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OPERATING AND TEST EXPERIENCE OF EBR-II\*

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ABSTRACT

EBR-II has operated for 27 years, the longest for any Liquid Metal Reactor (LMR) power plant. During that time, much has been learned about successful LMR operation and design. The basic lesson is that conservatism in design can pay significant dividends in operating reliability. Furthermore, such conservatism need not mean high cost. The EBR-II system emphasizes simplicity, minimizing the number of valves in the heat transport system, for example, and simplifying the primary heat-transport-system layout. Another lesson is that emphasizing reliability of the steam generating system at the sodium-water interface (by using duplex tubes in the case of EBR-II) has been well worth the higher initial costs; no problems with leakage have been encountered in EBR-II's operating history. Locating spent fuel storage in the primary tank and providing for decay heat removal by natural convective flow have also been contributors to EBR-II's success. The ability to accommodate loss of forced cooling or loss of heat sink passively has resulted in benefits for simplification, primarily through less reliance on emergency power and in not requiring the secondary sodium or steam systems to be safety grade. Also, the "piped-pool" arrangement minimizes thermal stress to the primary tank and enhances natural convective flow. These benefits have been realized through a history of operation that has seen EBR-II evolve through four major phases in its test programs, culminating in its present mission as the Integral Fast Reactor (IFR) prototype.

I. PLANT DESCRIPTION

EBR-II is a U.S. Department of Energy 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. Current plans are to operate EBR-II to at least a 40-year lifetime. Achievement of a 40-year lifetime with high reliability is important for the near-term goals of the Integral Fast

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Reactor (IFR) program and advancement to the demonstration/commercialization phase. The following key characteristics common to LMRs make extended-life operation feasible, based primarily on the use of sodium as the primary coolant: low-pressure, limited thermal stress, limited corrosion of components and simplicity of layout in both the primary and secondary sodium systems. A major objective of any plant design should be to take maximum advantage of these characteristics.

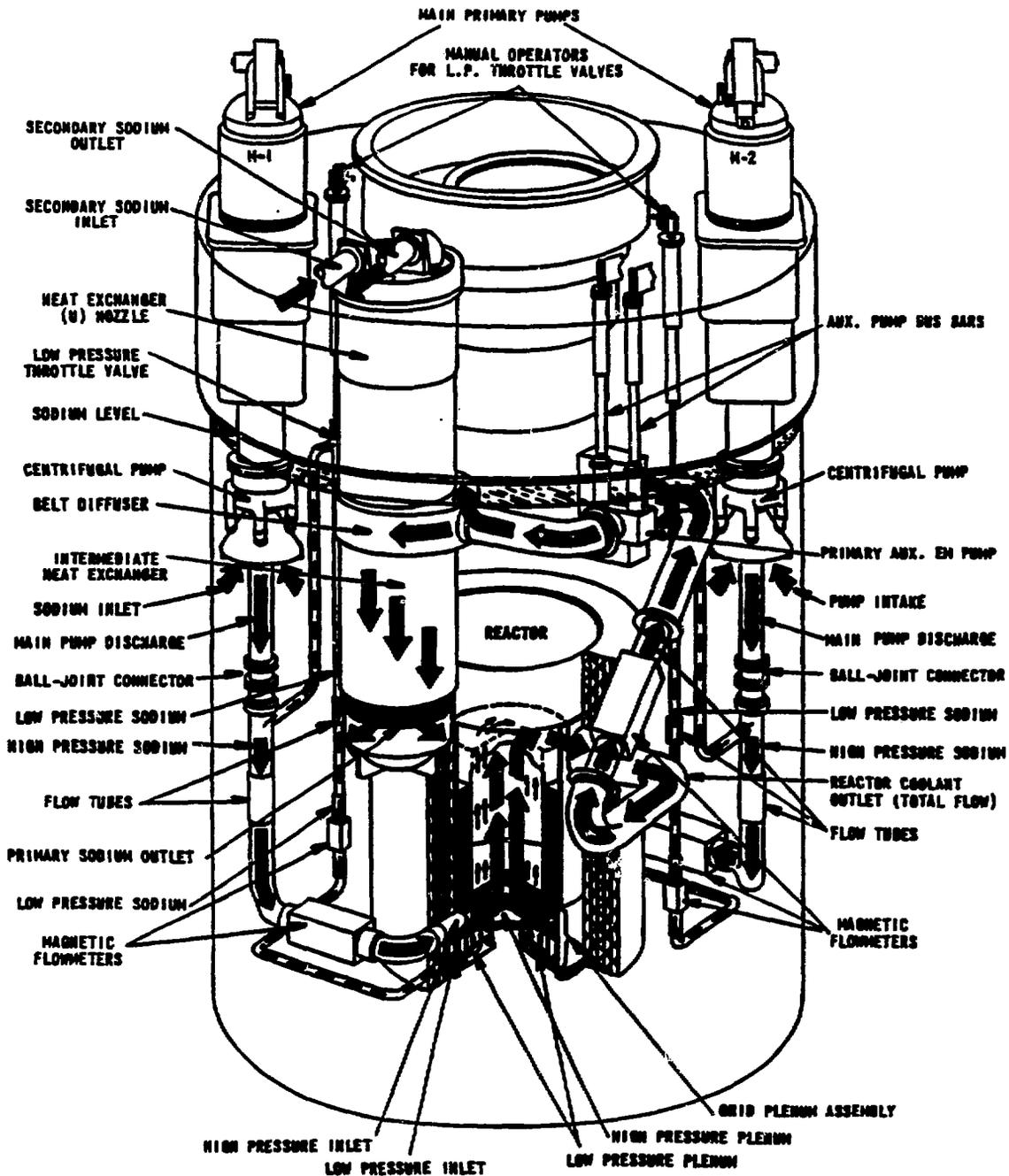
EBR-II is a piped-pool design with the entire primary system and reactor vessel submerged in a large pool of sodium contained in the primary tank (Fig. 1). The two submerged primary sodium pumps pump sodium into the reactor vessel via reactor inlet piping. The sodium passes upward through the core and is discharged from the reactor vessel into the reactor outlet pipe which transports the hot sodium to the Intermediate Heat Exchanger (IHX) where it transfers the heat to the secondary sodium system. The primary sodium is then discharged from the IHX into the bulk sodium in the primary tank. The entire process takes place within the primary tank and the contained piping need not be leak tight. There are no active valves in the primary system. Additionally, major components of the fuel handling system are submerged in the primary sodium and remain in place during reactor operation. In particular, location of the fuel storage basket in the primary tank allows fuel to be transferred to and from the primary tank during reactor operation.

