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OPERATING AND TEST EXPERIENCE WITH EBR-II, THE IFR PROTOTYPE*

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ABSTRACT

The Experimental Breeder Reactor number 2 (EBR-II) has operated for 30 years, the longest for any liquid metal cooled reactor (LMR) power plant in the world. Given the scope of what has been developed and demonstrated over those years, it is arguably the most successful test reactor operation ever. Tests have been carried out on virtually every fast reactor fuel type. The reactor itself has been extensively studied. The most dramatic safety tests, conducted on 3 April, 1986, showed that an LMR with metallic fuel could safely accommodate loss of flow or loss of heat-sink without scram. EBR-II operated as the Integral Fast Reactor (IFR) prototype, demonstrating important innovations in safety, plant design, fuel design and actinide recycle. The ability to accommodate anticipated transients without scram *passively* resulted in significant simplification of the reactor plant, primarily through less reliance on emergency power and not having to require the secondary sodium or steam systems to be safety grade. These features have been quantified in a probabilistic risk assessment (PRA) conducted for EBR-II, demonstrating considerable safety advantages over other reactor concepts.

Fundamental to the superior safety and operating characteristics of this reactor is the metallic U-Pu-Zr alloy fuel. Performance of the fuel has been fully proven: achieved burnup levels exceed 20 at.% in the lead test assemblies. A complete set of fuel performance and safety limits has been developed and was carried forward in formal safety documents supporting conversion of the core to IFR fuel. The last major demonstration planned was to assess the performance of recycled actinides in the fuel and to confirm that passive safety characteristics are maintained with recycled actinide fuel in the core.

KEYWORDS

EBR-II; electrorefining; fast reactor; Fuel Cycle Facility; IFR; liquid-metal cooled; liquid-metal reactor; LMR; metallic fuel; nuclear fuel cycle; passive safety; plutonium; reactor design; reactor safety; recycling.

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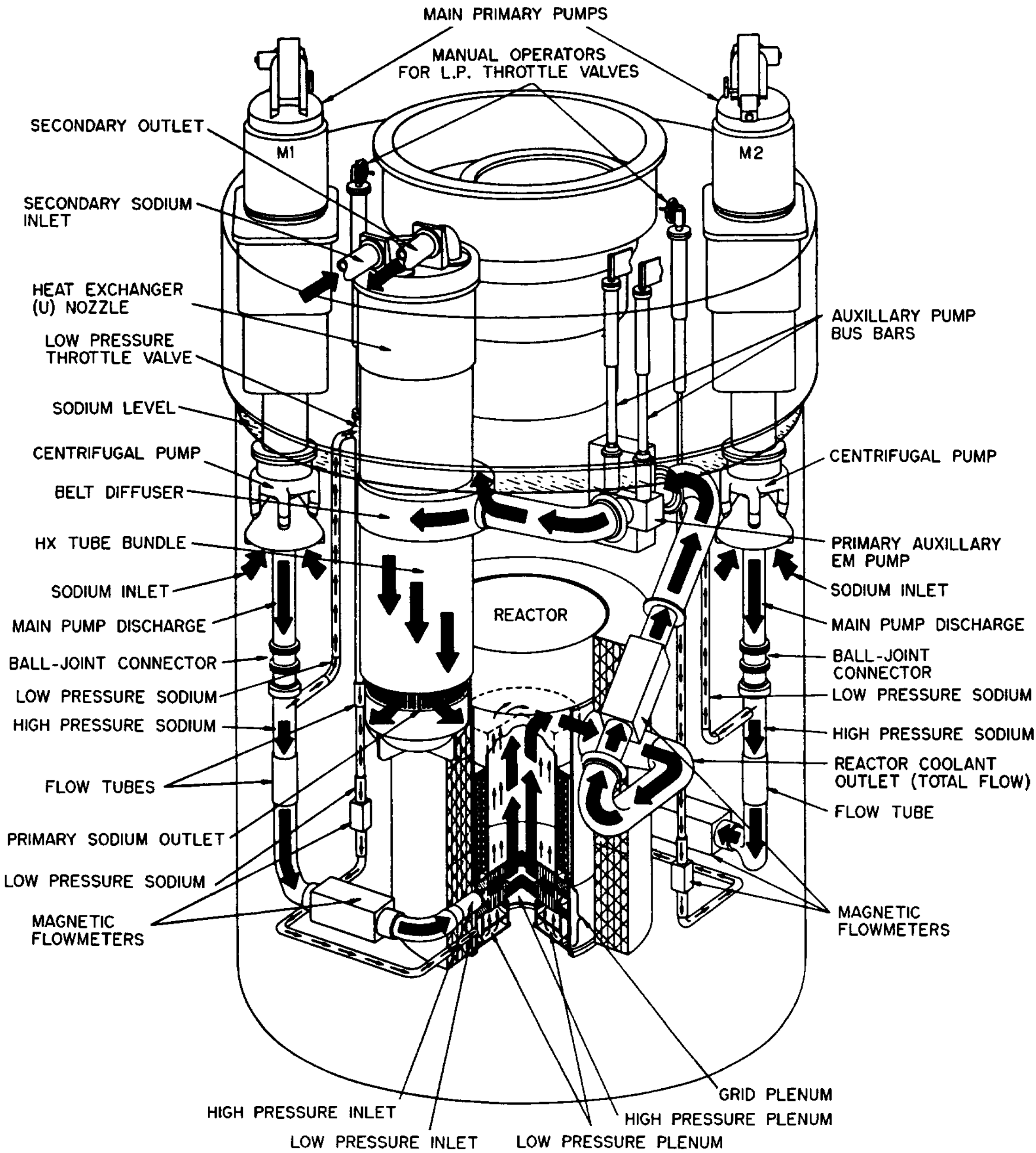

Fig. 1. Schematic of the Primary System.

DESCRIPTION OF THE PLANT

EBR-II is a U.S. Department of Energy (DOE) facility located at the Idaho National Engineering Laboratory (INEL) and operated by Argonne National Laboratory (ANL). EBR-II began power operation in 1964 and has supported a variety of programs and initiatives since that time, while maintaining a record of highly reliable operation (Sackett *et al.*, 1990; Lambert *et al.*, 1990a). Plant capacity factors typically exceed 70% and have approached 80%, even though the reactor supports an extensive testing program, while at the same time producing electricity as a complete liquid metal cooled reactor (LMR) power plant (Lehto *et al.*, 1990). The following key characteristics, common to LMRs, make reliable operation feasible: Low-pressure sodium coolant, limited thermal stress, limited corrosion of components, and simplicity of layout in both the primary and secondary sodium systems. These same features mean that there are no intrinsic factors that limit the potential plant life of an LMR.

There were two primary objectives for the original EBR-II facility: To establish the feasibility of fast reactors as power plants and to demonstrate simple nonaqueous techniques for the on-site recycling of spent fuel. In the early stages of planning, EBR-II was regarded as an evolutionary step in the ultimate development of commercial-size fast reactors. EBR-II and the Fuel Cycle Facility (FCF) were conceived in 1953. Civil construction began in 1958 and was completed in 1961. The reactor was first operated at power in August 1964.

