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RECENT OPERATING EXPERIENCES AND PROGRAMS AT EBR-II

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ABSTRACT

Experimental Breeder Reactor No. II (EBR-II) is a pool-type, unmoderated, sodium-cooled reactor with a design power of 62.5 MWt and an electrical generation capability of 20 MW. It has been operated by Argonne National Laboratory for the U. S. Government for almost 20 years. During that time, it has operated safely and has demonstrated stable operating characteristics, high availability, and excellent performance of its sodium components.

The 20 years of operating experience of EBR-II is a valuable resource to the nuclear community for the development and design of future LMFBR's. Since past operating experience has been extensively reported, this report will focus on recent programs and events. Specific areas which will be briefly summarized are as follows:

- Operational Reliability Testing of Fuels
- Installation and Operation of a Computer-controlled, Fast-speed Control-rod Drive
- Recent Maintenance Experience on a Primary Sodium Pump and A Sodium-wetted Fuel-handling Transfer Arm
- Investigation and Identification of a Major Source of Impurities in the Primary Sodium
- Retrieval of a "Lost" or "Dropped" Subassembly, and
- Speculation on the Future of EBR-II

INTRODUCTION

Experimental Breeder Reactor No. II (EBR-II) is an unmoderated, sodium-cooled reactor with a design power of 62.5 MWt. EBR-II has a pool-type primary system and a complete power plant, which produces 20 MW of electrical power through a conventional turbine-generator.

EBR-II was constructed for the AEC by Argonne National Laboratory between 1957 and 1963. Initial power operation of EBR-II began in 1964. During the last 20 years, EBR-II has operated safely and has demonstrated stable operating characteristics, high availability, and excellent performance of its sodium components.

The original goal of EBR-II operation was the demonstration of the feasibility of a sodium-cooled fast reactor operating as a power plant with fuel-processing capabilities provided by an adjacent fuel processing facility. After initial operation, EBR-II's role as a demonstration plant was reoriented to that of an irradiation facility in 1965. Since that time, the focus of the irradiation program has evolved from a very conservative, steady-state program to the present Operational Reliability Testing (ORT) Program. The ORT Program utilizes EBR-II for a more aggressive irradiation program consisting of run-beyond-clad-breach (RBCB) tests and operational transient tests simulating duty-cycle transients and mild over-power transients on fuel pins, and thermal-hydraulic testing of clusters of fuel pins under both normal and natural-convection modes of cooling for testing and verification of shutdown heat-removal codes.

During the course of operation of EBR-II and evolution of the irradiation program, many modifications have been made to enhance EBR-II's operational and experimental capabilities and a great deal of experience has been gained in the area of maintenance and repair of sodium components. Although there have been no equipment failures which have caused plant shutdowns exceeding four months, and no major or minor nuclear incidents, there have been several significant operational occurrences which should be of interest to the nuclear community.

This report will briefly summarize the present program of operational reliability testing at EBR-II; it will describe the new computer-controlled control-rod-drive system; and it will discuss several significant maintenance activities and two unusual events which have occurred in the last several years.

OPERATIONAL RELIABILITY TESTING AT EBR-II

Testing of fuels for eventual use in commercial LMFBP's has, until recently, been focused on steady-state irradiations to ascertain routine operating characteristics and on transient tests designed to explore behavior of fuels under severe accident conditions. Following the Three Mile Island accident, it was realized that emphasis must also be given to the milder, but more credible, range of transient which occurs between the two previously mentioned extremes of fuels testing.

A program has been developed at EBR-II to allow testing of fuels in this middle ground. This testing program has been designated the Operational Reliability Testing (ORT) Program and is designed to utilize EBR-II's operational capabilities in the following areas of investigation.

- Run-beyond-clad-breach (RBCB) testing of fuel elements under normal, steady-state operating conditions. This part of the program is an outgrowth of the steady-state run-to-clad-breach (RTCB) program, which has been in progress at EBR-II since the early 1970's.