The secondary sodium system transports heat from the primary sodium in the IHX to a series of superheaters and evaporators where superheated steam is generated and transported to a conventional turbine generator (Fig. 2). The secondary sodium system also has no active valves. The secondary sodium pump is a highly reliable and versatile electromagnetic pump. The superheaters and evaporators are of a duplex tube design which provides a double walled barrier between the sodium and the water/steam (Fig. 3). This design has performed very well and has been subjected to considerable analysis and examination over its operating life.

In addressing life-time considerations for EBR-II, critical components and key aging issues have been identified, surveillance and performance trending programs have been developed, and plant-life-extension planning has been integrated into technical support activities. The most important conclusion from this work is that by their nature, LMR systems have superior life-time potential.

## II. TESTING PROGRAMS

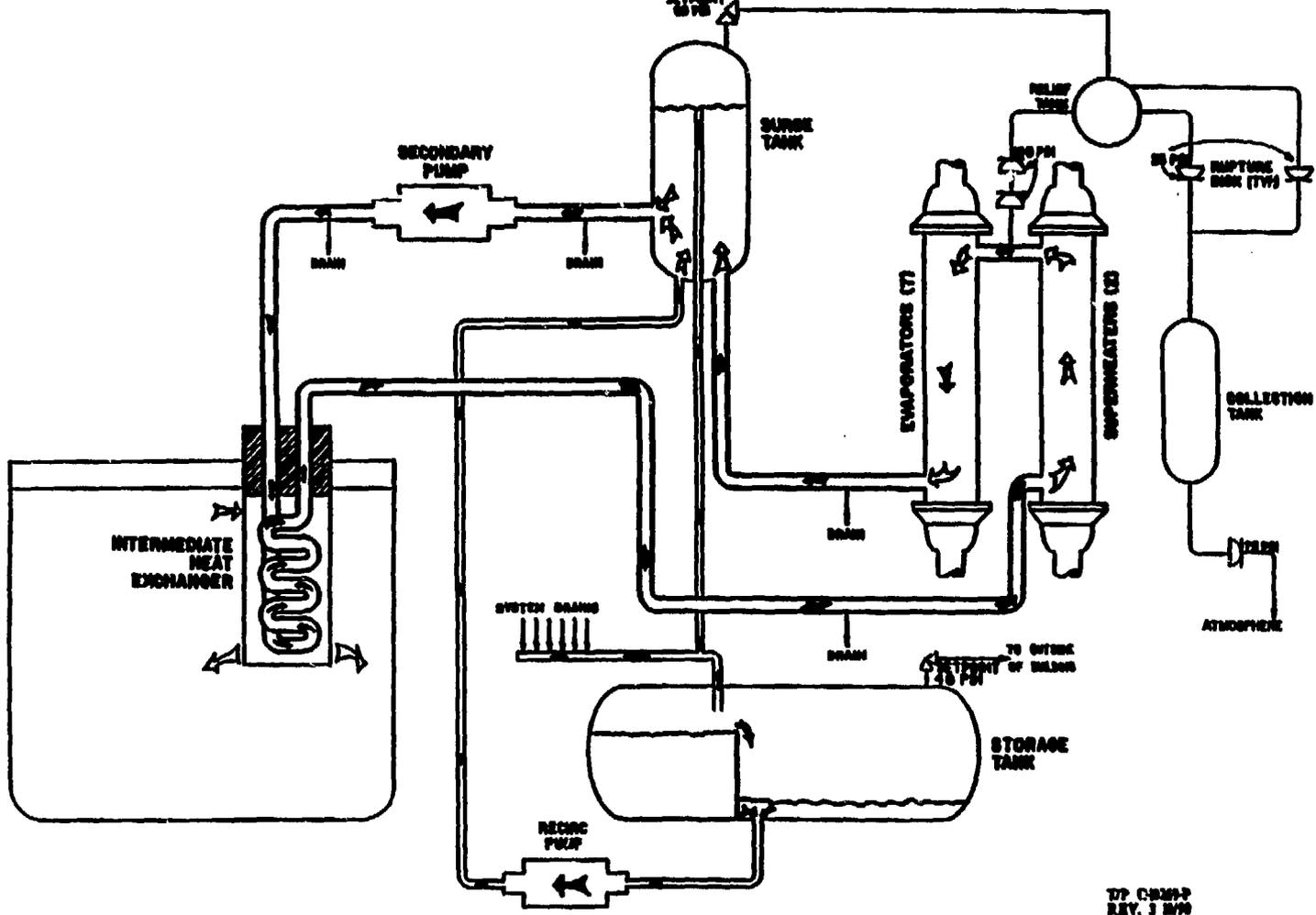
Testing programs to support LMR development and safety research have evolved in EBR-II from the initial mission of demonstrating feasibility of a complete LMR power plant with on-site fuel processing, to its current mission as the Integral Fast Reactor (IFR) prototype with its advanced metallic fuel (U-Pu-Zr) and reprocessing by electrorefining<sup>2</sup> (Fig. 4). The initial demonstration of fuel reprocessing 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 oxide fuel testing and materials development. In this second phase of testing, the fuel for both the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor Plant was developed. It gave the U.S. Program and EBR-II, as part