Remarkably, all of the basic design choices for EBR-II were correct. There was concern about the ability to keep sodium from leaking from piping, and it was decided to contain the reactor's primary cooling system in a large tank, so that leakage would not matter (Fig. 1). The piping, in fact, is designed with the expectation that sodium will leak from the connections to the primary pumps and past the control-rod drives in the reactor cover without being lost to the system. The secondary sodium system, which removes heat from the primary sodium, contains no valves in the main loop, to lessen the probability of leakage (Fig. 2).

Much attention was paid to preventing a sodium-water reaction at the interface of the secondary sodium system with the steam system. The heat exchangers have a unique design, with one tube contained within another. This duplex-tube arrangement ensures that if one tube develops a crack, any leakage will be stopped by the other (Fig. 3). This has worked very well over the years. EBR-II is one of the very few LMR plants that has not had leaks in the steam generators.

Other design choices that have proven to be exceptionally favorable include the metal fuel, an arrangement of the core and the heat exchanger that facilitates reactor cooling on loss of power, the ability to store spent fuel in the tank while the reactor is operating, and the use of passive devices to remove decay heat from the core. Notably, the hot sodium from the reactor core is piped directly to the intermediate heat exchanger (IHX), ensuring a nearly isothermal cold pool of primary sodium and limited thermal stress on the primary tank. Essential features that give EBR-II its outstanding safety characteristics are:

- Low-pressure, noncorrosive coolant;
- very high thermal inertia; excellent heat transfer;
- low stored energy;
- compatibility of fuel and coolant;
- simplified feedbacks.

These features have all been retained in developing the reactor component of the Integral Fast Reactor (IFR) system (Wade and Chang, 1988). These safety characteristics complement the safety attributes that result from the IFR metal fuel (see Chapters 2 and 4).

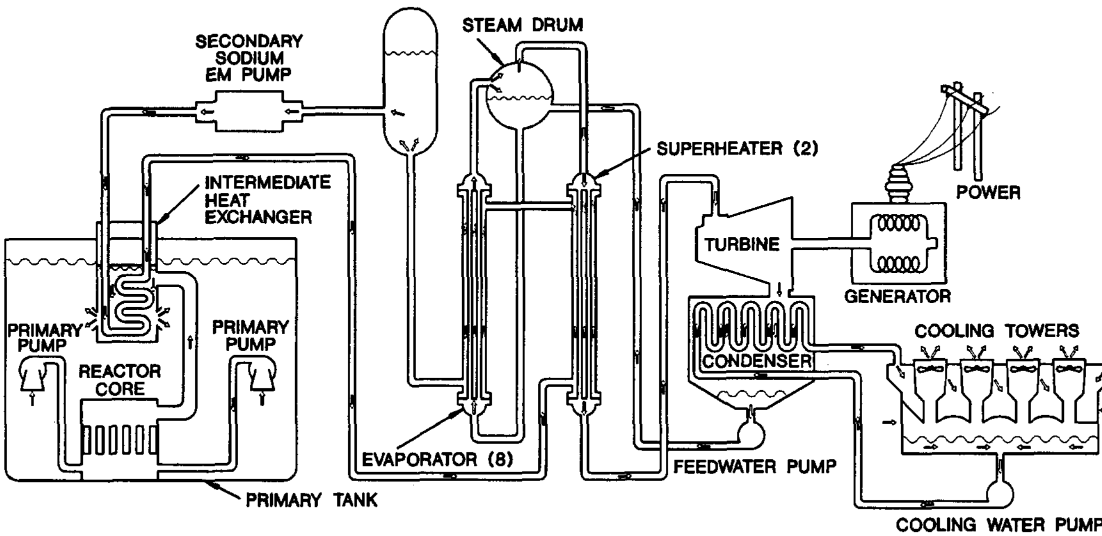

Fig. 2. The Three Systems: Primary, Secondary, and Steam.

The advantage of a low pressure coolant system is apparent: there is no possibility of depressurization and the consequent "blowdown" loss of cooling. LWRs require complex systems to compensate for this eventuality.

Sodium does not corrode the metals used in the LMR reactor structures and components, so that radioactive corrosion products are not formed in any significant amount. Water systems require extensive erosion/corrosion monitoring, and radioactive corrosion products circulating and depositing around the system make access for maintenance difficult. For the LMR, access for maintenance is simplified by the fact that radiation exposures to plant personnel are very low. After more than 27 years of operation, EBR-II is experiencing less than 0.1 man-Sv/a personnel exposure, whereas LWRs now average 2 to 3 man-Sv/a. Other LMRs around the world have comparable exposure records. A noncorrosive coolant also implies reliable performance of components and improved plant availability.

Submerging all the primary systems—the core, primary pumps, intermediate heat exchangers and primary piping—in a pool of molten sodium makes very long times available for remedial action in the event of a loss of heat removal or an overpower transient. Typical designs have a heat capacity in the range of 1°C per full-power second. With the very large margin to boiling of the coolant and appropriately designed thermal expansion, such a core can comfortably ride through even extreme transients. The 1986 EBR-II demonstrations of unprotected loss of cooling and loss of heat sink have been widely reported (Planchon *et al.*, 1987); these simulations of extreme accidents that involve rapid power transients were remarkable for being unremarkable.

The very high heat-transfer rates of both fuel and coolant lead to very low stored energy. The stored thermal energy in fuel is typically less than that required to boil sodium in the associated fuel channel. Thus, typical loss-of-cooling calculations show that there will be no coolant boiling if *either* the reactor shutdown (scram) system functions *or* the flow is not blocked, so that it can coast down at a near-normal rate.

Removal of decay heat is also greatly simplified. Sodium cooling provides efficient natural circulation. The General Electric Power Reactor Innovative Small Module (PRISM) concept (see Chapter 3) uses natural circulation of air for cooling the vessel for ultimate heat removal (in case none of the normal heat transfer systems are operating); EBR-II uses dipstick coolers. In either case, totally passive cooling is available with ample margin for degraded situations. Decay heat removal by external cooling of the reactor vessel under emergency conditions has been shown to be practicable for reactor sizes up to 300 MWe (see Chapter 3) and could be extended by increasing the vessel heat transfer coefficient. Dipstick cooling could in principle be used for any size reactor. The design constraints on natural circulation are discussed in Chapter 2.

EBR-II has been analyzed to determine the factors that limit its usable technical lifetime. Critical components have been identified, as have other issues that influence the rate of aging; surveillance programs have been developed for monitoring the performance of various components and for detecting performance trends that could indicate incipient problems; analysis of the potential for extending the lifetime of a plant such as EBR-II has been integrated into the technical support activities. The only limiting factor, beyond easily replaceable components such as instruments, is radiation damage to structures in the core. By applying straightforward principles, an LMR can be designed to have an extremely long service life.

