Conf-820465--2

CONF-820465--2

DE83 007671

SEVENTEEN YEARS OF LMFBR EXPERIENCE: EXPERIMENTAL BREEDER REACTOR II (EBR-II)

By

W. H. Perry

Manager of Plant Operations

G. L. Lentz

Assistant Manager of Plant Operations

W. J. Richardson

Operations Staff Specialist

G. C. Wolz

Operations Staff Specialist

EBR-II Project

Argonne National Laboratory
P. O. Box 2528
Idaho Falls, Idaho

Submitted for Presentation
at the
American Power Conference
Chicago, Illinois
April 26-28, 1982

I. INTRODUCTION

Experimental Breeder Reactor II (EBR-II) is a small but complete liquid metal fast breeder reactor (LMFBR) power plant operated by Argonne National Laboratory for the United States Department of Energy at the Idaho National Engineering Laboratory. It has performed safely and reliably for 17 years. Designed and constructed with the technologies of the 1950's, EBR-II continues to serve as an important facility for the national LMFBR program. Much has been learned from operating EBR-II to facilitate the design, licensing, and operation of large commercial LMFBR power plants.

EBR-II is continuing to perform better than had been originally envisioned and is compiling up a growing record of achievements. The plant is easy to operate and maintain and produces very low levels of radioactivity release to the environment and very low dose levels to plant personnel. EBR-II has sustained a respectable plant capacity factor during the last six years (73.7%), and it has achieved high fuel burnup and low rates of driver-fuel failure (one failure per 5000 elements) with the standard EBR-II metallic fuel pins.

During EBR-II's operating history, a number of plant transients, some planned and others not, have been experienced. None has caused significant damage to either the balance-of-plant or the reactor core itself. To a great extent, this record is attributable to the safety features inherent in the design of the primary system. These safety features are based on the large thermal capacity of the primary sodium pool and the capability for natural circulation of the sodium through the core.

II. PLANT DESCRIPTION

EBR-II comprises a sodium-cooled fast reactor with a designed thermal power output of 62.5 MW, an intermediate closed loop of secondary sodium coolant, and a steam plant that produces 20 MW of electrical power through a conventional turbine-generator.[1,2] A neighboring Hot Fuel Examination Facility (HFEF) is used to examine fuel elements, reconstitute previously irradiated experiments for further irradiation in the reactor, and examine irradiated experiments. Part of the HFEF was originally the Fuel Cycle Facility (FCF), which was designed to reprocess spent fuel from EBR-II.

The main parts of the EBR-II plant are the reactor, sodium boiler, and power plant. The reactor building is a steel containment vessel that houses the reactor and primary system. EBR-II uses the pool-type concept in which the reactor, major primary-system components and piping, and much of the fuel-handling equipment are submerged in a large, double-walled tank called the primary tank. Sodium is circulated in a single pass by two main primary pumps through the reactor, through a single outlet pipe to the intermediate heat exchanger, and back to the bulk sodium. An auxiliary electromagnetic pump in the outlet pipe operates continuously to aid in the transition to natural convection circulation for the removal of decay heat if both main primary pumps become inoperative. In the extremely unlikely event of loss of all forced circulation of sodium, decay heat would still be removed either by natural circulation of both the primary and secondary sodium and subsequent rejection to the steam system or by direct rejection to the atmosphere through two passive sodium-potassium (NaK) shutdown coolers in the primary tank.

The secondary system and the sodium-to-water steam-generating equipment are in the sodium boiler building. Nonradioactive secondary sodium is pumped to this building, where it passes through two superheaters and seven evaporators and goes to the surge tank.

The power plant is a conventional steam plant which operates at 1330 psig with about 235°F of superheat. A "zero-solids" water chemistry program is used in the steam generating system. Hydrazine is used to remove traces of oxygen and morpholine to adjust pH in the range 8.8 to 9.2. The treatment for the condenser cooling water was recently changed from a chromate-based to a phosphate-based treatment.