- Transient testing of present and advanced fuel-element designs under slow-ramp-rate (0.01-10% power/s) conditions. These tests are designed to serve two functions: (1) show that conservatively designed oxide fuel and blanket pins and assemblies can survive a variety of duty cycle and PPS-terminated events without performance degradation, and (2) provide data from aggressively designed pins that amplify the effects of a given event. The tests in this program can be categorized into three general areas:

  Breaching Threshold: Extended single transient overpower events on preirradiated pins.

  Duty Cycle Events: Periodic 15% overpower events and periodic reduced-power operation superimposed upon normal steady-state operation.

  Breached Pin Behavior: Transient overpower events on a breached pin.

- Shutdown-heat-removal testing (SHRT) to investigate natural-convective-cooling phenomena in EBR-II under a variety of protected and unprotected transient categories.

The first two transients on experimental fuels in the ORT Program were performed in February 1983. These were slow (0.1% power/s), manually controlled transients which resulted in an approximate 60% overpower transient on an oxide fuel test subassembly and an oxide blanket test subassembly. These transients were part of the "breaching threshold" area of testing. Further transient testing in this area will be conducted on a once-per-year basis. The follow-on transients currently planned will be controlled by the automatic-control-rod-drive system (ACRDS) and will impose approximately 60% overpower transients on the experiments at a ramp of about 8% power/s.

Beginning in February 1984, a 15% overpower transient at a ramp of 0.5% power/s will be conducted at the end of each reactor run. These will be part of the "duty cycle event" tests.

The remainder of the "duty cycle tests" and most of the RBCB testing will be integrated with steady-state reactor operation. Shutdown-heat-removal testing (SHRT) will be conducted in the same testing time frame as the 60% overpower transients because the core configuration required for the SHRT is very similar to that required for the 60% overpower testing.

AUTOMATIC CONTROL ROD DRIVE SYSTEM (ACRDS)

The original reactivity control system limited control-rod motion to a speed of about 5 in./min. This resulted in a reactivity-insertion capability of less than $0.01/s. The transient testing program identified a need for a control-rod-drive system capable of inserting reactivity at a rate of $0.09/s, using a standard EBR-II control rod which has a reactivity worth of about $0.70-$0.90.

Since a reactivity insertion rate of $0.09/s corresponds to a rate of change of power in excess of 10%/s, it was obvious that the power transients could not be manually controlled. The required rate of change of power, coupled with the added requirement of maintaining the actual reactor power within 5% of the desired power profile, resulted in the decision to use a computer-controlled rod drive system on one of the eight control rods in the EBR-II reactor.

The new control-rod-drive system was installed as a test in October 1982. Following the initial testing and checkout of the system with both a low-worth, stainless steel dummy control rod and a standard EBR-II control rod, the ACRDS was used to run 13 transients. The transients were part of the initial program designed to qualify EBR-

II's driver fuel and systems for the subsequent ORT program transients.

The power profile defined for the fuel qualification testing was a 4 MWt/s power increase from 26 MWt to 62.5 MWt, a 12-min hold at full power, and then a rapid down ramp terminating at the initial power of 26 MWt. Because the maximum drive speed of the ACRDS was limited for safety reasons, it was necessary to scram one of the other seven control rods while simultaneously removing reactivity with the ACRDS rod to achieve the rapid down ramp desired.

The results of both the testing of the ACRDS and the qualification of EBR-II's fuel and balance-of-plant systems were excellent. The computer-controlled rod drive repeatedly produced the required power profile within the required degree of accuracy and reliability, and the driver fuel operated throughout the transient period with no indication of fuel pin failure.

Following the testing in November 1982, the ACRDS design was modified to provide two ranges of rod drive speed in automatic control. The additional speed range corresponds to the drive speed of the original control rod drives and will be used for slower transients which require more precise control of the power profile than can be accomplished manually. The modified ACRDS will be installed as a permanent modification during the spring shutdown in 1984 so that it will be available for the ORT transient testing which will begin in June 1984. In addition to its use during transient tests, the ACRDS will also be used to provide steady-state power control during routine operation. It is also envisioned that the control program will be expanded to allow use of the ACRDS during routine power changes on startups and shutdowns. This would be done via an interactive program which would allow the reactor operator to determine the power change and rate of change desired, enter the required parameters, and initiate the change.

