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SAFETY AND OPERATING EXPERIENCE AT EBR-II: LESSONS FOR THE FUTURE*

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ABSTRACT

EBR-II is a small LMFBR power plant that has performed safely and reliably for 16 years. Much has been learned from operating it to facilitate the design, licensing, and operation of large commercial LMFBR power plants in the US. EBR-II has been found relatively easy to keep in conformity with evolving safety requirements, largely because of inherent safety features of the plant. Such features reduce dependence on active safety systems to protect against accidents. EBR-II has experienced a number of plant-transient incidents, some planned, others inadvertent; none has resulted in any significant plant damage. The operating experience with EBR-II has led to the formulation of an Operational Reliability Test Program (ORTP), aimed at showing inherently safe performance of fuel and plant systems. One lesson resulting from the ORTP work is that a large plant should be designed to emphasize inherent safety features for public protection.

INTRODUCTION

The Experimental Breeder Reactor II (EBR-II) is a small but complete LMFBR power plant operated by Argonne National Laboratory at the Idaho Nuclear Engineering Laboratory. The power at which it is operated depends upon the particular experimental program for which it is being used, but it has a steady-state capability of 62.5 MWt and a corresponding gross electrical output of 20 MWe. The original mission of EBR-II was to demonstrate the performance of an LMFBR power plant having an integral fuel reprocessing and fabrication facility. It did so, first producing electrical power in 1964. In 1967 it began to be used as an irradiation facility for candidate LMFBR fuels and materials, and has continued in this role to the present. EBR-II has performed safely and reliably for over 16 years—it achieved a plant capacity factor of 77.2% in FY 1980 despite its involvement in experimental programs.

A schematic representation of the EBR-II plant is shown in Fig. 1, and a detailed description of it is given in Ref. (1). EBR-II uses sodium as the working fluid in its primary and secondary heat transport systems. The reactor mixed-mean outlet temperature at full power and flowrate is 473 C, about 420 C below the boiling point of the sodium. The primary system thus operates at a peak pressure of only about 345 kPa and is not susceptible to depressurization "flashing" effects such as took place in the TMI-2 plant. In addition, the primary sodium coolant reacts chemically with halogen fission products released from breached fuel elements, greatly inhibiting the loss of such radioactive species to the environment.

The primary system of EBR-II has a pool-type configuration, that is, the entire primary system and the intermediate heat exchanger (IHX) are submersed in a

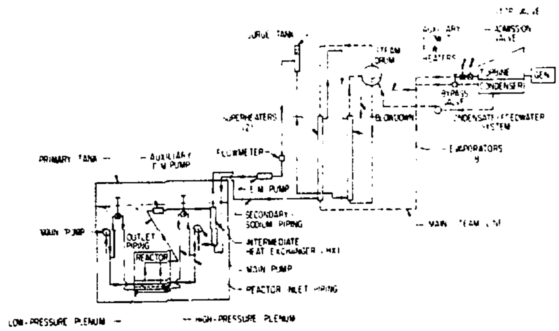

Fig. 1 Simplified Schematic of EBR-II Plant

333 m^3 tank of sodium. The two primary pumps take their input directly from the pool, at 371 C during normal operation. This large reservoir of sodium acts to mitigate the consequences of upset conditions in the plant, both in terms of thermal stress effects and of response requirements for the plant control and plant protection systems. The pool configuration has thus eased both the design and subsequent safe operation of the plant.

Much has been learned from operating EBR-II—both as a power plant and experimental facility—to aid the design, licensing, and operation of large commercial LMFBR power plants in the US. This paper discusses safety aspects of the operating experience at EBR-II and the lessons learned thus far from that experience. Increasingly, this learning process is focusing on the importance of using <u>inherent</u> safety-related characteristics of LMFBR power plants to assure public safety and also to enhance protection of the plant itself.

EVOLUTION OF EBR-II SAFETY POSTURE

With the evolution of the mission of EBR-II since it began operation in 1964, there has been a corresponding need to expand the operating capability of the reactor. Expanding that capability has required many changes to the plant and considerable analysis and testing to support them. This effort has also contributed to improved understanding of fast-reactor-safety issues.

