for the United States breeder reactor program began in May 1965 with the insertion of two experimental subassemblies that contained various structural specimens and fuel rods (mixed PuO_2-UO_2 and U-Pu alloys). Since that time, the complement of experimental subassemblies in the reactor has grown to as many as 65. Up to June 1977, a total of 9033 individual experiments were either completed or in the process of being irradiated. (An experiment is understood to be a single fuel element or a capsule.) Of the 9033 experiments, 2906 were mixed-oxide fuels; 463 were carbides, nitrides, or cermets; 4442 were metallic fuels; 1027 were various cladding and structural materials, and 195 were control materials, e.g., B, Eu, Ta, etc. In the course of the irradiation program, peak burnups and cladding temperatures of 17 at.% and 1500°F, respectively, have been achieved for mixed-oxide fuels. Peak fluences of 1.7×10^{23} nvt have been reached for structural materials and plenum pressures as high as 2900 lb/in.^{2} have been achieved for metallic fuel elements.

For the past three years EBR-II has operated with the following plant capacity factors: 1975, 66.1%; 1976, 76.9%; 1977, 71.5%. At the present time EBR-II is still serving as the nation's leading facility for testing fuels, cladding and structural materials, and instrument sources under near-prototypal conditions for future fast reactors.

6.0 EFFBR (Enrico Fermi Fast Breeder Reactor)

The design, construction, and operation of the EFFBR represent the first attempt in the United States to commercialize the sodium-cooled fast reactor as a system for electrical power generation. Unfortunately, an error in mechanical design led to the partial melting of several core

subassemblies. The costs of subsequent recovery operations, coupled with others beyond the funding capabilities of the owner-operators led to the premature retirement of the plant in 1972.

6.1 Purpose

The EFFBR was designed, built, and operated to: evaluate the economics of operating a commercial size sodium-cooled fast reactor for generating salable amounts of electricity, establish licensing and regulatory criteria for follow-on reactors, and evaluate the performance of large-scale plant systems.

6.2 History

Interest in the construction and operation of a commercial size fast breeder reactor power plant began in the early 1950's when industrial groups were invited by the USAEC to consider the economic and technical feasibility of generating electric power in commercial quantities with nuclear-fueled systems. As a result APDA (Atomic Power Development Associates), a nonprofit research group, was organized in March 1955 to formulate the conceptual development of promising systems. In August 1955, a second nonprofit organization, PRDC (Power Reactor Development Company), was formed to design, build, and operate a plant which was eventually named the EFFBR.[21]

A provisional construction permit was issued by the USAEC in August 1956 and work began immediately. For the next five years vigorous attempts were made by various intervenors to halt construction activities. Despite such attempts, construction continued and in June 1961, the U.S. Supreme Court cleared the way for plant completion and ultimate operation.

Nuclear operation at low power (approximately 1 MWt) began in August 1963 and continued throughout 1965, pending licensing for operation up to 200 MWt.[22] A provisional license to operate up to a power level of 200 MWt was granted in December 1965.

The first nine months in 1966 were devoted to the testing of plant systems under 100 MWt operating conditions. On October 5, 1966, during a power-increase test to 67 MWt, evidence of off-normal fission-product activities and fuel temperatures was noted. The plant was shut down and an intensive program was implemented to investigate, identify, and rectify the cause of what was almost certainly a flow-blockage condition.[23]

Two years were needed to implement the necessary remedial actions. A formal report of the flow-blockage incident was submitted to the USAEC in January 1969, and petition was made to continue with the power ascent program. Permission was granted in February 1970. Criticality was achieved in July 1970 and in October 1970 the plant was again operated at 100 MWt. Full licensed power of 200 MWt was reached on October 26, 1970. Operations continued with no evidence of previous difficulties until October 1972 when the plant was shut down in anticipation of decommissing activities.