MAINTENANCE ON PRIMARY SODIUM SYSTEM COMPONENTS

Contrary to commonly held beliefs, maintenance on sodium systems and sodium components is not impossible, nor is it an extremely dangerous evolution. In fact, maintenance on sodium systems and sodium-contaminated components which are either highly radioactive because of neutron activation and/or contamination with fission products has been performed at EBR-II many times in the last 20 years. During this time, the techniques and equipment have been continuously modified to improve the safety and efficiency of the maintenance operation and all operations are controlled with very thorough and detailed maintenance procedures.

Removal of a component from the primary tank is performed using three major pieces of equipment which are the nozzle adapter, the pulling caisson (or silo), and the handling caisson. The configuration of these handling system components varies with the size and expected radiation level of the primary system component being removed; however, the function of each handling system component remains essentially the same.

All removable primary system components are inserted into the primary tank through penetrations (nozzles) in the primary tank cover. The components are also suspended from the primary tank cover by the nozzles. A nozzle adapter is a device which provides a gas seal between the primary tank and the pulling caisson while a component is being moved into or out of the pulling caisson. The nozzle adapter contains an internal "trap door" or valve which can be closed to seal the primary tank after a component is removed from its nozzle.

The pulling caisson or silo is a variable volume column which is used to isolate the component in an oxygen-free environment while it is being transferred between the primary tank and the handling caisson. Depending on the size of the component being moved, the pulling caisson can be either a high-temperature fabric bag or a carbon steel cylinder and piston arrangement. The pulling caisson is only designed to provide a purged containment for the component and, as such, does not provide any structural support or protection for the component during transfer.

The handling caisson is a carbon steel container which can be fitted with mechanical braces to firmly position and support a primary tank component during its transfer to the sodium component maintenance shop (SCMS).

When a sodium-contaminated component has been transferred to the SCMS, it is washed by immersion in ethanol to remove residual sodium. The ethanol is circulated through the wash vessel while it is heated from ambient to about 60°C. The completion of the wash cycle is determined by monitoring the rate of change of hydrogen in the wash vessel nitrogen cover gas. The washing also partially removes loose fission-product contamination.

In the last two years, this process has been used to remove a primary pump for cleaning and refurbishment and also to remove the fuel handling system transfer arm for replacement of a bent shaft. The removal of the primary pump was performed in the spring of 1982. This maintenance activity had been anticipated a year in advance, and the early notice provided the opportunity to thoroughly check out the previously unused handling system components and fixtures and to perform "dry runs" on many of the handling operations. With a year's notice to train maintenance personnel and develop procedures, the actual removal, refurbishment, and reinstallation of the pump went very smoothly. The entire process took just over 40 days. In June 1983, the transfer arm failed unexpectedly. With the previous year's experience, the removal, repair, and replacement of the transfer arm was accomplished in just 46 days.

The philosophy concerning maintenance on major primary components has been to monitor them closely to detect signs of failure and to remove them only when problems occur or failure appears to be imminent. Routine preventive maintenance or inspection of the parts of primary system components immersed in sodium is not performed. Compatibility between this philosophy and reactor safety has been achieved by recognizing the potential for failure in safety-related primary systems, incorporating this potential into the basic safety philosophy when safety boundaries were determined, and by continuously testing for "operability" of safety-related systems.