For example, the plant-protection system (PPS) at EBR-II has been modified extensively to improve its performance (2,3,4). A summary of major changes is given in Table 1. The result is a cleaner and more efficient PPS, contributing to improving plant capacity factor and more effective protection. That experience is directly applicable to design of protection systems for larger LMFBR's—it is being more sharply focused to relate to major issues of PPS design, such as usefulness of "reactivity meters" (5) and the philosophy of design to ensure maximum reliability.

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TABLE 1. Major Changes to the EBR-II PPS

Trip Function	Action
Low Flow	Deleted several unnecessary trips associated with pump and MG-set conditions (winding temperature, clutch voltage, low current, etc.). Removed trips on high rate-of-change of flow, and added a trip on loss of power to the pumps. Upgraded low-flow trip circuitry, adding a trip to one flowmeter and deleting trips from two others.
Reactivity Insertion	Added a bypass for trips on short period after reaching approximately 1/2 power. Upgraded nuclear instrumentation to provided wide-range channels to cover full operating range.
Seismic	Replaced the single seismic detector with three detectors, tripping the reactor and isolating the containment building on either vertical or horizontal motion.
Fuel-element Cladding Breach	Replaced the automatic trip on high delayed neutron signal with administrative limits for shutdown, upgrading the systems diagnostic capability.
Sodium Spill in Containment	Deleted containment building isolation trips on high air temperature and high air pressure when it was shown that such conditions could not be reached. Upgraded isolation trips on high radiation in building.
Operational Interlocks	Deleted several interlocks from the reactor trip circuitry where their function could be served by administrative controls. Examples include control rods latched to their drives and high temperature in nuclear instrument thimbles.
Primary Sodium and Cover Gas	Deleted trips on high reactor inlet temperature, high reactor outlet temperature, high bulk sodium level and high argon cover gas temperature when they were shown to be unnecessary for protection. Upgraded trip circuitry on high subassembly outlet temperature.

The philosophy of safe design may be one of the most important results of this work. The experience gained with plant testing and modeling, particularly natural convective cooling (6,7), has demonstrated the potential for taking advantage of inherently safe characteristics of LMFBR's to minimize reliance on active safety systems for protection (8,9).

The result of this work has been distilled into the EBR-II Technical Specifications, formally approved in December 1976 as the first such safety document for an LMFBR (10). This document is continually being upgraded to reflect modifications to the reactor and improved understanding of its performance. As a result of this work, EBR-II is increasingly seen as a simple and safe reactor design, providing important information for large plant design.

OPERATING EXPERIENCE

During the 16 years that EBR-II has operated, a wide variety of problems have been encountered and solved. These problems have included binding of the primary pumps, a loose pipe in the IHX, a number of difficulties with the fuel handling system, binding of the rotating plugs in the primary tank deck, and handling of distorted subassemblies (11). Removal and replacement of components in the primary tank has been done routinely and without major difficulties. Techniques for detection and location of breached fuel elements, as well as for cleanup of the primary sodium and argon cover gas to remove fission products coming from such breaches, have been developed and implemented (12, 13,14).

During its operating lifetime the EBR-II plant has experienced many upset events or transients, most inadvertent, but some planned for test purposes. The planned transients were conducted as part of a natural convective test program mentioned earlier. These included three tests where the primary pumps were tripped from operation at 40% of rated flow and 36% of rated power with the primary auxiliary pump off, causing reactor scram on a low flow signal. In each test the secondary pump was tripped a different length of time after the primary pumps were tripped. The results were used to check out a plant dynamics computer model of EBR-II called NATDEMO (15). An example of an inadvertent transient was a situation where power to pump 1 was lost, followed by manual trip of pump 2 (to protect it), with the auxiliary pump on. The reactor scrammed from 100% power on low flow signal, a classical loss-of-flow event expected to occur in any plant a number of times over its operating life.