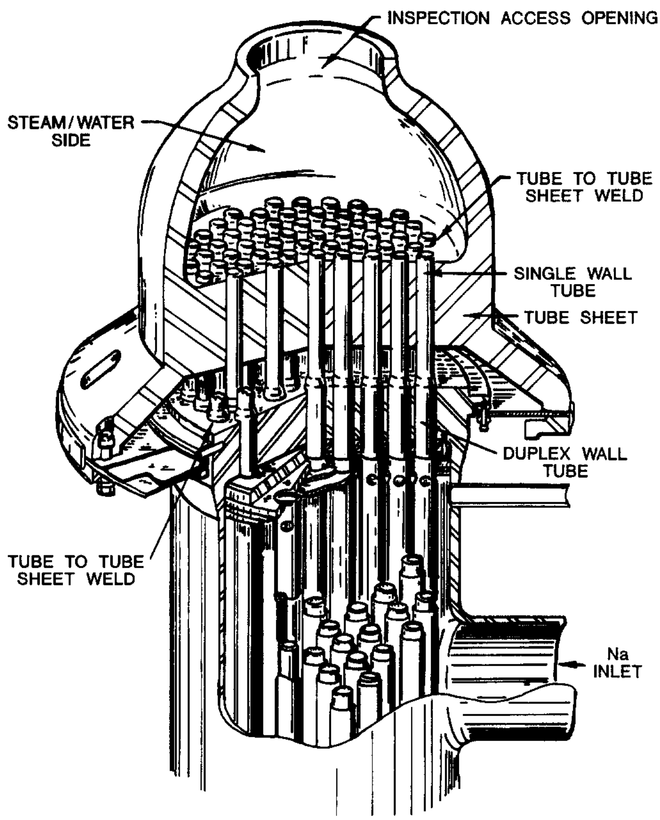

Fig. 3. Steam Generator.

EVOLUTION OF TESTING PROGRAMS

Testing programs to support LMR development and safety research have evolved in EBR-II from the initial mission of demonstrating feasibility of a complete LMR power plant with on-site fuel processing, to its mission as the Integral Fast Reactor (IFR) prototype with its advanced metallic fuel (U-Pu-Zr) and recycling of actinides by electrorefining (Wade and Chang, 1987). The initial demonstration of the fuel processing took place from 1964 to 1968, at which time EBR-II was modified to include a stainless-steel (SS) radial reflector to enhance irradiation capabilities for testing oxide fuels and developing materials. In the second phase of testing, the fuel for both the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor (CRBR) plant was developed. It gave the U.S. program, and EBR-II as part of it, the opportunity to comprehensively test the performance of a wide variety of fuels, including metal, oxide, carbide, and nitride (Lambert *et al.*, 1990a; Washburn *et al.*, 1979). It was from this experience that the superior performance of metal fuel under high irradiation was established. The burnup limit for EBR-II's driver fuel was increased considerably over those years as designs were improved. (Metal IFR fuel has now been taken to greater than 20% burnup.) The major benefits, however, come from the very favorable safety characteristics of the fuel as well as the much greater flexibility in the physics of metal-fueled cores (see Chapter 4).

Many of these advantages were demonstrated in the third phase of the EBR-II testing program, which began in the late 1970s and expanded the testing of conventional steady-state fuels to include investigation of the response of metal and oxide fuel, and of the plant itself, to off-normal conditions such as breached cladding and operational transients (see Chapter 5). Plant transients, such as loss of flow, loss of heat sink, and transition to natural convective flow were addressed. During this test phase, plant upgrades provided special facilities for in-core measurement of coolant flow and temperature, fission-gas handling systems for identifying breached fuel, and computer-controlled power shaping for transient tests. The most dramatic part of this work was a series of tests conducted in 1985 and 1986 that culminated in a demonstration of the ability to accommodate loss of flow and loss of heat sink without scram (Planchon *et al.*, 1987) (Figs. 4 and 5). Just as important has been testing more directly related to operation, laying the basis for simplified control and showing where automation can help assure operational safety.

The IFR is the product of this experience. The concept includes an uranium-plutonium-zirconium (U-Pu-Zr) metallic fuel, a pool-type LMR, core and plant designs to enhance passive safety, and an integral pyrometallurgical fuel-processing facility with remote-controlled injection casting of recycled metal fuel (Fig. 6) (Burris *et al.*, 1984). The work with EBR-II as the IFR prototype to develop and demonstrate the IFR technology had four major objectives:

- To demonstrate the satisfactory performance of recycled U-Pu-Zr fuel, with effective and essentially complete fissioning of recycled actinides,
- to demonstrate the inherent safety features of the IFR and to set the precedent for safety documentation appropriate to an IFR,
- to demonstrate the operational advantages of an IFR, such as benign response to controller malfunctions and the benefits of sodium for operation and maintenance,
- to demonstrate that operation of an IFR in compliance with regulations is facilitated by characteristics inherent in the design.

Demonstration of the integral fuel cycle will involve transferring spent fuel to the FCF for partially removing fission products, fabricating new assemblies by remote control, and returning them to the reactor (Fig. 7). This demonstration is facilitated by the fact that the reactor and fuel cycle facility are adjacent to each other, connected by a tunnel for transporting the fuel-handling cask.

Except for the electrorefiner, a very similar fuel cycle was demonstrated in EBR-II during the first five years of its operation (Stevenson *et al.*, 1987). Over 700 irradiated assemblies of all types were processed. Of these, 560 were fuel-bearing assemblies that were processed to separate the fuel from most of the fission products. This operation produced 34,500 reprocessed fuel pins, which were fabricated remotely into 418 assemblies and returned to the reactor.

The passive safety features of the IFR supplant the active and complex safety systems used in light water cooled reactors. This simplification promises substantial savings in maintenance and operations, and less need for preparation and maintenance of safety documentation. The format of the EBR-II safety documentation was revised to better support operation as the IFR prototype and to reflect the advantages of the IFR in areas of safety, health, and the environment. Also, operational safety features that would result in cost savings have been investigated. Innovative approaches are possible for such issues as number of operations personnel, training requirements for emergency response, automation of control and procedural functions, and reduced maintenance requirements due to fewer safety-grade systems and components. For example, EBR-II has no requirement for emergency electrical power to protect the core. Safety-grade components are limited to the primary system, and even there, no pumps or valves are required to be safety grade.