UNIDENTIFIED SOURCE OF IMPURITIES IN THE PRIMARY SODIUM

The purity of the primary sodium is monitored by on-line hydrogen and oxygen meters and by measuring the "plugging temperature" of the sodium. The argon cover gas is monitored continuously for the presence of helium, hydrogen, oxygen, and nitrogen with on-line gas chromatographs and for hydrocarbons with an on-line hydrocarbon analyzer. The normal plugging temperature for the primary sodium is about 140-143°C (285-290°F), and the impurities in the cover gas are about 10-12 ppm hydrogen and 7000 ppm nitrogen (oxygen and helium concentrations are negligible).

Following the maintenance shutdown in May 1982, a gradually increasing concentration of hydrogen was noticed in the cover gas. Additionally, the plugging temperature of the primary sodium was increasing. Initial attempts to identify the possible impurity and its source were complicated by the fact that the methane concentration in the cover gas was higher-than-normal in May and June 1982 and that carbon deposits had been found in one of the cover gas sampling system vapor traps. Based on the preliminary evidence, the probable sources were thought to be either oil or grease inleakage or alcohol contamination. The leak path was thought to be the No. 2 primary pump which had been refurbished during the May 1982 shutdown.

The No. 2 primary pump was absolved as a source of oil by disassembly and inspection of the possible leak paths of oil (all of which were outside the primary tank) in October 1982. Oil or grease were also discounted as the impurity after the methane concentration returned to normal levels coincident with continually high hydrogen concentration in the cover gas and slowly, but continuously, increasing sodium plugging temperature.

By January 1983, the primary impurity suspects had been reduced to either water or alcohol. The source of the impurity was still thought to be the primary pump which had been removed for maintenance; however, the mechanism for the storage and release of the impurity was still not understood. The leading theories were: (a) the shield plug structure for the primary pump was cracked and some alcohol had seeped into the shield plug while it was being washed in May 1982, or (b) there was a substantial amount of "free" water which had not been vented from the shield plug during the time it was filled with concrete and the concrete cured, and this water was being released into the sodium or cover gas through a crack which had formed during handling of the pump and shield plug in May 1982. Theory (b) was further supported by the fact that over 75 l (20 gal) of water had been evacuated from the shield plug of the No. 1 primary pump as a test in the early 1970's.

During the first two weeks of February 1983, the rate of increase of the plugging temperature increased substantially and the plugging temperature reached 171°C (339°F) by mid-February. The "free water" theory had been accepted as the most likely source of the impurity by this time. It was also believed that by core drilling several holes in the concrete fill of the No. 2 pump shield plug, communication between voids in the shield plug could be improved to the point where vacuum distillation could be used to de-water the shield plug.

The reactor was shut down on February 16, the upper components of the No. 2 primary pump were removed, and two 50-mm diameter holes were drilled into the concrete biological shielding of the shield plug. Water was found on top of the concrete shielding when the fill plugs were removed to allow core drilling. More water was removed from the shield plug as soon as vacuum distillation was begun. By the time the No. 2 primary pump was reassembled and the reactor had been started up, over 19 l (5 gal) of water had been removed from the shield plug. Even more exciting was the fact that traces of alcohol and long-lived gaseous fission products were found in the distillate. This was the final bit of information necessary to confirm the shield plug "free water" as the impurity in the cover gas.

The hydrogen concentration decreased quite rapidly from about 280 ppm to 70 ppm soon after the evacuation of the shield plug was started. By the first of April 1983, about 96.2 l (25.5 gal) of water had been evacuated from the shield plug. By the first part of May 1983, plugging temperature had been reduced to normal by the primary cold trap and the hydrogen concentration in the cover gas was almost normal. It has been estimated that only 3.8-5.7 l (1-1.5 gal) of water actually leaked into the primary system. When one considers that the primary system contains close to 340 x 10^3 l (90,000 gal) of sodium, the significance of any potential source of impurity in a sodium system can be put into proper perspective. Because of the sensitivity of the primary sodium to occurrences of this nature, the other shield plugs in the primary system which could cause similar incidents are being modified so that vacuum distillation can be routinely performed on them. Additionally, a portable vacuum distillation "cart" will be built for routine use on the concrete-filled shield plugs.