Another example was a transient of the overpower type, which occurred when a technician doing maintenance on the control console during power operation accidently leaned on a control rod drive switch. This caused a fuel-containing control rod to move further into the reactor, leading to an increase in power. The reactor scrammed at about 110% power on a high flux level signal. This was a classical transient overpower event which is also anticipated to occur several times in the life of the plant.

None of these events resulted in any significant damage to the plant, not only because of the proper operation of its engineered safety features, but also because of the inherent safety features designed into it. These features include a large negative power coefficient of reactivity, strong natural-convective flow in the primary and secondary systems to enhance coolability on loss of electrical power, and the use of passive safety systems wherever possible, e.g., for backup removal of decay heat.

A recent example of the benefit of relying primarily on inherent characteristics to provide protection was temporary failure of <u>both</u> normal and emergency ac power systems at EBR-II on March 2, 1981, traced to failure of a breaker in the emergency power switchgear. The reactor was at power, and scrammed automatically. The reactor systems were not endangered because, if necessary, the core could be cooled, and heat rejected from the primary system, using natural convective cooling only. In fact, had the operators walked away from the plant, with no electrical power available, reactor safety would not have been endangered.

Although the emergency power systems at EBR-II are not considered class 1-E (safety-grade), they are designed to be highly reliable with considerable redundancy and diversity. There are two emergency ac power buses, each backed up by an emergency diesel. The original failure of the emergency breaker resulted in a large current which tripped not only the breaker feeding the affected emergency power bus, but also the breakers for incoming normal power. The second emergency bus, having lost its normal electrical feed, could not be energized immediately because its backup diesel was being run as part of required weekly tests. It was connected manually to its emergency bus and emergency electrical power restored shortly after the incident.

It is to make reactors invulnerable to such failure of active systems that emphasis is given to designs that maximize inherent safety features, either extending the time available for protective action or eliminating the need. This desire has led to development of the Operational Reliability Test Program (ORTP) at EBR-II.

THE OPERATIONAL RELIABILITY TEST PROGRAM

Operating EBR-II both as an irradiation facility and an LMFBR power plant has produced a wealth of information, some relating specifically to fuels performance, and the rest more generally to plant systems behavior. An integrated systems approach to testing and operation began during the preparation of EBR-II for irradiating fuel elements having breached cladding (the Run-Beyond-Cladding-Breach or RBCB Program) in 1976. This preparation required questions to be addressed in three strongly interrelated areas: fuels performance, plant safety, and plant availability. At about the same time, tests were underway to measure natural convective flow conditions in EBR-II, using highly instrumented driver fuel subassemblies.

The Operational Reliability Test Program (ORTP) evolved directly from the original RBCB and natural convective work. Questions concerning the behavior of breached fuel elements under mild upset conditions led to the formulation of the Transient Experiment Test (TET) Program and under more severe upset conditions to the Local Fault Test (LFT) Program. The two programs require operation of EBR-II in a number of transient power modes. The natural convective work led to the Shutdown Heat Removal Test (SHRT) Program. Two new instrumented subassemblies are being fabricated and additional balance-of-plant instrumentation is being installed in support of this work.

Some work has also been started in the area of the man-machine interface (MMI). Impetus for this work came from safety review of EBR-II following the TMI-2 accident, from the potential use of EBR-II for testing of sensor validation techniques, and from the importance of such work as a "binder" in a systems approach to the design, licensing, and operation of a commercial LMFBR power plant. A major goal of the MMI work is to lay the basis for design of control systems which can be demonstrated to be highly reliable, contributing to the overall safety posture of the plant.

A considerable effort is underway to prepare EBR-II for the ORTP. Included is qualification of the driver fuel for transient operation, thermal-hydraulics and stress analyses of plant components, preparation of supporting safety documents, and the design and fabrication of special facilities. Potential failure modes of the driver fuel under the planned duty cycles have been identified, and experiments, both out-of-pile and in EBR-II are being carried out to assure fuel reliability. Results thus far are quite promising. This includes a series of 56 EBR-II transients just completed, in which the power is raised from 23.9 MWt to 62.5 MWt as fast as can be done with the current standard control rod drive mechanism, held at full power for 12 minutes, and then reduced rapidly to 23.9 MWt. No driver fuel breaches have been observed.