On November 27, 1972, the PRDC Executive Committee made the difficult decision to decommission the plant since it was evident that funding for a $50 000 000 uranium-oxide fuel-upgrade program could not be arranged. Of the $50 000 000 needed, $30 000 000 was pledged by American utilities and Japanese and Western European industrial groups. Had the requisite funding been realized the EFFBR could very well be operating today.

6.3 Design

Like EBR-II, the EFFBR was fueled with an enriched uranium metallic alloy, viz., uranium-10 wt % molybdenum with enrichment of 25.6 wt % ^{235}U. Zirconium was used as the cladding material. Fuel pins, clustered in groups of 140 on a square lattice and encased in stainless steel wrapper cans constituted a fuel subassembly. Upper and lower axial blankets in each fuel subassembly consisted of 16 stainless

steel-clad uranium-3 wt % Mo alloy rods with a ^{235}U content of 0.35 wt % The basic core configuration consisted of a cylindrical fuel portion 31 in. in diameter, 31 in. high, with upper and lower blanket sections each 17 in. in height. Surrounding the core region were lattice positions normally filled with inner radial blanket subassemblies, each containing 25 U-3 wt % Mo blanket rods. Surrounding the inner radial blanket were 494 lattice positions for outer radial blanket subassemblies. Between the outer radial blanket and the reactor vessel were 198 lattice positions for stainless-steel-filled subassemblies that served for thermal and radiation shielding. Shutdown and control functions were based on the use of 10 B_4C poison-type rods. Two were used for control and eight were used for shutdown reactivity.

Sodium, used as the primary coolant, flowed upward through the core and radial blankets, into a common plenum, out through three equispaced nozzles 30 in. in diameter, and into three sodium-to-sodium intermediate heat exchangers. Three single-stage centrifugal mechanical pumps, one in each loop, took suction downstream from the heat exchangers and returned coolant back to the core and blanket. Three similar pumps in the secondary sodium side circulated thermally hot but nonradioactive sodium to three once-through steam generators.

The steam generators were vertical shell, combination cross and counterflow, once-through units having water and steam flow within the shell side and sodium on the outside. Each generator had a designed heat transfer capacity of 150 MWt.

Each tube bundle was composed of 1200 combination involute-serpentine single-walled tubes, each 5/8 in. in outside diameter with a wall thickness of 0.042 in. All tubes and tube-sheet material were made from 2.25% Cr-1% Mo alloy ferritic steel. Water and steam flow patterns were such that saturated steam was superheated in the outlying serpentin tube bundle.

The stainless steel reactor vessel was composed of four parts: a lower reactor vessel (which contained the core and blankets), a transfer rotor container, an upper reactor vessel, and a rotating shield plug container. The reactor vessel was 114 in. in diameter, 1.50 in. thick in the lower plenum region and 2.00 in. thick in the upper regions. The transfer rotor container, used for fuel storage and transfer, was attached to the lower reactor vessel. The upper reactor vessel, eccentric with respect to the lower vessel, was cylindrical in shape and 174 in. in diameter. The upper portion of the vessel was sealed at the top by the rotating shield plug which, in turn, supported the control mechanisms, fuel holddown mechanism, and an offset fuel handling device.

The reactor vessel was surrounded by a graphite neutron shield located inside the primary shield tank. Shielding, principally unborated and 5% borated graphite, was designed to protect components such as coolant pipes, exit port, transfer rotor drive, neutron detector tubes, heater cables, gas lines, pipe hangers, cable trays, etc.

Fuel handling operations were based on the use of a system referred to as the fuel transport facility (FTF). The FTF, which carried one subassembly at a time, consisted of a transport car and two fuel handling machines, one in the reactor building and the other in a fuel and repair building. The FTF, loaded with a fresh subassembly, was driven to the reactor building where the subassembly was transferred to the transfer rotor assembly through a fuel exit port. The rotor moved the subassembly to a position under the rotating plug and offset handling mechanism. The subassembly was then transferred to the reactor by the offset handling mechanism. Spent subassemblies were removed from the reactor in essentially the reverse manner. Spent subassemblies were steam cleaned, stored for decay, cut to remove core sections, and shipped off-site for reprocessing.