The reactor was designed with 12 fueled control rods. Four of the control rods positions have been converted into instrumented subassembly facilities or other in-core test facilities. Any one of the remaining eight control rods can be used for reactor control; all control rods are used for scram. Two fueled safety rods provide additional removable reactivity during reactor operation and also provide shutdown reactivity during fuel handling. The control and safety rods are similar to standard driver subassemblies but contain only two-thirds the number of fuel elements. The standard driver subassemblies contain uranium-fissium metallic fuel that is sodium-bonded to a stainless steel jacket.

III. SAFETY OF THE EBR-II DESIGN

One of the foremost concerns in the operation of nuclear power plants is the removal of decay heat from the core immediately after plant shutdown. The design of the primary and secondary systems and certain favorable characteristics of sodium allowed EBR-II to be built with a number of inherent safety features that address this concern. These features include a large negative power coefficient of reactivity, strong natural convective flow in the primary and secondary systems, passive safety systems (NaK shutdown coolers) for backup removal of decay heat, and a large volume of primary sodium that provides an enormous heat sink.

Decay heat is normally removed by forced-circulation coolant flow through the core with either the main primary pumps or the auxiliary pump and subsequent transfer of the heat to the power plant by natural-circulation flow in the secondary (intermediate) sodium loop.

In the worst-case situation of no pumping power available and no power plant available for dissipation of the decay heat, natural circulation through the core provides sufficient heat removal. This heat is transferred from the primary coolant to the two passive NaK shutdown coolers, from which it is exhausted to the atmosphere. Hence, loss of the decay-heat-dissipation capability provided by the secondary sodium system and the power plant system is not a serious problem. An economic benefit also results from this situation: neither the secondary sodium system nor the power plant need be a safety-grade system.

Some of the more significant upsets that can occur at EBR-II without damage are:[3]

(1) Loss of Flow (LOF) with Reactor Trip. Loss-of-flow tests have been conducted to measure the adequacy of natural convection for

core cooling. The results of these tests confirmed the existence of convective flow at rates high enough to cool the core immediately after a reactor trip upon complete loss of forced flow.[4]

(2) LOF without Reactor Trip. The results of the above tests strongly suggest that EBR-II could also undergo a complete loss of forced flow without a reactor trip and still avoid sodium boiling in the core or failure of the primary-coolant boundary. As the primary sodium heats up on loss-of-forced flow, the reactor would shut itself down from reactivity feedback. Decay heat would subsequently be rejected by natural convective flow.

(3) Loss of Heat Sink without Reactor Trip. Loss of cooling in the secondary sodium or steam systems without remedial action would not have serious consequences or cause fuel damage at EBR-II. As the primary sodium heats up, the reactor would shut itself down through reactivity feedback. The temperature increase in the primary sodium would be easily accommodated without damage by the primary-system structures, and the heat would be dissipated either by the passive-shutdown cooler system or by parasitic heat loss.

The inherent safety features of EBR-II thus make it immune to the consequences of component failure compounded by operator error, such as occurred at TMI-2.

IV. THE CHANGING ROLE OF THE EBR-II FACILITY

The purpose of the design and early operation of EBR-II was to demonstrate the feasibility of an LMFBR operated as a power plant with an integral fuel-reprocessing facility.[5] By March 1965, the original mission of EBR-II had been accomplished: EBR-II had operated safely and successfully as a power plant while using fuel reprocessed in the integral FCF, and thermal and dynamic behavior of the core proved to be very much as predicted.

In May 1965, EBR-II began its role as a steady-state irradiation test facility with the insertion of the first experimental subassembly. Shortly after this time, an ambitious program was launched to irradiate and test a large number of fuels and structural materials.

A significant portion of the irradiation program has been endurance testing of reactor fuels to identify the fluence limits on fuels and to help develop the burnup capability of fuels. This emphasis on endurance testing gradually progressed from a program of running fuels to cladding breach (RTCB) to a program of running fuels beyond cladding breach (RBCB).