The thermal-hydraulics and stress analyses are being done in two stages. The first stage, already completed, was a series of macroscopic analyses to identify regions of potentially high stresses for more detailed study. The second stage, to be completed in September 1981, is finite-element analysis of potential high-stress regions. Theoretical worst-case duty cycles are being used for conservatism in this work, which is based upon meeting the requirements of the ASME Design Code Section III and Elevated Temperature Code Case 1592. It has been found thus far that:

1. creep damage of EBR-II components is small;
2. the limiting stress region in the plant is the junction between the upper elliptical head and the tubesheet of the IHX, in which about 3000 cycles of the most severe type (overpower transients terminated by scram) are allowed; and
3. other plant components can sustain at least 10,000 cycles of the most severe type.

The allowable number of fatigue cycles is based upon a new plant; thus damage caused by previous operation must be estimated in order to calculate the allowable additional number of cycles. EBR-II has experienced about 318 scrams from 20% power or greater to date. Assuming each such scram produced the same fatigue damage to the IHX as that resulting from the most severe transient used in the analyses (3×10^{-4}/cycle), the cumulative damage to the IHX to date was estimated at 0.1. That is, operation to date has resulted only in about a 10% reduction in the life of the IHX.

The safety documentation to support transient operation of EBR-II for the ORTP will be based upon the driver fuel qualification work, the stress analyses of key components, and reactor and plant dynamics analyses. Limits on reactor operation and trip settings for the plant protection system will be determined from the dynamics analyses.

Special facilities for the ORTP include the Breached Fuel Test Facility (BFTF), installed in EBR-II in 1979; the Fuel Performance Test Facility (FPTF), to be installed in 1981, and the Cluster Test Facility (CTF), just in the planning stage. The BFTF is a caisson-like arrangement that fits over the top of an RBCB test subassembly in the fifth row of the EBR-II core (see Fig. 2). It contains flow and temperature measurement instruments, and more important, a delayed-neutron (DN) detector to measure the progress of a breach in the test subassembly. The FPTF is similar to the BFTF, except that it also contains a capability for programmable variation of flowrate through the test subassembly. The CTF is a row five facility that will permit concurrent monitoring of the DN release characteristics of four surrounding subassemblies using one DN detector and a flow-switching arrangement. The value of the CTF is that it will allow more RBCB tests to be run in EBR-II than could be done with the BFTF or FPTF.

One of the two new subassemblies under fabrication for the SHRT Program is a special driver fuel subassembly (XX09), containing 59 driver fuel elements. It has thermocouples at its inlet and outlet, as well as in the element spacer wires at various elevations, plus two flowmeters. The other subassembly (XX10) is a mock-up of a blanket subassembly, containing 19 stainless steel rods. It also has two flowmeters, inlet and outlet thermocouples, and spacer-wire thermocouples. These two subassemblies, both to go into row five at the same time, have very different neutronic and thermal-hydraulic characteristics. They will thus provide detailed measurements of temperature distributions and flowrates in subassemblies with different characteristics under steady-state and transient conditions. This information is needed for modeling whole-reactor dy-

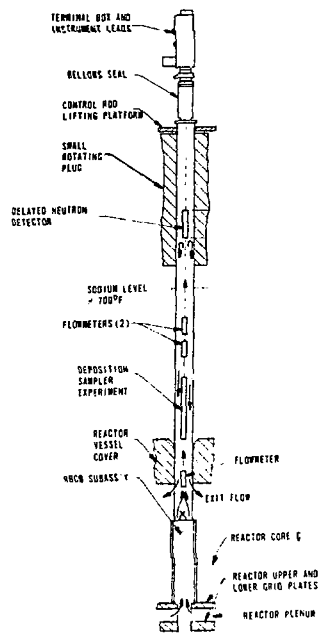

Fig. 2 Schematic of Breached Fuel Test Facility (BFTF)

namic thermal-hydraulic behavior. Further SHRT testing is scheduled to begin in 1983.