6.4 Operation

As cited above, Section 6.2, permission to operate at power levels between 1 and 200 MWt was received in December 1965. Extensive plant testing ensued for the first nine months of 1966 under power levels up to 100 MWt. Included in the tests was a 60-hr run at 100 MWt with an electrical output of 1100 MW-hr.

During this period two difficulties were encountered. The once-through sodium-water steam generators developed leaks at the tube-to-tube sheet weld joints. A much more severe problem, however, was a flow blockage condition which became apparent on October 5, 1966, during a planned power increase to 67 MWt. The initial evidence of difficulty appeared in two forms: an inconsistency between predicted and measured reactivity, and the presence of abnormal radiation levels as sensed by a gaseous fission product monitor. Upon such evidence the plant was shut down without further incident to prepare for investigative action.

As the result of intensive investigation the origin of the flow blockage was traced to a wedge-shaped zirconium piece which had become dislodged from a conical flow guide. The piece traveled under lower plenum coolant flow to a position which partially blocked entrance flow to four fueled subassemblies. As the result of flow blockage fuel material melted. Gaseous fission products were released and reactivity balance was upset by fuel rearrangement.

Approximately two years were needed to define the details of the blockage, assess the damage, and remove the dislodged zirconium piece and its sibling counterparts. A third year was devoted to relicensing negotiations with the USAEC and to preparing the plant for restart.

Approval to reload the core and to resume operation was granted in February 1970. This permitted unloading fuel subassemblies that had been in the core during the flow-blockage incident and reloading the

core with new fuel subassemblies modified to prevent similar blockage effects. Criticality was achieved in July 1970. The system was brought to 67 MWt with two-loop flow in September 1970 and to 100 MWt with three-loop flow on October 1, 1970. Tests conducted throughout the period verified the proper operation of all previously existing and modified plant systems. Modifications included an on-line malfunction detection analyzer, steam generator tube-to-tube sheet repairs, delayed neutron detectors, and core subassembly flow-guard devices.

Power was increased to 133 MWt with two-loop flow in October 1970, and finally to the licensed full power level of 200 MWt with three-loop flow on October 16, 1970. The subsequent test program under 200-MWt operating conditions verified the stability and performance of reactor and plant systems.

At 200 MWt the reactor operated with a sodium inlet temperature of 550°F and an average outlet temperature of 770°F, corresponding to a three-loop flow of 8.53×10^6 lb of sodium per hour (6667 gpm per loop). Under these conditions gross electrical generation was 65 MWe with steam at 600 psia and 730°F.

At that time fuel supply was limited. Fuel discharged as the result of the 1966 flow blockage was awaiting requalification. Plans were made for reloading the core with uranium oxide fuel and increasing the power level to the original design level of 430 MWt. Unfortunately, as cited in Section 6.2, funds for the proposed program were not forthcoming. On November 27, 1972, the decision was made to decommission the plant.

7.0 SEFOR (Southwest Experimental Fast Oxide Reactor)

The design, construction, and operation of SEFOR was prompted principally by the need to demonstrate the safety characteristics of oxide-fueled fast reactors. Funds for the design and construction of the

facility were provided in part by the Southwest Atomic Energy Associates (a group of 17 investor-owned utilities located in the southern and southwestern portions of the United States). Other organizations that contributed funds for design and construction included the Karlsruhe Laboratory of West Germany, Euratom, and the General Electric Company. Funds for research, development, and operation were provided by the USAEC.[24]

7.1 Purpose

SEFOR was designed, built, and operated to: demonstrate the existence of and to measure the value of the prompt Doppler temperature-feedback coefficient; obtain physics and engineering data at fuel compositions, temperatures, and crystalline states characteristic of prototypal operating conditions; and demonstrate the overall stability of fast oxide-fueled systems under steady-state, mild-transient, and extreme-transient conditions. All of these objectives were realized.