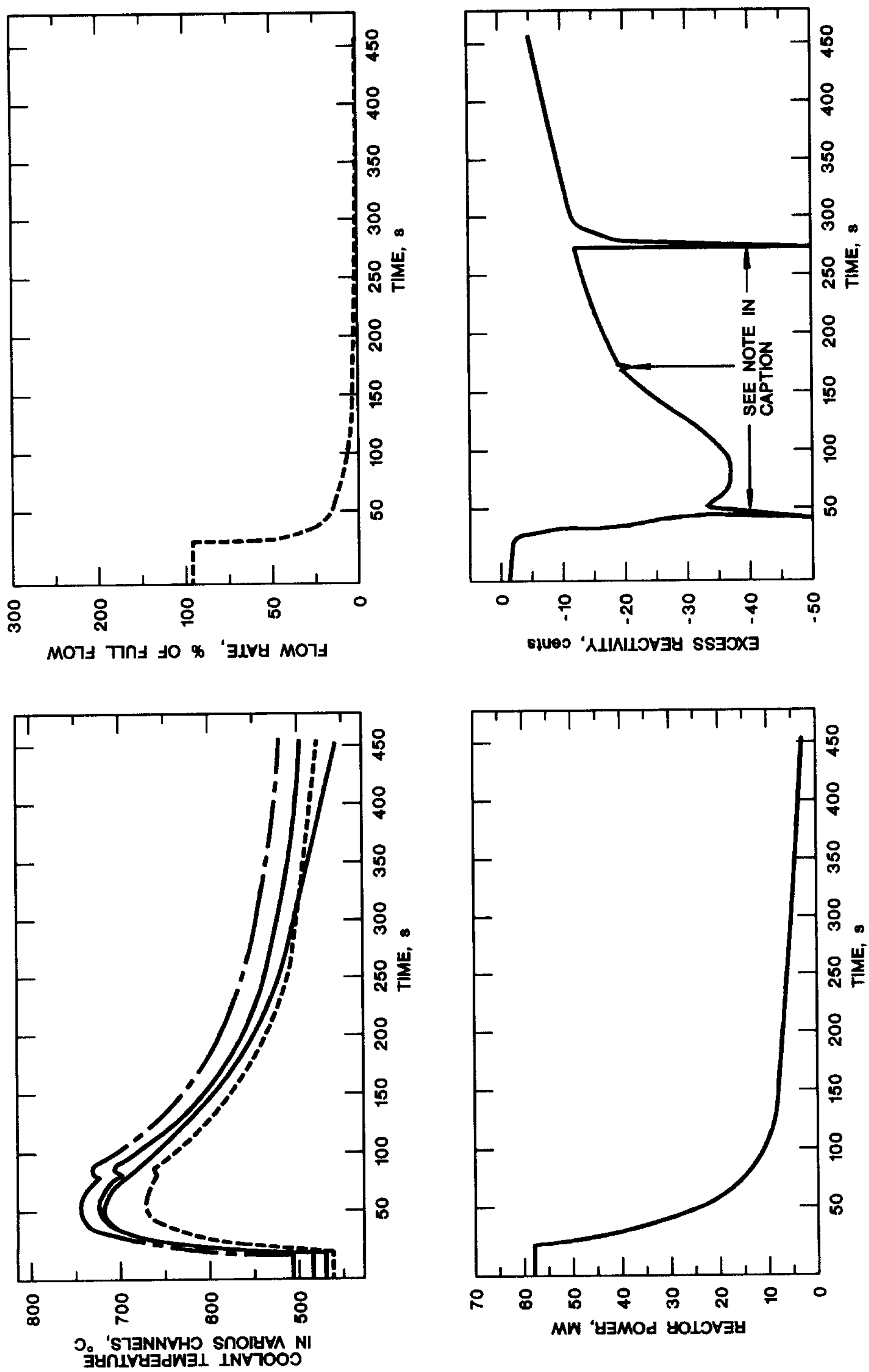

Fig. 4. Response of Plant to Loss of Flow (LOF).
Note: The indicated discontinuities in the reactivity plot are artificial, the result of range changes.

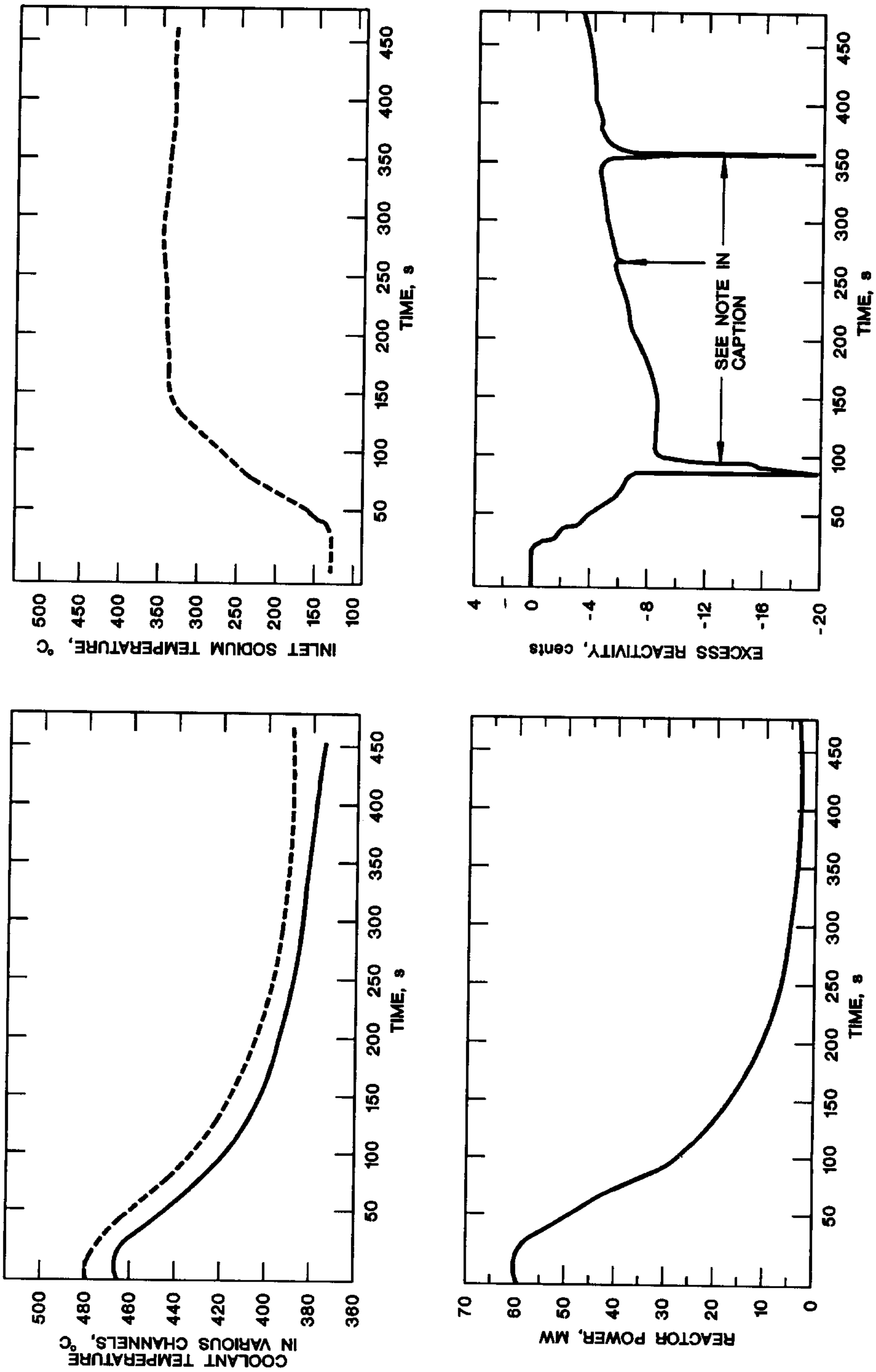

Fig. 5. Response of Plant to Loss of Heat Sink (LOHS).
Note: The indicated discontinuities in the reactivity plot are artificial, the result of range changes.