CONCLUSIONS--LESSONS FOR THE FUTURE

Our first conclusion follows directly from the work required to prepare EBR-II for the RBCB Program. This work included developing safety arguments to permit removal of the DN detector from the reactor scram circuit, setting limits on DN signal for manual shutdown of the reactor, and addition of a special cleanup system to remove noble-gas fission products from the argon cover gas and of a special trap to remove radioactive cesium from the primary sodium. We found that fuel performance impacted plant safety and plant operation to such an extent that all three areas must be treated using an integrated systems approach. Involved here are strongly interrelated requirements for: (a) the strategy for detection and location of breached fuel elements, (b) limits of safe operation with such elements under normal, anticipated, and unlikely upset conditions, (c) the design of a rapid fuel handling system, (d) purification of the primary sodium and cover gas, and (e) maintenance of components in the primary system.

Our first conclusion is, then, that only with an integrated systems approach, one aspect of which is illustrated here, can a commercial LMFBR power plant be designed, constructed, licensed, and operated to sustain a high plant factor over its required lifetime.

Our second conclusion derives directly from our operating experience with EBR-II to date and from formulating the ORTP and preparing the plant to carry it out. A pertinent example of the operating experience was the recent temporary failure of both the normal and emergency ac power systems at EBR-II, a benign event largely because of inherent safety features of the plant. The ORTP-related planning has required us to develop a very detailed understanding of the dynamic behavior of the whole EBR-II plant under a wide range of both intentional and inadvertent transient conditions. This has permitted us to take full advantage of the many inherent safety features that were originally designed into the EBR-II plant.

Our experience has been that the addition of extra safety systems often has the effect of increasing design and operating complexity. In contrast, the thrust of the original PPS upgrading effort at EBR-II was to reduce the complexity of safety systems when it had been confirmed that inherent features of the plant provide protection. The result was an improvement of plant operating efficiency as well as improvements in safety. As occurred with the recent failure of the emergency power system, it is primarily the failure of active engineered systems that challenge the reactor and operating crews.

Our second conclusion, then, is that the plant protection system, containing active (engineered) systems, should be designed to protect the plant itself, but that to the extent possible passive (inherent) systems should be used to assure public safety. Inherent systems may well also provide plant protection.

An LMFBR power plant should be designed to be "walk-away safe," that is, to accommodate all credible accident scenarios without release of radioactive material to the environment and without requiring operator intervention. EBR-II would likely do so because of inherent safety features, including a strongly negative power coefficient of reactivity and two diverse systems for removing decay heat from the reactor, one of which relies only on natural convection.

The strong negative power coefficient of EBR-II is due to the small size of its core, in which heatup of the sodium coolant enhances neutron leakage, a negative reactivity effect. In a large LMFBR, where leakage is not a dominant effect, another inherent means of inserting negative reactivity would have to be used. Such a means has been proposed by both the French (16, 17) and British (18), and is supported by independent analysis (19). The proposed approach is to utilize thermal expansion of control rod drive shafts heated by sodium leaving the reactor to further insert the control rods into the reactor under loss of primary flow conditions. With careful attention to design, this approach might well eliminate coolant boiling under loss-of-flow conditions with failure to scram, thereby precluding a power excursion driven by positive void reactivity. Demonstration of this capability would considerably enhance the licensability of the plant.

A major requirement for "walk-away safety" is that long term decay heat removal not require electrical power. It is also desirable that the decay heat removal system be independent of the intermediate sodium and the steam system, since failure of a steam generator could affect both.

As noted earlier, the primary coolant system of EBR-II has a pool-type configuration, that is the primary system is completely immersed in a large tank of molten sodium. This configuration facilitates decay-heat removal by natural-convective heat transport systems. It also damps the thermal response of the primary system to upset conditions originating in the balance of plant as well as in the primary system itself. Finally, it provides sufficient room for location of a dedicated decay-heat removal system in the primary tank. Our third conclusion is thus that an LMFBR power plant having a pool-type primary system configuration is particularly well suited for incorporation of inherent safety features.

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