7.2 History

Design effort officially began in March 1964 and was essentially completed by mid-1967.[25] A preliminary safeguards report was submitted to the USAEC in October 1964, along with an application for a construction permit. Site preparation, near Fayetteville, Arkansas, began in January 1965. Upon the receipt of a provisional construction permit in September 1965, construction of the reactor containment vessel began. The reactor building was completed in May 1967, all major equipment items were installed by March 1968, and the plant was officially transferred to the SAEA in October 1968. Preoperational testing was completed in early 1969 and the plant achieved criticality shortly thereafter (May 1969). A series of power ascension tests was initiated in early 1970. These continued for approximately one year and ended with the completion of prompt criticality studies on Core I in mid-1971. Core-I fuel was replaced by a second loading, Core II, which was specifically designed for experiments under "harder" neutron flux conditions.

Experiments with Core II were completed in January 1972. Upon completion of the experimental program the facility was shut down and subsequently decommissioned.

7.3 Design

Of vital importance to the SEFOR experimental program was the need to eliminate or at least to minimize, fuel expansion-reactivity effects during transient studies. Unless this were done feedback reactivity from fuel expansion would "add" to other feedback components and prevent a clean separation of the Doppler temperature coefficient.

Two features of design effectively minimized fuel feedback effects. These were the use of dished fuel pellets, and the use of fuel rod and subassembly tighteners to prevent radial expansion effects. Fuel pellets, with a length of approximately 5/8 in. and a diameter of 0.875 in. were made from a coprecipitated mixture of PuO_2-UO_2 which had a ^{239}Pu and ^{241}Pu content of 18.7 at.%. The powder mixture was formed into dish-shaped pellets by cold-pressing and subsequent sintering operations.

The pellets, in turn, were loaded into Type 316 stainless steel cladding tubes, 0.040 in. thick, and 0.970 in. in diameter. The resulting fuel column, 33.81 in. in height, was separated at axial midplane by a spacer-spring arrangement, designed to accommodate axial expansion effects.

The basic loading units consisted of hexagonal 316 stainless steel tubes, 0.060 in. in wall thickness and 3.160 in. across the flats. These were filled with six fuel rods and a central moderator rod filled with BeO pellets. Surrounding the moderator rod was an expandable sleeve to reduce the radial motion of fuel rods during power increases.

Fuel subassemblies were supported on a 2.50-in.-thick core support plate equipped with 109 perforations, each designed for pole

piece acceptance and bayonet latching. A torsion-bar clamping system was used to force all 109 subassemblies together into a rigid array to prevent subassembly bowing effects and associated reactivity changes during power changes.

The core was nested inside a Type 304 stainless steel vessel designed for steady-state coolant temperature of 700-800°F and a transient capability of 1000°F. Molten sodium, at a temperature of 700°F and a maximum flowrate of 4300 gpm, was driven by a linear induction electromagnetic pump down and around the outside of a shroud into a lower plenum, upward through the core, into an upper plenum, and out through an outlet pipe to a sodium-to-sodium intermediate heat exchanger. A second electromagnetic pump was used to circulate 550°F secondary sodium through the heat exchanger and deliver sodium at 670°F to an air blast heat exchanger for heat dissipation. A parallel electromagnetically pumped system with a capacity of 250 gpm was used as an auxiliary system for removing up to 1 MWt of decay heat in the event of primary pump failure. The heat rejection system was designed for the nominal removal of 20 MW of thermal power.

SEFOR was controlled through reflector action: ten 6-in.-thick nickel segments (rods) completely surrounded the core at a position outside the primary vessel. This arrangement guaranteed reactivity control in the event of core damage during transients. Each segment or rod was 49 in. long and weighed approximately 1200 lb. Two of the 10 rods were used for fine control; the remaining eight were used for coarse control. All were hydraulically driven.

Transient initiation and control was exercised through the use of the FRED (fast reactivity excursion device). The FRED consisted of a rod with a calibrated poison slug. Transients were initiated by ejecting the rod under gas pressure from the core in approximately 100 ms. The rod was then stopped and reinjected back into the core in approximately 1 sec.