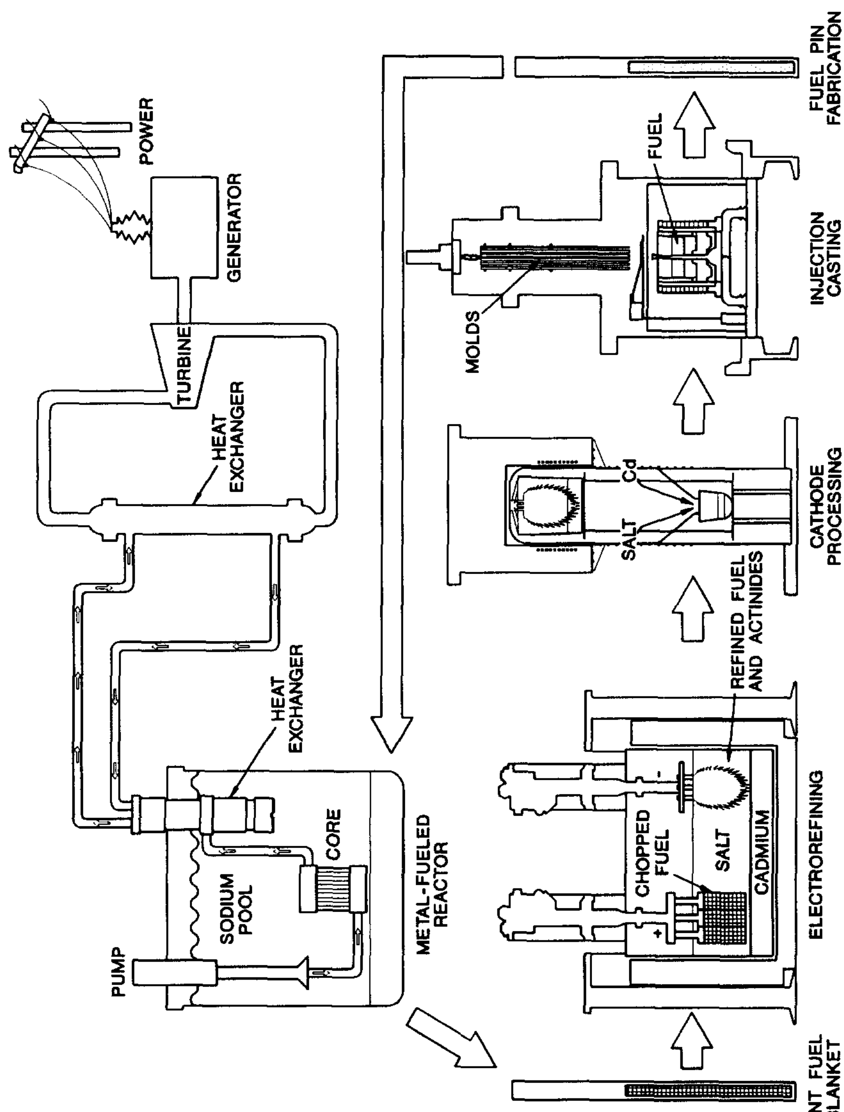

Fig. 6. The IFR Concept.

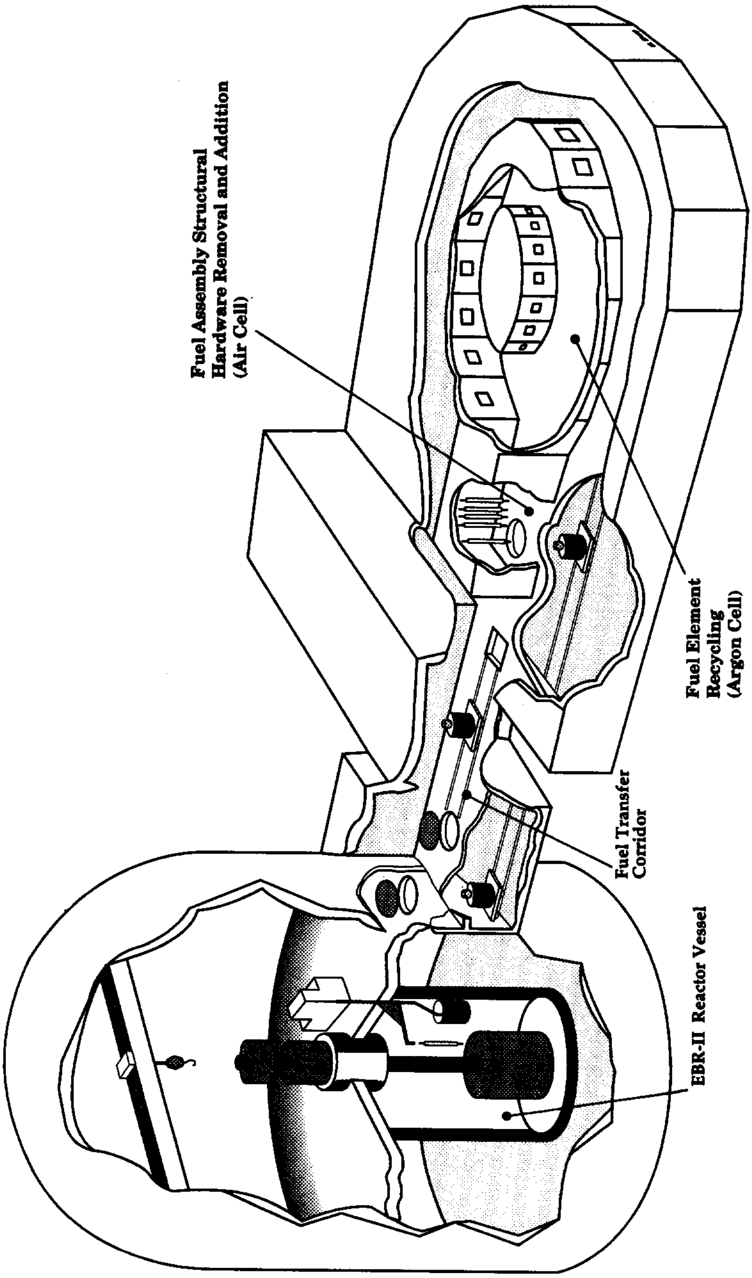

Fig. 7. Schematic of EBR-II and the Fuel Cycle Facility (FCF).

ROLE OF OTHER TEST FACILITIES AT ANL-WEST

EBR-II's success builds on the lessons learned from its predecessor, the Experimental Breeder Reactor number 1 (EBR-I) (Haroldsen *et al.*, 1963). EBR-I was first operated in August 1951, and on 20 December 1951 it produced electric power, the first ever from a nuclear power plant. It demonstrated that a fast reactor could be operated reliably, showing the simplicity of working with liquid metal coolant and the practicality of metal fuels in fast reactor systems.