PRIMARY COOLING SYSTEM

T/P-C-10075-A  
 REV. 1 10/81

Fig. 1



DP 6289-3  
REV. 3 11/70

**SECONDARY SODIUM SYSTEM (SIMPLIFIED)**

Fig. 2

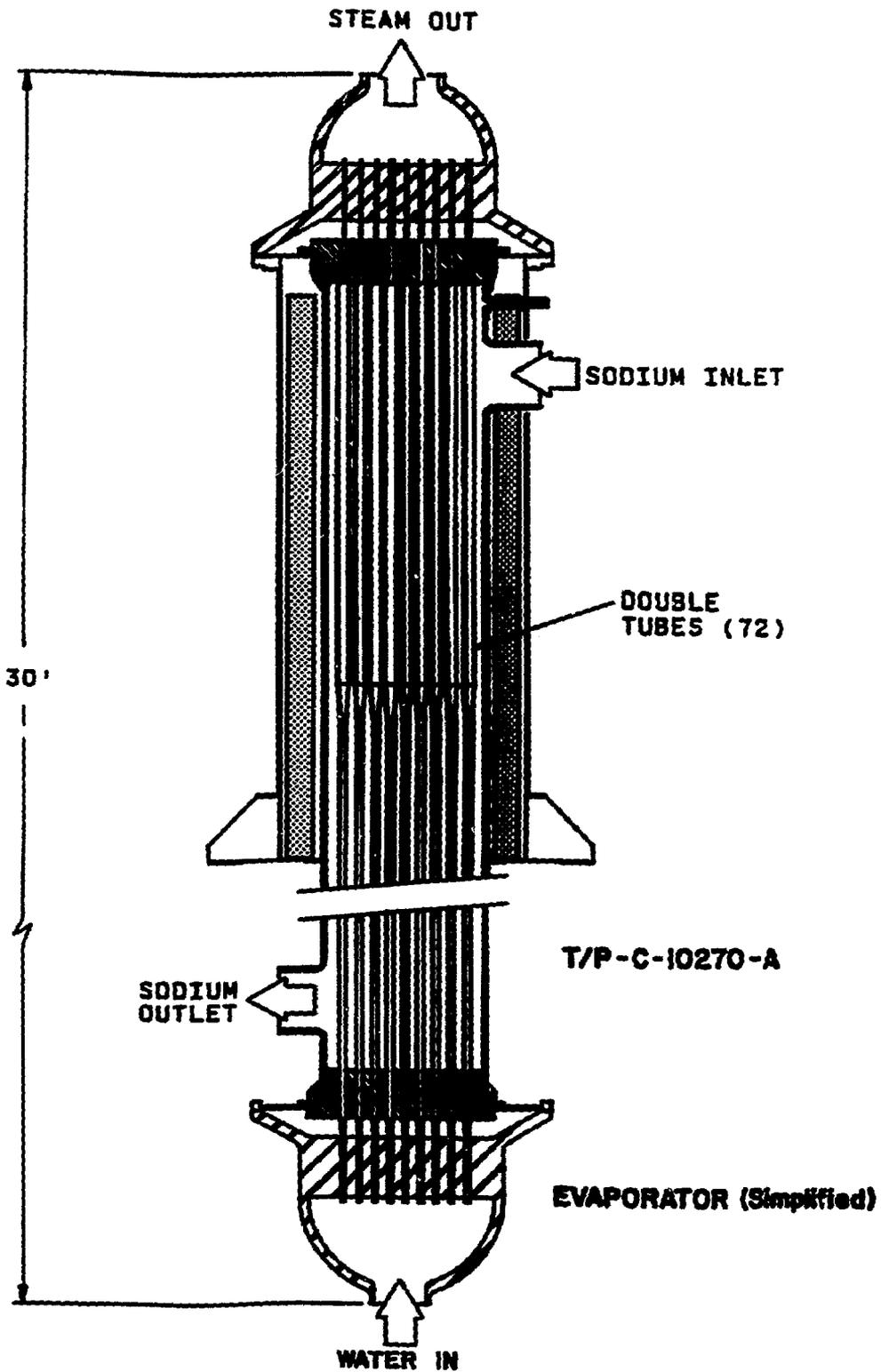


Fig. 3

# MISSIONS OF EBR-II DURING TWENTY-SEVEN YEARS OF POWER OPERATION

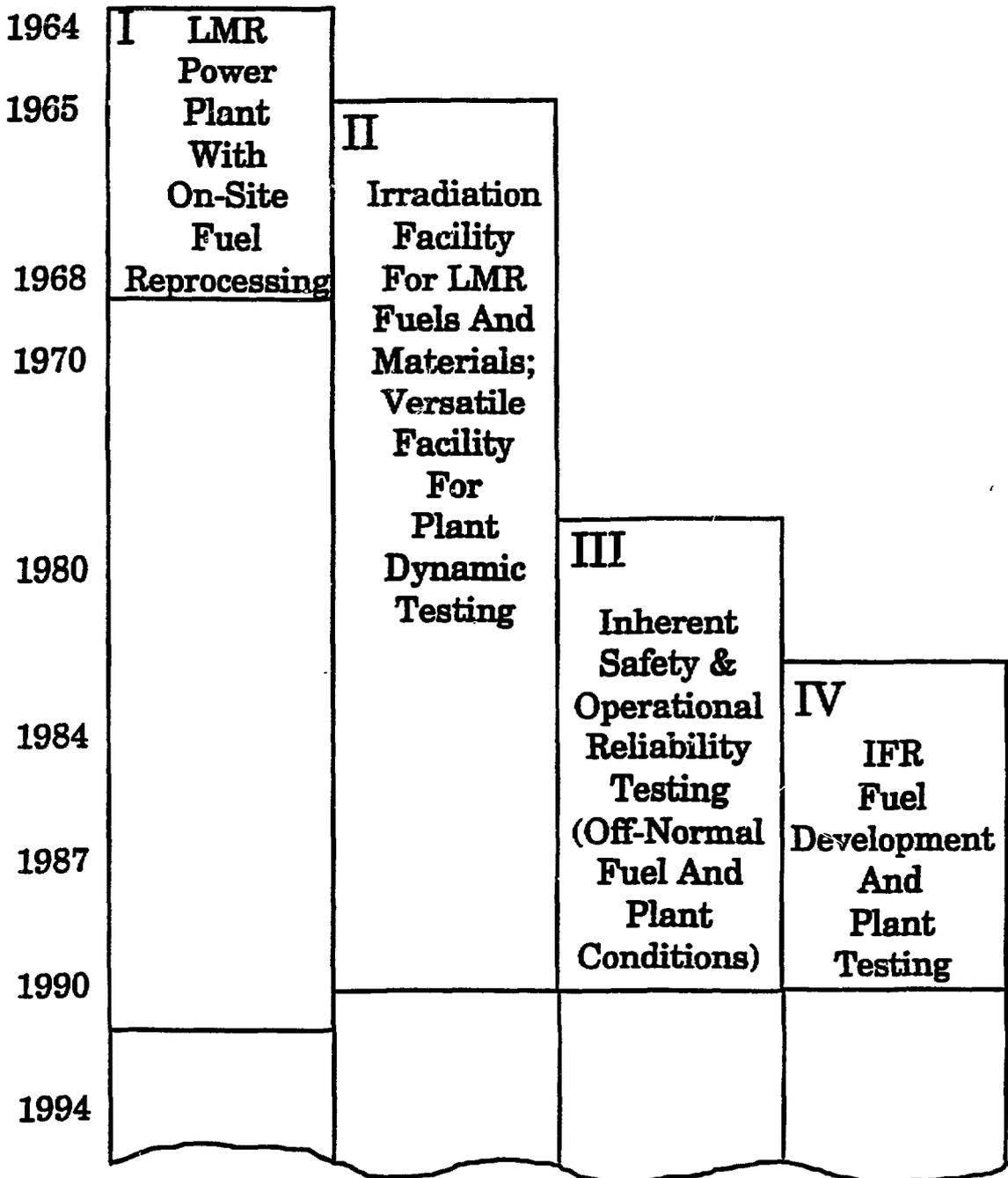


Fig. 4

of it, the opportunity to comprehensively test the performance of a wide variety of fuels, including metal, oxide, carbide and nitride. It was from this experience that the benefits of metal fuel irradiation performance were established. EBR-II's burnup limit for its driver fuel was increased considerably; metal IFR fuel has been taken to greater than 18% burnup.

The third phase of the EBR-II testing program began in the late 1970s and expanded conventional steady-state fuels testing to include tests of the response of metal and oxide fuel and the plant to off-normal conditions, such as breached cladding and operational transients. Plant transients, such as loss-of-flow and loss-of-heat sink and transition into natural convective flow were addressed. During this test phase, plant upgrades included providing special instrumented test facilities for in-core measurements, fission gas handling systems for breached fuel testing and computer controlled power shaping for transient testing. The most dramatic result of this work was the series of tests conducted in 1985-1986 which culminated in a demonstration of the ability to accommodate loss-of-flow and loss-of-heat sink without scram<sup>3</sup> (Figs. 5 and 6). Just as important has been the testing more directly related to operation, laying the basis for simplified control, and opportunities for automation without creating concern for safety.