In 1975, an extensive program was started to upgrade the EBR-II facility to permit operation of the reactor for extended periods with breached fuel. Before then, extended operation with breached fuel would have led to unacceptable radiation levels in the reactor building because of a high leak rate of the primary-tank cover gas. The major changes made were (1) installation and successful operation of a cover-gas cleanup system (CGCS) that uses a cryogenic distillation process, and (2) reduction of the leak rate of the cover gas to the reactor building. With the completion of these changes, the RBCB program was started in June 1977. A cesium trap was installed in the primary purification system in March 1978 and had (through March 1982) removed an estimated 345 Ci (12.8 TBq) of cesium from the primary coolant.

As the Fast Flux Test Facility is assuming its role as the LMFBR program's steady-state irradiation facility, EBR-II is gradually concluding its steady-state program. The EBR-II Project is now qualifying the reactor as a test-bed facility for performing a more severe test program. This program is a reflection of the post-TMI emphasis on the milder but more credible nuclear accidents and probable upset conditions that might occur in a commercial LMFBR. The program will look at the effects of reactivity transient rates of up to 10¢/s. This range was selected to fill the testing void between steady-state irradiations and the lower end of the capabilities of Argonne's Transient Reactor Test (TREAT) Facility. Over the next five years, this comprehensive program, known as the operational reliability testing (ORT) program, will use EBR-II in the following areas of investigation:

(1) Testing breached fuel elements under normal operating conditions -- the RBCB program.

(2) Testing the response of present and advanced fuel-element designs to low-ramp-rate ($\leq$ 10¢/s) transient-overpower and duty-cycle events -- the transient-fuel-testing program.

(3) Testing breached or distorted fuel elements and bundles under off-normal transient conditions, and doing associated work in detecting and locating local faults -- the local-fault-testing (LFT) program.

(4) Natural-convective-flow testing of the whole plant -- the shutdown-heat-removal-testing (SHRT) program.

(5) Developing and testing improved methods of reactor control and diagnosis -- man-machine interactions (MMI).

The RBCB tests are under way. The second of the transient tests is scheduled for mid-1982.[6]

V. OPERATING RECORD

A. General

Table I provides a brief overview of the significant events in the operation of EBR-II. No major or minor nuclear incidents have been experienced and no plant shutdown due to equipment failure has exceeded four months. During the last 10 years, the longest unscheduled outage was 8-1/2 days. Most of the equipment failures occurred during the early years and were due to design deficiencies. The failed components were successfully repaired or modified.

Table II summarizes the capacity factor for EBR-II since 1970. The factor increased considerably between 1973 and 1976. This increase is attributed to two factors. The first was a serious effort to upgrade the reactor shutdown system by removing numerous anticipatory trips which were causing many spuriously initiated shutdowns. Few automatic reactor trips have recently been experienced: none in 1980, five in 1981, none so far in 1982. The second factor was the modification of EBR-II's very conservative shutdown criteria for breached fuels. The gradual relaxation of the shutdown criteria was a natural result of the experience gained in the irradiation program which was increasingly emphasizing the irradiation of fuels to and beyond cladding breach.

The average annual capacity factor since 1976 has been 73.7%. Although this is not as high as would be desired for commercial application, it is, nevertheless, an excellent record for a research reactor. It is also indicative of the reliability and maintainability that can be achieved in a pool-type sodium-cooled fast reactor.

TABLE I

EBR-II OPERATIONAL MILESTONES

July 1955	Original authorization to design EBR-II (Public Law 84-141)
November 1956	Award of architect-engineer contract; start of construction design
November 1957	Award of first major construction contract for reactor containment building
September 1961	Reactor made "dry critical" (without sodium); critical mass 230.16 kg of ^{235}U
February 1963	Sodium filling of primary and secondary systems completed
November 1963	Reactor made "wet critical" (with sodium); critical mass 181.2 kg of ^{235}U
July 1964 to March 1965	Stepwise approach to power of 45 MWt
May 1965	Began operation as steady-state irradiation facility
September 1970	Stepwise power increase to 62.5 MWt completed for enhancement of irradiation program
June 1972	Operation with new radial stainless steel reflector around core
June 1977	Run-beyond-cladding-breach (RBCB) program initiated
June 1978	Received American Nuclear Society Special Award for continuing contributions to the nation's FBR program
December 1980	Designated as a cogeneration facility
February 1981	Conducted whole-core transients to qualify EBR-II driver fuel for the ORT program

TABLE II

EBR-II Plant Capacity Factor

Year	Plant Capacity Factor, %
1970	57.9
1971	39.1
1972	46.9
1973	49.9
1974	58.7
1975	66.1
1976	76.9
1977	71.5
1978	72.8
1979	71.1
1980	77.1
1981	73.0

Plant capacity factor (%) is defined as [MWt-hours produced/(calendar-hours x 62.5 MWt)] 100.