The Transient Reactor Test Facility (TREAT) (ANL, 1987; 1992) provides the capability for studying the response of fast reactor fuel to bounding overpower transients. The TREAT reactor is an air-cooled, thermal, heterogeneous reactor used to simulate postulated reactor transients and transient undercooling events. Its primary mission is to conduct safety-related tests in support of the IFR technology development program. Those tests include overpower transient tests to determine dynamic behavior of fuel during reactor excursions, overpower transient tests to investigate fuel-coolant interaction phenomena, steady-state tests at power with loss of flow to investigate coolant interaction phenomena, and combination loss-of-flow and transient-overpower tests. TREAT also provides neutron radiography services for experimental fuel irradiation programs and other experiments. Tests have demonstrated that modern IFR fuel has a great deal of margin before failure during overpower events, and that propagative failure is not likely. Further, axial motion of the fuel, as it softens within the cladding, provides a reliable and very effective shutdown mechanism for hypothetical severe overpower transients.

The Zero Power Physics Reactor (ZPPR) (Lawroski *et al.*, 1972) is the culmination of a long line of critical facilities at ANL—facilities that were instrumental in the development of the extensive reactor physics capabilities of the U. S. ZPPR is an experimental critical facility in which different reactor core designs can be mocked up, operated at a low power, and characterized according to the physics properties of the core and surrounding regions. ZPPR provides experimental physics data for the design of fast-reactor demonstration plants and large fast-reactor central-station power plants. Operational and design parameters such as critical mass, control-rod worth, power-generation distribution, breeding-blanket effectiveness, and the effect of neutron flux on support structures are measured for configurations that exactly duplicate the neutronics of the proposed design. Also measured and confirmed are safety-related parameters fundamental to the demonstration of a safe design, such as the Doppler coefficient and the sodium-void coefficient.

Rounding out the important test facilities associated with the IFR program are the Hot Fuel Examination Facility (HFEF) (Bacca, 1980) and the Fuel Cycle Facility (FCF) (Hesson *et al.*, 1963; Stevenson *et al.*, 1965; Bacca *et al.*, 1980; Heubner *et al.*, 1981). They are large hot-cell complexes designed to provide an inert gas atmosphere for examining and processing highly irradiated fuels and materials. The combination of the original FCF and HFEF established much of the current technology of remotely controlled fuel processing, handling, and examination.

PERFORMANCE OF IFR FUEL

Safety and operability issues associated with the behavior of metal fuel under irradiation, including its potential for failure and propagation of faults and its behavior when breached, have been thoroughly examined (see Chapter 5). Metallic fuel has been studied in EBR-II from the time it started up, with the last seven years devoted to development of U-xPu-10Zr fuel (where x = 0 to 26 wt%) for the IFR concept.

The irradiation program, complemented by out-of-reactor tests and analyses, was also directed to furthering understanding of the performance of IFR fuel during off-normal conditions. A key question was performance during steady-state operation following cladding breach. The testing has shown that such operation does not degrade the fuel and has no adverse impact on safety or reliability (see Chapter 5; Lambert *et al.*, 1985 and 1990b; Seidel *et al.*, 1990).

A major concern with metal fuel is the potential for liquid-phase penetration of the cladding at temperatures at or above the eutectic temperature of the alloys of certain components of the fuel and cladding. This temperature varies with fuel and cladding type. Out-of-reactor tests have shown limited penetration at very low rates at the temperature at which the first of the liquid phases appears (Pahl *et al.*, 1990). At higher temperatures, the penetration rate is faster; a test pin (U-19Pu-10Zr) heated to 800°C showed a 26% reduction in the cladding thickness after 1 h. Since that time at temperature is much greater than typical accident sequences, this aspect of the safety of the metal fuel is established.

These observations confirmed the results of earlier EBR-II tests (Liu *et al.*, 1990 and Lahm *et al.*, 1990), in which metal fuel of varying burnup was operated at temperatures up to 800°C in a 61-pin assembly. The assembly operated for ~42 min before failure of a high-burnup pin. The breach occurred where the fuel and cladding interacted, and the breach location and failure mode agreed well with the pretest prediction. The still-intact fuel pins from that assembly were put into another fuel assembly and irradiated to the end of the reactor run, which took them slightly beyond cladding breach. Two breaches occurred, one at 10 at.% and the other at 10.2 at.% burnup. Both occurred at the fuel-restrainer dimple, at the same burnup as previous breaches with this type of fuel. The lower-burnup pins were then reincorporated into still another test, and were irradiated to >11 at.% without breach. Postirradiation examination of the pins showed no significant fuel/cladding interaction, although there was some fuel restructuring due to the high temperature operations. These tests demonstrated the safe and reliable operation of metal fuel following long-term over-temperature operation. The breached-fuel test facility (BFTF) in EBR-II provided an opportunity to conduct a similar test with IFR fuel.

IFR FUEL SAFETY CASE

A "fuel safety case" is a detailed analysis of the safety characteristics and limitations of a proposed new type of fuel. The structure of the analysis for the IFR fuel safety case was patterned after the Nuclear Regulatory Commission (NRC) Standard Review Plan (NRC, 1987). This structure has been adopted to ensure that it would be prototypic of future IFR safety analysis reports. The methods involve both analysis and experiments, including verification of performance through demonstration in EBR-II.

Apart from changes in fuel behavior, there are a number of important neutronic differences from the EBR-II experience base. Because the principal fuel is plutonium, a full IFR core will have a smaller delayed-neutron fraction β=0.0053, as compared to a β of 0.007 for LWRs. This difference changes the kinetics of the reactor slightly.

To address the issues of fuel behavior and reactor transients associated with the transition to a plutonium-based fuel, a full thermal-hydraulic analysis of all the relevant accidents in the duty cycle has been completed (Chang and Hill, 1993; Chang *et al.*, 1994), and the results have been shown to be essentially independent of β, and therefore of the precise core loading.

The reliability of the cladding as a barrier to the release of radioactive materials depends on the inherent strength of the cladding materials relative to the potentially applied loading. The two cladding materials used for IFR fuel pins are ferritic-martensitic alloy HT9 and austenitic alloy Type 316 Stainless Steel. The mechanical behavior of both of these alloys is relatively well understood, and both are routinely used in the FFTF reactor (Leggett and Walters, 1993) as fuel cladding and duct material. Most of the mechanisms leading to cladding rupture are related to plastic strain, which, in a metal-fuel system, is due to increase in gas pressure in the pin, not to fuel/cladding mechanical interaction. As temperature increases in a transient, gas pressure increases while cladding strength is reduced. The margin to failure can be predicted using fairly simple models (see chapter 5).

An important factor in determining cladding strength is loss of cladding thickness due to wastage. Mechanisms that have been identified as contributing to cladding wastage are scratches on the cladding surface, solid-state diffusion of fuel constituents and fission products (especially lanthanides) into the cladding, and eutectic liquefaction at the fuel-cladding interface.

"Eutectic liquefaction" is the term given to the formation of low-melting-point phases at the fuel/cladding interface. These phases are formed by interdiffusion of iron and nickel from the cladding and uranium from the fuel, forming an alloy at the interface. There is a threshold temperature below which there is no liquid formation at the interface; above this temperature the liquefaction of the cladding proceeds at a rate that is temperature dependent. For high burnup U-20Pu-10Zr fuel with HT9 cladding, the threshold temperature has been determined to be 655 $\pm$ 5°C (see Chapter 5). There is evidence that at lower burnups the threshold temperature is much higher. A temperature limit of 650°C has been adopted to limit the effects of fission products and actinides carried over in the recycled fuel. Further in-core testing may allow this limit to be increased.