The IFR is derived from the sum of this experience. The concept includes a uranium-plutonium-zirconium (U-Pu-Zr) metallic fuel, a pool type LMR, high breeding ratio core designs, core and plant designs to enhance passive safety and an integral pyrometallurgical fuel processing facility with remote injection casting of reprocessed metal fuel<sup>4</sup> (Fig. 7). Use of EBR-II as the IFR prototype to develop and demonstrate the IFR technology has four major objectives as follows:

**A. Fuel and Fuel Cycle:**

To demonstrate satisfactory performance of reprocessed U-Pu-Zr fuel, in conjunction with the ANL Fuel Cycle Facility.

**B. Safety:**

To demonstrate safety advantages and set the precedent for safety documentation appropriate to an IFR.

**C. Operation:**

To demonstrate operational advantages of EBR-II/IFR considering benign response to controller malfunctions and benefits of sodium for operation and maintenance.

**D. Regulation Compliance:**

To demonstrate that compliance with regulations involving operation of EBR-II/IFR is enhanced by virtue of characteristics inherent in design.

The role of EBR-II as the IFR prototype follows from demonstrated behavior of U-Pu-Zr metal fuel in which burnup has exceeded 18 at.%,<sup>1</sup> demonstrated passive safety features of the metallic fuel<sup>3,5</sup> and soon to be demonstrated fuel reprocessing, and the integral closed fuel cycle.<sup>6</sup> EBR-II is also the test facility designated to demonstrate self consumption of plutonium and actinides.<sup>7</sup>