RETRIEVAL OF SUBASSEMBLY X379*

Almost every operating facility has encountered events such as component malfunctions or failures, operator errors, engineering oversights, etc., which leaves one asking, "How could that have happened?", or "Why didn't we think about that? On November 29, 1982, EBR-II encountered just such an event.

Refueling the EBR-II core involves remote transfers of subassemblies between the in-tank subassembly storage basket and the core. Each transfer involves engaging and disengaging the subassembly with a manually operated fuel-transfer arm. The transfer arm is equipped with a manually operated sensing device which indicates proper engagement of the subassembly and a locking device which prevents disengagement of the subassembly. Following engagement of the subassembly, it is manually rotated in a horizontal plane between the core transfer point and the in-tank storage basket. This transfer occurs beneath the primary sodium and is controlled from the operating floor of the reactor building.

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*"EBR-II--Search for the Lost Subassembly," by R. W. King, H. W. Buschman, J. Poloncsik, J. S. Remsburg, and H. W. Sine, Argonne National Laboratory, presented at the 1983 Winter Meeting of the American Nuclear Society.

Since beginning operation in 1964, over 24,000 transfer operations had been completed. However, on November 29, 1982, subassembly X379 was being transferred from the storage basket to the core. Although proper indications of subassembly engagement and locking had been noted by the operator when the subassembly was removed from the storage basket, sensing devices on the core gripper indicated that the subassembly was not on the transfer arm when it reached the core transfer point. Subsequent checks confirmed that the subassembly had, in fact, dropped from the transfer arm during the transfer. This meant that the subassembly had dropped either to the bottom of the primary tank or onto the core face.

Parallel activities to develop, test, and fabricate search and retrieval tools, to locate the subassembly, and to develop a retrieval plan were started immediately. A systematic search plan using several probes, including one probe with two retractable arms, located the subassembly over the reactor vessel neutron shield and core face.

The orientation of the subassembly was determined by taking vertical and azimuthal profile measurements on the subassembly with a probe which was inserted through a nozzle opening which normally houses the main-core gripper assembly. The suspected orientation of the subassembly was verified and refined by trial-and-error positioning of a dummy subassembly in a partial, full-scale mockup of the reactor vessel neutron shield and core face.

The retrieval was considered to be feasible by using a simple wire-loop-type snare on the end of a probe. The snare was successfully demonstrated several times using the mockup. The actual retrieval was accomplished on the first attempt on December 28, 1982. Once the subassembly was secured in the snare, it was returned to the transfer arm and removed from the primary tank for inspection using routine fuel handling methods.

The cause of the incident was believed to be misalignment between the transfer arm and the in-tank storage basket. These components were realigned and thoroughly checked out before routine fuel handling was resumed. In addition, the fuel handling procedures were expanded to provide additional checks when the engagement of a subassembly is verified. In June 1983, when the transfer arm was removed from the primary tank for repairs, the mouth of the carrier block (i.e., the piece of the transfer arm which supports a subassembly) was chamfered to further preclude dropping another subassembly.

THE FUTURE OF EBR-II

In addition to the invaluable contributions EBR-II has made in the area of steady-state and transient fuels testing, the safe and efficient operation of EBR-II during the last 19 years represents a significant portion of the existing U.S. LMFBR operating experience. It is expected that EBR-II will continue to contribute to this experience base for at least another 10 years since there is at present no apparent limitation due to any single component, plant system, or any other identified situations which would limit the plant life to less than 30 years.

The present program of Operational Reliability Testing is scheduled to continue through 1986. Other missions which are being considered for the EBR-II reactor include testing of major system components (i.e., evaporators or superheaters) of new designs or new materials, continued development of metal fuels with emphasis on possible adaptation of the core and other on-site reprocessing and refabrication facilities for use with advanced fuels such as U-Pu-Zr and possibly Th-U-Pu-Zr alloys, or perhaps just extended operation to demonstrate the capabilities of LMFBR's, in general, and the capabilities of the pool-type LMFBR's, specifically.