During off-normal transients, eutectic liquefaction at the fuel-cladding interface may consume part of the original solid-state interdiffusion zone in the cladding, or may propagate into the cladding. From the observed rates of cladding wastage and penetration, one can accurately predict the thickness of the load-bearing portion of the cladding, which is the quality of greatest interest. The total wastage for analyses of fuel pin behavior in transients can be determined by adding the steady-state wastage to the calculated penetration during the transient. In performing these analyses, the total wastage from multiple transients is assumed to be cumulative. This yields a conservative estimate of cladding thickness and strength, and of the probability of cladding breach during normal or off-normal transients. It should be emphasized that, as has been demonstrated in many in-core tests, simple cladding breach does not constitute a safety problem.

The absence of fuel melting has traditionally served as one of the design criteria for nuclear reactor fuels. In the case of metallic fuels it is important to distinguish between melting of the fuel itself and limited eutectic liquefaction at the fuel-cladding interface. Fuel melting occurs when the fuel temperatures exceed the local solidus temperature. The primary reason that fuel melting has traditionally been used as a design criterion is that, at least for oxide fuels, centerline melting can lead to contact of molten fuel with the cladding and rapid cladding failure. Metallic fuels have much lower melting temperatures than oxide fuels (1100°C compared to 2700°C), so cladding failure caused by this mechanism is much less likely. Nevertheless, fuel melting is not allowed for the IFR fuels during either normal operation or off-normal events, eliminating the need to consider cladding failure induced by molten fuel contact. This also assures that molten fuel cannot relocate within the pin so as to increase reactivity or increase fuel/cladding mechanical interaction. A significant advantage of EBR-II as an irradiation test facility is that these conclusions could be verified by in-core testing.

Further discussion of fuel behavior can be found in Chapter 5.

EBR-II PLANT SAFETY TESTS

During the 1980s, a number of plant tests were conducted in EBR-II, which, taken collectively, demonstrate the safe operation of an advanced metal-fueled LMR. Following an early demonstration of natural convective cooling in EBR-II, tests were conducted that led to two demonstrations that the reactor, running at full power, would safely shut itself down without benefit of scram upon loss of forced cooling, and upon loss of heat sink (Planchon *et al.*, 1987). In each case, the reactor was shut down passively by negative feedbacks, and transient and equilibrium temperatures were measured to be below those of concern for fuel integrity and reactor safety.

The first test reproduced what would happen if all electric power were lost to an IFR plant. In a large light water cooled reactor (LWR), many emergency systems would actuate to ensure that the reactor core was cooled. The most important of these systems is the control-rod scram system that shuts the reactor down when actuated. For the EBR-II test, that system was deliberately not actuated. The reactor was brought to full power, special scram protection was added, in case the did test not go as predicted, and all coolant pumps were shut off. No safety systems were actuated, including the scram system, and the reactor was allowed to respond passively to the event. The result was that the reactor shut itself down without intervention by operators or safety systems and without damage of any kind, a remarkable result. Temperatures increased, but the increase in temperature was what caused the reactor to shut down on its own, with no damage at all.

What makes this happen? Metal fuel, because it runs more nearly at the temperature of the coolant, operates with a relatively small Doppler-feedback coefficient of reactivity. Consequently, there is very little *positive* reactivity added when the fuel temperature decreases as power decreases. But the negative reactivity feedback associated with increase in coolant temperature is large, and results in rapid reduction in power (see Chapter 4).

The second of those major tests at EBR-II duplicated another extreme challenge to plant safety—loss of the ability to reject heat from the system. It this were to happen in an LWR, and if the reactor were not shut down by its scram system, the reactor core would seriously overheat. In the test in EBR-II, full reactor power was established, the ability of the system to reject heat from the primary coolant was eliminated, with the reactor trips bypassed. The coolant temperature of the reactor rose less than 50 C°, which was sufficient to shut the reactor down.

These are sensational results. Two of the most severe accidents that can threaten nuclear power systems have been shown to be of no consequence to safety or even operation of EBR-II. The reactor was inherently protected without requiring emergency power, safety systems, or operator intervention. The difference was especially apparent when, about a month later, the Chernobyl accident occurred. Taken together, these were dramatic demonstrations of the safety advantages of the IFR.

Another test involved run-up of the primary coolant pump (Lehto *et al.*, 1988), to show that the IFR will respond benignly to extreme overcooling events. Starting at 32% of normal flow and an initial power-to-flow ratio defined to be 1.0, primary flow was increased to 100% in 20 s. The power rose along with the flow, and leveled off at a power-to-flow ratio of 0.9. Consequently, the temperature of the coolant leaving the core was lower than at the start of the test.

The dynamics of such an event are complex. Increased coolant flow causes the overall core to cool and contract, which causes an increase in reactivity, which increases the power and the fuel temperature, which in turn limits the reduction of core temperature and leads to a new equilibrium power level. In the EBR-II test, that new power level was benign. During the experiment, the secondary flow was conservatively controlled to keep the inlet temperature nearly constant. It was also shown (by analysis) that the power increase would be even less with a control strategy that allowed reactor inlet temperature to increase as a natural consequence of the increase in primary flow. Thus, the transient overpower caused by primary pump runout has been shown by analysis and test to not be a safety problem for an IFR.

Obviously, the results of the passive safety demonstration programs have broad implications for the safety aspects of the design and operation of advanced LMRs. For example, they have shown the importance of having negative reactivity feedback associated with increases in coolant temperature, especially when coupled with a *low* Doppler coefficient of reactivity. The importance of long flow-coastdown times for the primary coolant pumps was demonstrated, as was the need for detailed overall thermal-hydraulic design that enhances natural convective cooling. The demonstrations have also indirectly suggested reactor designs that minimize the reactivity swing caused by burnup. Such designs reduce the reactivity that must be vested in control rods, and mitigate the consequences of rod-induced transient-overpower accidents.

The test results also suggest advanced control strategies in which reactor power is controlled over the load-following range by adjusting either the primary system flow, the secondary system flow, or the turbine steam supply. The feasibility of these control schemes for metal-fueled LMRs has already been demonstrated (Chang *et al.*, 1988).

THE PROBABILISTIC RISK ASSESSMENT FOR EBR-II

A probabilistic risk assessment (PRA) was conducted for EBR-II to quantify the probability of damaging events occurring at EBR-II (Hill *et al.*, 1993). This study is also discussed in Chapter 4. It showed that the expected frequency of core-damaging incidents was very low ($<1.6X10^{-6}$ y^{-1}) and that passive-safety characteristics improve reliability considerably. Not only are failures of active systems easily accommodated, but so are operator errors (either of commission or omission). This would be expected from a reactor system that can accommodate loss of flow or loss of heat sink without scram.