# LOSS-OF-HEAT-SINK WITHOUT SCRAM TEST IN EBR-II

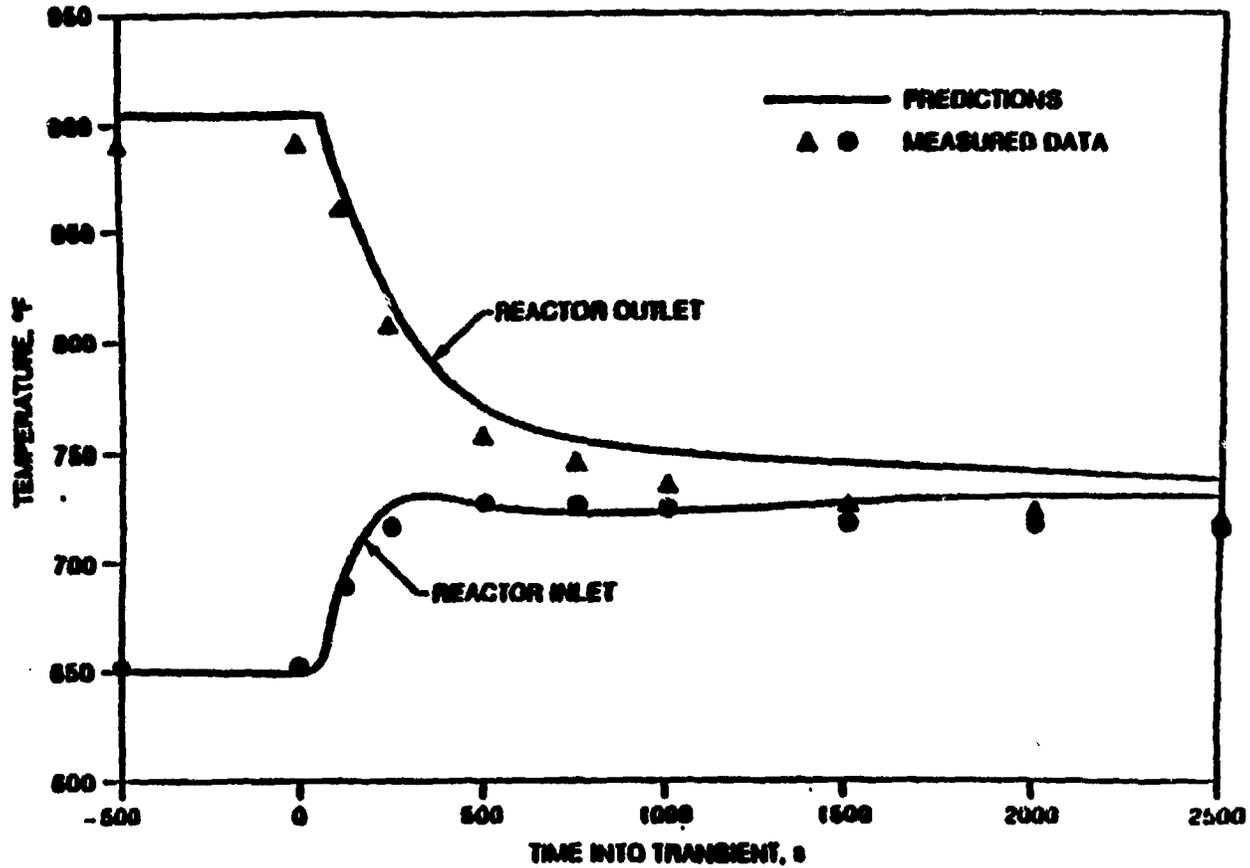


Fig. 6

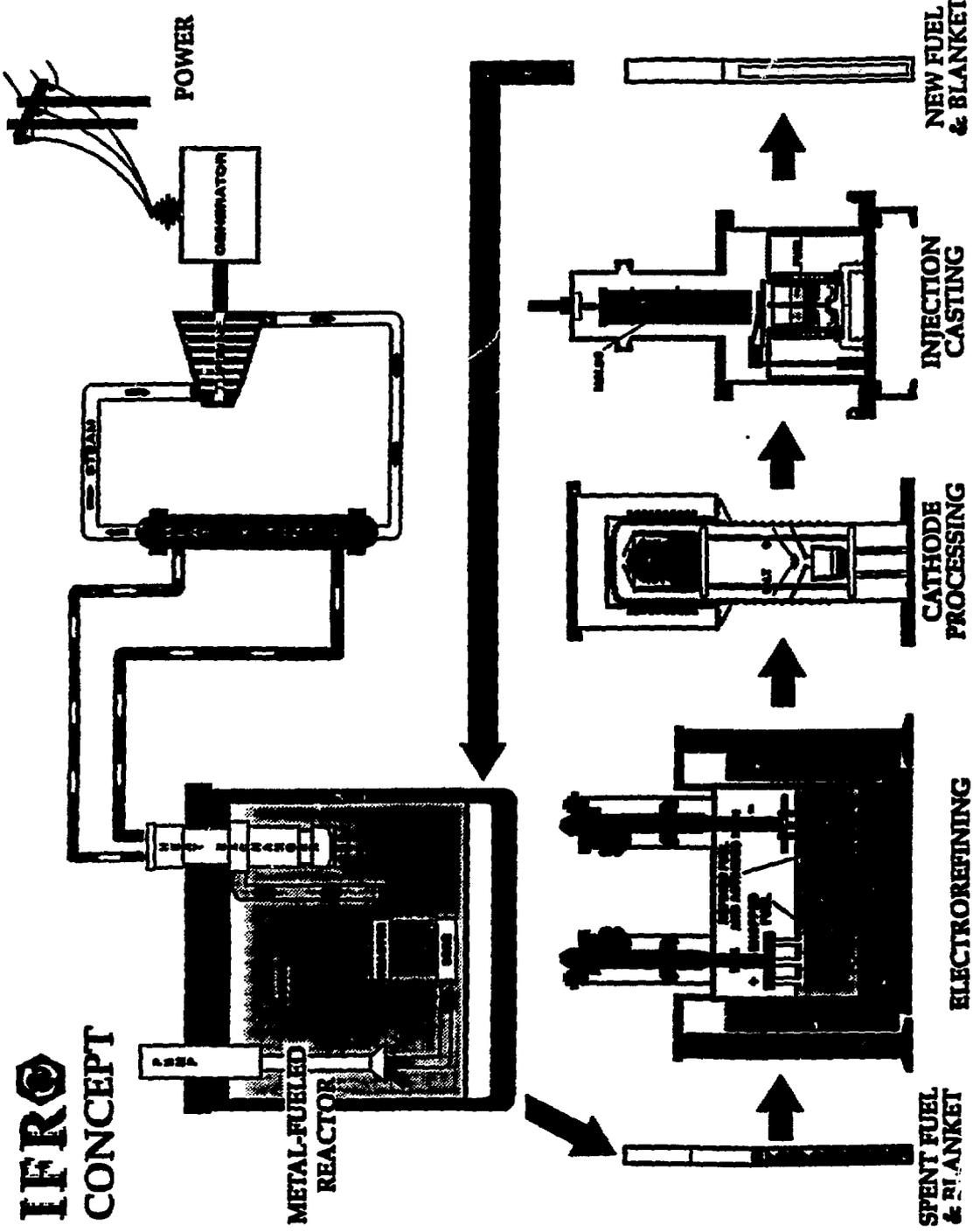


Fig. 7

Incorporation of the integral fuel cycle involves transferring spent fuel to the Fuel Cycle Facility for reprocessing, remote fabrication into new assemblies and returned to the reactor (Fig. 7). The U-Pu-Zr fuel and the electrorefiner are central to the process. The initial step of the processing occurs in the electrorefiner where the chopped fuel (cladding included) is dissolved in a halide salt solution. The fuel remains in the metallic state during the cycle and the cladding material and fission products are separated to the salt phase and the uranium and plutonium are collected at the cathode.