A reactor scram, delayed 400 ms with respect to FRED ejection, completed a typical transient test. Slug worths were variable, ranging from 50¢ to as much as 1.28$. Additional details are given in Section 7.4 below.

An interesting and unique feature of the SEFOR plant was a refueling cell, a shielded enclosure 13.5 ft wide, 17 ft long, and 28 ft high, located above the reactor. High purity argon which filled the cell permitted sodium-contaminated components to be withdrawn from the primary vessel and examined through windows in the shielding walls.

Fuel handling was the primary function carried out in the refueling cell. Fuel rods were moved individually to and from the reactor, a storage tank, and work stations within the cell. Operations performed at work stations included visual examinations, dimensional inspections, gamma-scanning, leak checks, disassembly, and canning operations. In the absence of fuel material in the cell, personnel equipped with self-contained breathing apparatus could enter the cell for maintenance and other duties.

The entire facility was located inside a steel dome-shaped containment building 50 ft in diameter and 114.5 ft in height. The facility was situated near the center of a 620-acre site 18 miles southwest of Fayetteville, Arkansas.

7.4 Operation

The initial loading of Core I fuel began on April 6, 1969. Four weeks later, on May 3, 1969, initial criticality was achieved. Additional rods worth approximately $1.00 were loaded to permit the calibration of control rods. Material worth measurements followed. Fuel worth was measured relative to sodium, steel, and B_4C poison. As a result of these measurements it was shown that some fuel rods contained more plutonium than others. This unexpected phenomenon was eventually

traced to a faulty fuel batch during fuel fabrication. The faulty rods were subsequently replaced and the loading was adjusted to complete the full core outline. Of the 648 positions available 634 were occupied by fuel rods and 14 with B_4C poison rods. A series of tests including radiation surveys, temperature, pressure and flow coefficients, and oscillator measurements was then made at power levels ranging from 0 to 1 MWt. Of particular importance was a natural convection test conducted with the reactor operating at 1 MWt and all pumps and blowers shut off to simulate a complete loss of plant power. The results proved conclusively the existence of adequate convective flow.

During the ascent to power extensive steady-state and oscillator measurements were performed. The design power level of 20 MWt was reached on January 29, 1971. The plant was shut down, the FRED (fast reactivity excursion device) was installed, and the system was prepared for the fast transient portion of the experimental program. Subprompt-critical transients using slug worths of 80 to 94¢ were conducted with initial power levels ranging from 1 to 10 MWt. Cover gas analyses and rod inspections were periodically performed to verify the integrity of the fuel.

The first superprompt-critical transient was performed on August 14, 1971, using a slug worth of $1.15 at an initial power level of 2 MWt. Seven additional transients were then conducted. Of these the maximum was based on a slug worth of $1.28 from an initial power level of 8 MWt. No evidence of fuel damage was noted. This completed the measurements planned for Core I.

Core I was converted to Core II by exchanging the BeO-filled tightener rod in each subassembly for stainless steel counterparts. The removal of BeO effectively hardened the neutron spectrum and permitted comparative measurements of the Doppler temperature coefficient under differing neutron spectra.

Tests made at zero power and during the ascent to full power at 20 MWt closely followed those for Core I. Six superprompt-critical transients were conducted. Conditions ranged from a slug worth of $1.12 at 2 MWt to $1.21 at 8 MWt. After completion of the transient tests, steady-state and oscillator studies were resumed for power levels up to 20 MWt. The final test consisted of a 90 MW-day fuel irradiation. Upon completion of this the system was shut down to begin decommissioning activities.

The most important tests conducted in SEFOR were, of course, measurements of the Doppler temperature coefficient. A value determined from the Core II transient ammounted to a T dk/dT of -0.0060 ± 0.008, approximately the same as the calculated value of -0.0063. The corresponding value for Core I tests was approximately 20% higher. The results clearly demonstrated the inherent capability of the prompt negative Doppler effect to limit the energy release during rapid transients in sodium-cooled ceramic-fueled fast reactors.

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