B. Maintenance Experience and Operational Problems

During the operating lifetime of EBR-II, a substantial amount of experience and information has been gained in the areas of sodium-system and component maintenance; equipment, instrumentation, and component reliability in a sodium environment; and personnel exposure and radiation/contamination control associated with a sodium-cooled fast reactor.[7]

The major problems encountered in removing components from either the core or the primary tank are contamination from fission products (from the RTCB- and RBCB-program subassemblies), radiation from

^{24}Na and ^{22}Na isotopes, and the high chemical activity of the residual sodium on the components. In addition, components removed from the core region of the primary tank (i.e., from inside the neutron shielding) are highly radioactive because of neutron activation.

Despite the contamination from ^{24}Na, ^{22}Na and various fission products, mainly ^{137}Cs, the annual average exposure to maintenance personnel has decreased from slightly over 1 rem (10 mSv) in 1975 to 172 mrem (1.72 mSv) in 1981. This relatively low average exposure is achieved because: (1) the rapid decay (15-h half life) of ^{24}Na, which is the principal activation product in the primary sodium; and (2) the development and use of a cesium trap to remove the principal fission product, ^{137}Cs. The cesium trap is part of the sodium purification system and can be used during reactor operation. The ^{22}Na activity is only a minor contributor to personnel exposure, because it has a low equilibrium activity.

The high chemical reactivity of sodium with oxygen is controlled by keeping sodium-contaminated components in an inert atmosphere while removing them from the primary tank. Inerted pulling fixtures ("caissons" or "pipes") are used for this purpose. If the component is being removed from the core region, the lower portion of the pulling fixture is shielded with lead. The removed components can be transferred to a sodium cleanup facility for alcohol washing to remove residual sodium before maintenance. Highly activated core components are normally not cleaned or repaired; they are allowed to oxidize slowly and are then discarded. New components are installed.

The performance of components of the primary system and the fuel-handling system has been very good.[8] No failures that were not readily repaired or replaced have occurred in either systems.

Buildup of sodium and sodium oxide in the cover-gas spaces of the primary pumps and the fuel-handling components has caused most of the problems with primary-tank and fuel-handling components. The sodium oxide and sodium builds up in the clearances and causes binding. Corrective action generally consists of cleaning the component, enlarging clearances where feasible, and improving sodium drainage. Mechanical failures in the control-rod drives have been another major problem in the primary tank.[9] These failures occurred in the gripper jaws three times and in the sealing bellows 11 times.

The seals of the rotating shield plugs have been a continual source of difficulty since the system became operational.[10,11] The rotating shield plugs are the mounting platform for the control-rod drive mechanisms and the fuel-handling mechanisms. The plugs seal the primary tank by a dip ring (or blade) and seal-trough arrangement. The trough is filled with a tin-bismuth alloy of low melting point. When the reactor is operating, the surface of the alloy is kept frozen. During fuel handling, the alloy is melted to allow the shield plugs to rotate. Although this arrangement does work, the shield plugs are sometimes difficult to rotate. The difficulty is caused by oxidation of the tin-bismuth alloy in the trough and accumulation of sodium and seal material in the annulus area between the large-rotating-plug support structure and the plug itself. To maintain the rotational capability of the shield plugs, access ports for cleaning have been provided. The sodium and seal-alloy accumulations in the annulus area must be removed about every two years, and the tin-bismuth alloy oxide must be removed from the seal trough about every three months.

The components of the secondary sodium system and the power plant have been essentially trouble free except for some early design and construction deficiencies. Two problems in these areas deserve further comment, however.