The cathode is processed and the resulting fuel and actinides are melted in a casting furnace and new fuel pins are made by an injection casting technique. Except for the electrorefiner, a very similar fuel cycle was demonstrated in EBR-II during the first five years of reactor operations.<sup>8</sup> Over 700 irradiated assemblies of all types were processed. Of these, 560 were fuel bearing subassemblies which were processed to separate the fuel and most of the fission products. This operation produced 34,500 reprocessed fuel elements that were fabricated remotely into 418 subassemblies and returned to the reactor.

EBR-II has another equally important role for the IFR. This derives from the passive safety features of the concept and promises substantial savings in operations, safety documentation, preparation and maintenance, and environmental and radiological impacts. The EBR-II safety documentation is currently being 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, work is ongoing to investigate and demonstrate the operational safety features of the IFR that result in savings in operations. This promises innovative approaches to 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.

### III. FUEL PERFORMANCE: METAL

Safety and operability issues related to metal fuel are associated with the irradiation behavior and potential for fuel failure and fault propagation, and the behavior of the fuel under breached conditions.

Metallic fuel has been studied in EBR-II since reactor startup, with the last five years devoted to development of U-xPu-10Zr (x = 0 to 19 wt %) fuel for the IFR concept. Experience has shown, that with the specified design parameters, fuel expansion and fractional gas release are predictable and manageable with no appreciable fuel/cladding mechanical interaction. Tests have been done to extend the burnup of the U-Pu-Zr fuel to greater than 18 at.%.<sup>9</sup> The design provides for a 73% planar smear density, a fuel to plenum ratio of one, a cladding outside diameter of 0.584 cm (0.23 in.), a wall thickness of 0.0381 cm (0.015 in.) and a D9 cladding material. Tests have also been done with advanced alloys such as HT9. A key feature of the metal fuel is its ability to expand and remain compliant at operating temperatures. At about 1 to 2 at.% burnup, the fuel expands to the cladding because of the large volume of entrained fission gas. Above 2% burnup, the gas bubbles interconnect and the gaseous fission products are released to the plenum (Fig. 8). It is these features that

## Fuel Performance Results

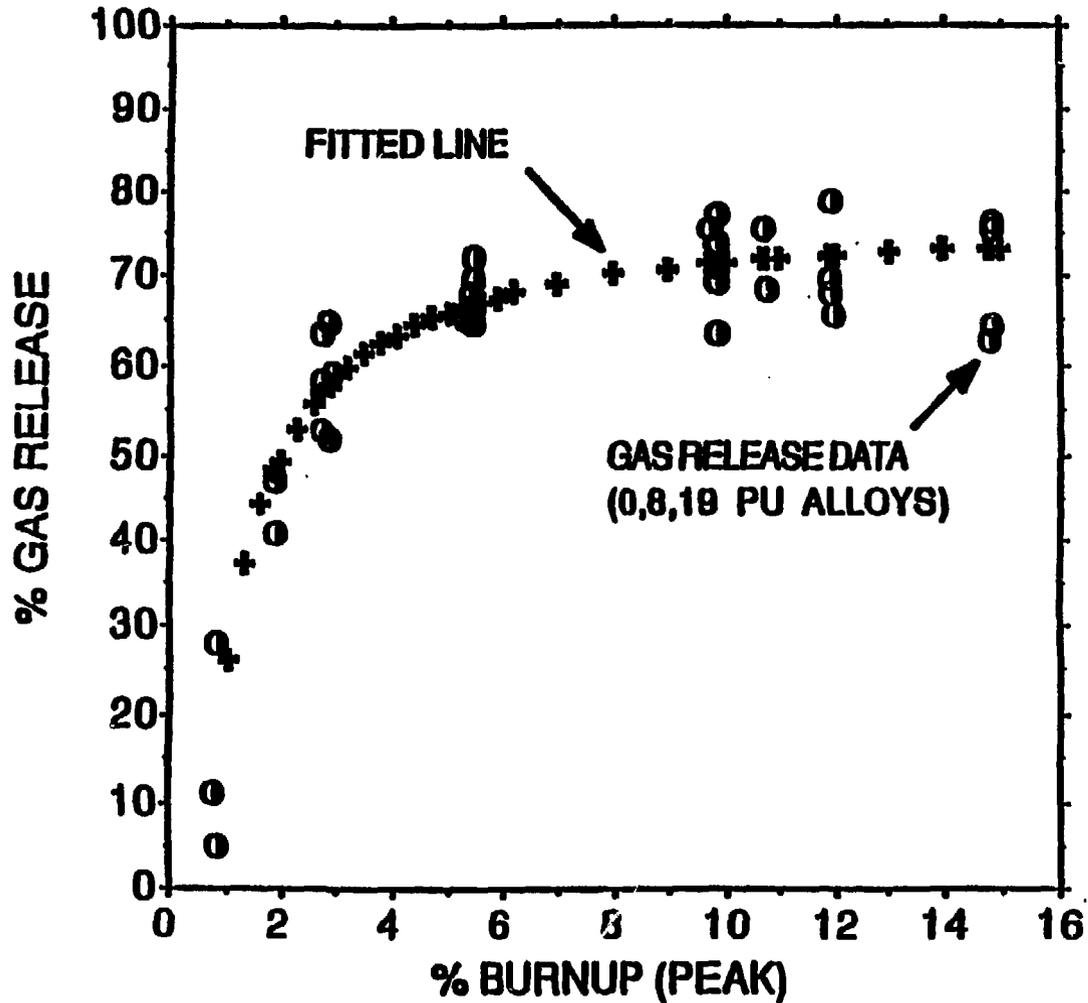


Fig. 8

allow high burnups without cladding failure. The effect of increasing Pu content is to reduce the axial fuel elongation as a function of burnup. This is believed to be due in part to fuel slug cracking during free swelling below 2% burnup.