The PRA addressed external hazards as well as those associated with plant operations. Criteria for core damage were very conservatively chosen; damage to the core was assumed whenever the limit in a Technical Specification related to fuel temperature was predicted to be exceeded. Under such a conservative assumption, an incident that only moderately exceeded the criterion would not be catastrophic. Even with this definition (damage at Technical Specification limits), 95% of protected and unprotected internally initiated events led to no "damage." The most extreme event was loss of flow without scram, for which (as mentioned earlier) the response has been shown to be benign (no damage to fuel), and for which the margins can be improved by simple changes in plant design.

Interestingly, reactivity-insertion events have been found to be relatively unimportant. This is primarily because there is relatively little reactivity worth carried in the control rods but there are strong reactivity feedbacks from the core.

The study showed a much lower damage frequency for EBR-II than for any other type of reactor now in operation. Perhaps the most value from such a study, however, is that it can be used as part of a risk-management program related to operations, training, maintenance, and modifications. A number of improvements to safety and operating reliability of systems at EBR-II have been made, and more are planned. For example, it has been determined that there is no safety need for an auxiliary pump for the primary coolant, nor for the cocked safety rods that are used in LWRs during fuel handling to mitigate the possible consequences of inadvertently inserting excess reactivity. This PRA also provides information important for the design of future IFR plants.

POTENTIAL PLANT LIFE

At the time EBR-II was designed, little was known about the effects which would ultimately limit the lifetime of a fast reactor. As an experimental facility, EBR-II was designed with considerable flexibility. As the EBR-II facility was approaching 20 years of operation in the early 1980s, ANL began a process of formally identifying and evaluating aging, reliability, and life-extension issues, and developing plans for assuring that there are no technical features of the plant that would require an early termination of operations (King *et al.*, 1991). A complete engineering and operational assessment of all major and most minor plant systems was performed.

That assessment resulted in a list of critical components, within or directly associated with the reactor and primary systems, that were judged to have the most potential to inhibit the ability of the plant to operate reliably to 30 years; i.e., those components whose failure or malfunction would cause a plant shutdown of three months or longer, or which would reduce plant reliability significantly. The list is in Table 1.

In addition to the engineering and operation assessment, there is need for an ongoing research and development effort that is directly germane to plant-life extension. That work includes studies of fatigue and thermal stress in major components, and investigation of neutron-irradiation damage to reactor materials. All these projects provide the analytical and materials-testing support required to establish a technical foundation for extended plant lifetime.

The cyclic-fatigue study included analyses of thermal stress and creep in the reactor vessel, vessel cover, intermediate heat exchanger (IHX), primary piping, superheaters, and secondary sodium piping. The analyses were performed to ensure that these major components could safely withstand a planned series of transient-overpower tests to investigate the operational safety of oxide and metal fuel. The limiting component was found to be the IHX, but the number of cycles which the IHX would experience in 30 years of life, including transient testing, was found to be less than half of what is allowed by the American Society of Mechanical Engineers' (ASME) code criteria. In addition, the IHX is a replaceable component.

In-reactor tests and studies of neutron radiation damage have been performed since the early days of EBR-II operation. Samples of different reactor materials were installed in the reactor core, blanket, and reflector regions in surveillance assemblies in the mid-1960s, to study the effects of a high-energy neutron environment on structural reactor materials. The materials of most interest were various varieties of Type 304 SS, representing the reactor grid plate, reactor cover, reactor vessel, primary tank, and the reactor vessel neutron shield (graphite sealed in Type 304 SS canisters). The significant age-related concerns for these materials were loss of ductility of the nonreplaceable components, void-induced swelling of the grid plate, and swelling of the graphite shielding that surrounds the reactor vessel wall inside the primary tank.

Of primary concern for extended plant life were the grid plate and the graphite shielding, because in EBR-II they are not replaceable and are subject to neutron damage, being close to the core. Some of the irradiation samples were located in regions of the core that have a higher neutron flux than the grid plate and graphite shield are exposed to. Thus, in 20 years the samples reached a neutron fluence equivalent to the grid-plate fluence at 45 to 50 years of operation. Tensile tests of the Type 304 SS samples have shown that residual ductility remains well above allowable minimums, and void swelling is low. Likewise, tests on the irradiated graphite samples indicate that the graphite is in a densification phase and has not yet started to swell at these fluences. It was also found that there was no stored energy in the graphite.

These assessments, analyses, and tests indicate that the reasonable expected technical lifetime for EBR-II is well beyond 30 years—it would have been 50 years or more before any aging limits were reached.

With that phase completed, the plant-life extension program was refocused to build upon the original results of the plant engineering and operations assessment. The list of critical components continues to be the focal point in allocating resources for surveillance, diagnostics, preventive maintenance, spare parts, and upgrade modifications for improving long-term reliability, the goal being a lifetime of 40 years or more. The analyses of thermal stress and cyclic fatigue in major components have been updated to cover a hypothetical 40 years of

Table 1. Critical Components in EBR-II.

Nonreplaceable or Nonrepairable	Replaceable or Repairable
Safety rod drive system	Control-rod drive system
Reactor grid-plenum assembly	Fuel handling system
Rotating plugs and their seals	- core gripper
Primary system instrumentation	- holddown
Primary tank	- reactor cover-lift mechanism
Reactor vessel neutron shield	- fuel storage basket
	- fuel transfer arm
	Primary sodium pumps
	Intermediate heat exchanger (IHX)
	Reactor building polar crane

operation. They indicate that, with very conservative assumptions, the limiting component (the IHX) will have accumulated only 60% or less of the allowable lifetime cycles. As indicated above, some of the irradiated samples examined already have accumulated enough neutron exposure to establish physical characteristics beyond the 40-year operational life. Other irradiated samples are undergoing examination and testing to add to the database.

In the rest of the plant, the issues associated with aging and life extension are not as critical, because the secondary heat transport and steam systems are accessible, maintainable and replaceable. Both the secondary and steam systems are nonradioactive. The secondary system uses sodium as the heat transport medium, resulting in a simple, low-pressure system with no valves in the main loop and a highly reliable electromagnetic pump with no moving parts. The steam generator (superheater and evaporators) has also proven to be highly reliable.

Furthermore, an EBR-II superheater was removed from service in the late 1970s and replaced with a modified evaporator, after there was a degradation in heat transfer efficiency. The superheater was destructively examined and found to be in excellent condition. The sodium side had little or no evidence of corrosion or erosion, and the steam side exhibited normal effects of exposure to saturated and superheated steam. The heat transfer degradation was determined to be the result of relaxation of the prestress of the mechanically-bonded duplex tubes. The replacement superheater has metallurgically-bonded duplex tubes and has not experienced any degradation.

CONCLUSION

The 30 years of EBR-II operation have established an important new direction for LMR research and have provided a basis for confidence that it will succeed. It is a critical time for LMR development worldwide, and EBR-II as the IFR prototype has provided important answers for today's issues of safety, cost, fuel recycle and waste storage. Demonstration of this technology in the 1995 time frame would have provided the basis for the U.S. to move to the next phase—construction of a demonstration IFR plant—when the time comes.

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