In 1974, a slight reduction occurred in the outlet steam temperature of one of the two superheaters. The two EBR-II superheaters are identical in all aspects except for the type of bonding used during fabrication of the duplex tubing. In one superheater, the two tubes of the duplex are metallurgically bonded to each other. The superheater that had the anomalous behavior had mechanically bonded tubes. Since this was the first sign of abnormal behavior in any of EBR-II's eight evaporators and two superheaters, the superheater was removed for destructive examination. An evaporator with metallurgically bonded tubes was removed from the system in 1980, converted to a superheater, and installed in place of the removed superheater in 1981. Subsequent destructive examinations of the duplex tubes in the degraded superheater showed that the mechanically bonded duplex tubes had separated.

In 1980, multiple tube failures began occurring in a feedwater heater, which was original equipment. The tubes in this heater were Monel 400, which is about 68% Ni, 29% Cu, and 1-2% Fe and Mn. The heater was replaced in 1981. Destructive examination of the feedwater heater showed that exfoliation was the primary cause of the tube failures.

VI. CONCLUSIONS

Operating experience at EBR-II over the past 17 years has shown that a sodium-cooled pool-type reactor can be safely and efficiently operated and maintained. The reactor has performed predictably and benignly during normal operation and during both unplanned and planned plant upsets.

The duplex-tube evaporators and superheaters have never experienced a sodium/water leak, and the rest of the steam-generating system has operated without incident. There has been no noticeable degradation of the heat transfer efficiency of the evaporators and superheaters, except for the one superheater replaced in 1981. There has been no need to perform any chemical cleaning of steam-system components.

Operation of EBR-II has produced a wealth of information. As an irradiation facility, EBR-II has generated specific information on the behavior of oxide, carbide, and metallic fuels. As an LMFBR power plant, EBR-II has produced general information related to plant-systems and equipment design, plant safety, plant availability, and plant maintenance.

REFERENCES

1. Koch, L. J., al., "Hazard Summary Report, Experimental Breeder Reactor II (EBR-II)," ANL-5719, Argonne National Laboratory, May 1957.

2. Koch, L. J., Loewenstein, W. B., and Monson, H. O., "Addendum to Hazard Summary Report, Experimental Breeder Reactor II (EBR-II)," ANL-5719 (Addendum), Argonne National Laboratory, June 1962.

3. Sackett, J. I., et al., "Design Features to Maximize Simplicity, Operability, and Inherent Safety of LMFBR's," Trans. Am. Nucl. Soc. 38, p. 615, June 1981.

4. Singer, R. M., et al., "Studies Related to Emergency Decay Heat Removal in EBR-II," Proc. Int. Meeting on Fast Reactor Technology, Seattle, Aug. 19-23, Vol. III pp. 1590-1598, 1979.

5. Perry, W. H., et al., "EBR-II: Summary of Operating Experience through July 1981," ANL/EBR-117, Argonne National Laboratory (in publication).

6. Seidel, B. R., and Allen, A. E., "Operational Reliability Testing at EBR-II," Nucl. Eng. Int. 26 (318), pp. 42-44, September 1981.

7. Leman, J. D., et al., "EBR-II - Ten Years Later," Mechanical Engineering, 99 (1), pp. 33-39, January 1977.

8. Perry, W. H., et al., "EBR-II - Sixteen Years of Successful Operation with a Pool-type Reactor," Trans. Am. Nucl. Soc. 38, p. 606, June 1981.

9. Cerutti, B. C. and Waldo, J. B., "Ten Years of Operating Experience with Mechanical Components of the EBR-II Fuel Handling System (1963-1972)," ANL-7836, Argonne National Laboratory, December 1973.

10. Cooper, L. P., et al., "Operating Experience with the EBR-II Rotating Plug Freeze Seals," ANL-7617, Argonne National Laboratory, December 1969.

11. Clark, A. F., et al., "Operating Experience with the EBR-II Rotating Plug Freeze Seals," 1969-1975, ANL-75-10, Argonne National Laboratory, March 1976.