The irradiation program, as well as out-of-reactor tests and analyses are also directed to furthering the understanding of IFR fuel performance during off-normal conditions. A key question is the fuel performance during steady-state operation following cladding breach. RBCB tests have been done on different ternary alloy fuels with various cladding types. Test pins with burnup in the range of 3 to 12 at.% have been prethinned and irradiated in the reactor to breach and beyond. In all cases the results have been benign, demonstrating the safe operation with failed fuel and the lack of failure propagation. A test pin of U-19Pu-10Zr was irradiated for 233 days beyond breach with no adverse consequences to reactor safety or operation.<sup>10</sup>

A major concern of metal fuel is the potential for liquid phase penetration of the cladding at temperatures at or above the eutectic temperature of the fuel and cladding. This temperature varies with fuel and cladding type and has been determined to be greater than 700°C for all fuel and cladding combinations. Out-of-reactor tests have shown limited and very low penetration rates at the temperature of formation of the initial phase.<sup>11</sup> At higher temperatures, the penetration rate is much faster; a test pin (U-19Pu-10Zr) heated to 800°C showed a 26% reduction in the cladding thickness after one hour. Since this time at temperature is much greater than typical accident sequences, the safe features of the metal fuel are demonstrated.

This conclusion was verified by EBR-II test XY-22,<sup>11,12</sup> in which metal fuel of varying burnup was operated at temperatures up to 800°C in a 61-element subassembly. The subassembly operated for 42 minutes before failure of a high burnup element. The breach occurred at the fuel cladding interaction. The breach location and failure mode agreed well with the pretest prediction. The remaining elements were reconstituted in another subassembly, and then run-to-cladding-breach and slightly beyond to the end of the reactor run. Actually, two breaches were encountered, one at 10 at.% and the other at 10.2 at.% burnup. Both breaches occurred at the fuel restrainer dimple at the same burnup where previous breaches of this fuel type were experienced. The lower burnup elements were then reconstituted into another test and were irradiated to >11 at.% without breach. Postirradiation examination of the elements showed no significant fuel/cladding interaction with some fuel restructuring due to the high temperature operation. These tests demonstrated the safe and reliable operation of metal fuel following long-term over-temperature operation and that element lifetime was not shortened.

During the 1980s a number of plant tests have been conducted in EBR-II, which taken collectively, demonstrate the safe operation of a metal fueled advanced LMR. Following an early demonstration of natural convective cooling in EBR-II, tests were conducted that led to a demonstration of safe reactor shutdown following a loss of forced cooling and loss of heat sink from full power without scram.<sup>13</sup> In each case the

reactor was shut down passively by negative feedbacks and transient and equilibrium temperatures were demonstrated by measurement to be below concerns for fuel integrity and reactor safety.

A primary coolant pump run-up test<sup>14</sup> was also conducted to demonstrate possible response to this over-cooling event. Primary flow was increased from 32 to 100% in 20 seconds from an initial power-to-flow ratio of 1.0. Power followed flow and leveled off at about 90%. Thus, the final power-to-flow ratio was less than 1.0 and core exit temperature was less than at the starting point. This was because of the negative fuel temperature coefficient caused by the increased temperature of the fuel. During the experiment, the secondary flow was conservatively controlled to keep the inlet temperature nearly constant. It was also demonstrated 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 EBR-II.

The results of the whole plant testing and the passive safety demonstration programs have broad implications for safety design and operation of advanced LMRs. For example, these tests have demonstrated the importance of negative feedback reactivity and magnitude of the Doppler Coefficient, the importance of longer flow coastdown times of primary coolant pumps, and the need for detailed overall thermal-hydraulic design to enhance natural convective cooling. They have also indirectly suggested reactor designs that emphasize high internal breeding to minimize the burnup swing. In this way, the amount of reactivity vested in control rods is minimized and the problems of rod induced transient overpower accidents are mitigated.

These tests also suggest advanced control strategies in which reactor power is controlled over the load following range by either primary system flow, secondary system flow and/or turbine admission valve. The feasibility of these control schemes for metal fueled LMRs has already been demonstrated.<sup>15</sup>

#### IV. LMR OXIDE FUEL TESTING IN EBR-II

In cooperation with PNC of Japan, EBR-II has conducted an important oxide fuel testing program to address the performance of the fuel with breached cladding or in response to overpower transients. These tests provided data on failure detection, fuel-sodium reaction product formation, fuel pin performance and primary circuit contamination.

To answer early concerns about fuel safety and reliability, preparations were made in 1976-77 to ready EBR-II for RBCB testing.<sup>16</sup> They included: installing systems to control fission gas activity and to remove cesium from sodium; upgrading the delayed-neutron (DN) detectors; and several defected fuel tests.<sup>17</sup> Four tests of natural failures were next performed, with encouraging results.<sup>18</sup> A program of fourteen tests was then begun in 1981 between the U.S. Department of Energy and the Power Reactor and Nuclear Fuel Development Corporation of Japan. Irradiations were completed by 1986.

The four phases of the program involved: (1) Scoping tests to determine the feasibility of RBCB operation in EBR-II, (2) tests to determine the kinetics of fuel-sodium reactor product formation, (3) tests to address the signals associated with fission gas or delayed neutrons, and (4) tests to address defective welds, large diameter pins, blanket elements and other variables.

When RBCB testing of mixed-oxide fuel pins began in EBR-II, lack of real experience of the rate and extent of fuel-sodium reaction in-reactor made it uncertain how long breached pins could safely continue operation. That reaction might lead to fuel loss, failure propagation, or even formation of local blockages in test subassemblies were palpable fears. The scoping tests<sup>18</sup> dispelled this early concern for reactor safety and indicated, in an empirical fashion, that operation with fuel failures could continue for days to weeks without difficulty.

Interest next focussed on understanding the behavior of individual RBCB pins: how reaction affected their performance, how much contamination they might release, and whether their condition could be monitored in-reactor. These three questions became the objectives of the fourteen tests performed during 1981-1986. In large measure the results were positive, confirming earlier findings.<sup>19</sup> Although the reaction product opens breaches and raises fuel temperatures, once it forms a coherent layer, it suppresses loss of fuel and fission products and promotes release of DN signals that signify its presence. If required, RBCB operation for weeks to months seems feasible without serious pin deterioration or significant contamination of the primary circuit. Normal radiation embrittlement of cladding, which promotes breach extension, or high-temperature operation, which prohibits reaction product formation, may be two conditions that limit benign operation. A final phase of testing has now begun in EBR-II with emphasis on the RBCB behavior of large diameter advanced oxide pins.

Source subassemblies are being used to preirradiate a variety of 7.5 mm pins to burnup of 5 at.% by early 1991. Variables include linear power, fuel stoichiometry, and fuel smear density. These pins will be predefected in the plenum region and reirradiated to establish the impact of RBCB operation on pin thermal and mechanical performance, including fuel center melting. Tests have already been performed to establish impact at zero burnup; it was found to be small. Another test is under irradiation to simulate the RBCB behavior of a pin bundle at high burnup with these large diameter pins.

These source pins then will be employed in transient overpower tests to be performed in 1992 and 1994 in EBR-II, and in whole pin transient overheating tests in a hot cell (beginning 1992). Both test series will also contain reference size pins from the Phase I program TOP-4 tests which have been taken to 15 at.% burnup in a test extension in EBR-II.

Other tests will probe the characteristics of DN release, fuel loss mechanisms from breached pins, and breach propagation potential of advanced claddings. The fuel loss tests involve development of a special contamination capsule, and tests of a special triple-station DN monitoring system now being proved out-of-reactor.

## V. PLANT LIFE POTENTIAL

As The EBR-II facility was approaching 20 years of operation in the early 1980s, programmatic commitments required reliable operation to at least 30 years. ANL began a process of identifying and evaluating aging, reliability, and life-extension issues that could preclude achieving the operational goal, and developing plans for assuring reliable extended-life operation.<sup>20</sup>

In order to accomplish this, a complete engineering and operational assessment of all major and most minor plant systems was performed.

This 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 preclude or otherwise inhibit the ability of the plant to operate reliable to 30 years. The criteria for developing the list included the potential for failure or malfunction of the component to cause a plant shutdown of three months or longer; and the potential for the component to significantly reduce plant reliability. The critical components list is shown in Table 1.

Table 1 Critical Components List

### Nonreplaceable or Nonrepairable

Safety Rod Drive System  
Reactor Grid-Plenum Assembly  
Rotating Plugs/Seals  
Primary System Instrumentation  
Primary Tank  
Reactor Vessel Neutron Shield

### Replaceable or Repairable

Control Rod Drive System  
Fuel Handling System  
(in-tank)

- Core gripper
- Holddown
- Reactor Cover-Lift Mechanism
- Fuel Storage Basket
- Fuel Transfer Arm

Primary Sodium Pumps  
Intermediate Heat Exchanger (IHX)  
Reactor Building Polar Crane

In addition to the engineering and operational assessment, other work was ongoing that, while directed towards research and development programs, has direct application and benefit to the plant-life extension effort. This work included thermal stress and cyclic fatigue studies for major components, and reactor material neutron radiation damage studies and tests. These projects provided the analytical and material testing support required to establish a technical basis for extended life operation.

The cyclic fatigue study included thermal stress and creep analyses of the reactor vessel, vessel cover, IHX, primary piping, superheaters, and secondary sodium piping. The analyses were performed to ensure that these major components could safely withstand a series of transient tests planned for the facility. The most highly stressed component was identified for each of the transient series. The limiting component overall, based on

allowable thermal cycles, was identified to be the IHX. However, the cumulative cycles at 30 years of life, including transient testing, was still less than half of the allowable cycles based on ASME code criteria. In addition, the IHX is a replaceable component.

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

The grid plate and graphite shielding were identified to be of primary concern when considering plant life extension because of their nonreplaceable nature and susceptibility to neutron damage due to their proximity to the core. Some of these 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 reach a neutron fluence equivalent to the grid plate fluence at 45-50 years of operation. Tensile tests of the 304 SS material samples have indicated that residual ductility remains well above reasonable allowable minimums, and void swelling is low. Likewise, tests performed on the irradiated graphite samples indicate that the graphite is in a densification phase and has not yet started to swell at these fluences. Tests also indicated that there was no stored energy in the graphite.

From the results of these assessments, analyses, and tests, indications were that the reasonable expected technical lifetime estimate for EBR-II was well beyond 30 years and possibly 50 years or more before approaching any aging limits. Also, with the newly defined mission of operation as the IFR prototype during the 1990s, and other new programs coming into the facility, it became important that EBR-II be able to operate reliably to a 40-year lifetime (1964-2004). With that as the goal, the plant-life extension program was refocused to build upon the original results of the plant engineering and operational assessment. Thus, emphasize much more of the technical evaluations of the irradiated materials samples in order to establish the minimum usable lifetime of the nonreplaceable components; to provide additional technical bases to support operation to a minimum of 40 years.

In consideration of a 40-year lifetime, the critical components list continues to be a focal point for direction of resources for surveillance, diagnostics, preventive maintenance, spare parts, and upgrade modifications for improving long-term reliability. The major component thermal stress and cyclic fatigue analyses have been updated to consider 40 years of operation. The updated analyses indicate that with very conservative assumptions, the limiting component IHX will have accumulated only 60% or

less of the allowable lifetime cycles. As indicated above, some of the irradiated materials samples examined already have accumulated neutron exposure to establish physical property characteristics beyond the 40 year operational life. Other irradiated material samples are undergoing examination and testing to add to the material database.

In the balance-of-plant, the issues associated with aging and life extension are not considered to be as critical because of the accessibility and maintainability of both the secondary heat transport system and the steam system. 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 proven to be highly reliable. Other power-producing sodium-cooled reactors have experienced difficulties with steam generators, but the duplex tube design used in EBR-II has been very reliable.

Furthermore, an EBR-II superheater was removed from service in the late 1970s after a degradation in heat transfer efficiency was experienced; and replaced with a modified evaporator. 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 pre-stress of the mechanically bonded duplex tubes. The other superheater has metallurgically bonded duplex tubes and has not experienced any degradation.

## V. CONCLUSION

The 27 years of EBR-II operation has established an important new direction for LMR research and has given us confidence that it will succeed. It is a critical time for LMR development worldwide, and the IFR is providing important answers for today's issues of safety, cost, reprocessing and waste storage. The demonstration of this technology in the 1995 time frame provides the basis in the U.S. for moving to the next phase.

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