

OPERATIONAL-SAFETY-TESTING EXPERIENCE AT EBR-II

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## OPERATIONAL-SAFETY-TESTING EXPERIENCE AT EBR-II

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### ABSTRACT

Operation of EBR-II has required that issues of safety and reliability be continually addressed. This has been necessary not only to support routine operation, but also to support test programs that bear upon safety. Major issues are (1) operability of fuel elements with breached cladding (local fault prevention), (2) assurance of adequate convective cooling upon loss of pumping power (decay heat removal), and (3) demonstration of benign response of fuel-elements and subassemblies to protected overpower transients (shutdown system margin and reliability. Test programs to address these issues at EBR-II are described. Particular emphasis is given to the potential capability at EBR-II for mild transient testing, consistent with its role in addressing LOA-1 (line of assurance one) safety issues.

### INTRODUCTION

Operational-safety testing deals with two major areas, operating reliability and reactor safety. The major concern for public safety is accidental release of fuel and fission products. Release that would be the most damaging would occur suddenly and in large quantity. Therefore, safety research has concentrated on eliminating major accidents.

Increasingly, emphasis is being placed on features intended to prevent minor accidents that have the potential for disrupting the core [1]. This has been necessary for a number of reasons. First, to prevent major accidents it is important to prevent minor ones. Second, there is increased awareness of the need to avoid those accidents which would increase public apprehension over nuclear power, even though there is no significant fission product release. Third, there is the desire to limit the economic impact of damaged components. A payoff of such work is that, properly done, it can also contribute to improved operating reliability and availability of reactor systems.

Safety research for prevention of accidents may be categorized as follows:

1. reactor and plant designs to reduce the probability of an accident,
2. provision of systems to reliably shut the reactor down, and
3. provision of systems to ensure removal of decay heat.

Especially for the first category, one is dealing with issues of operating reliability as well as safety.

The EBR-II experience and future role in addressing operational-safety testing may be divided into four somewhat independent parts. The first is testing fuel elements with breached cladding, both in open core positions [2] and in special in-core test facilities [3]. This program will be extended to include more severely breached elements to address issues of local fault development.

The second is whole-plant heat-transport system testing directed to questions of decay heat removal. This program is approximately two-thirds complete with the first series of tests conducted in 1973-74 [4,5] and the second series in 1978-79 [6]. Further tests will more closely simulate accident conditions and will include both fuel and structural instrumented subassemblies in-core.

The third is testing to determine fuel-element and shutdown-system response to plant transients. Much data has already been obtained from loss of pumping power events [6,7], and the effects of plant startup and shutdown on breached elements have been observed [2]. This experience will be extended by providing a mild transient-overpower capability and "programming" such transients so that data can be more reliably obtained.

The fourth area, largely a fallout of EBR-II operation, is the collection of operating and maintenance data to improve the operator/plant interface and reliability of operation. Past experience has demonstrated: the feasibility of routine operation, cleaning, and repair of components operating in sodium; operation of core components to high fluences without excessive damage; the feasibility of design and operation of systems to clean sodium of oxygen and cesium contamination, and cover gas of Xe and Kr contamination; the reliable operation of sodium-to-water heat exchangers; the development of the first Technical Specifications document for a LMFBR; the applicability of reactor diagnostic techniques, i.e., reactivity meters, drop rods, in-core active and passive instrumentation; operation of a sodium-cooled system at high plant factor with new zero environmental impact; and the feasibility of failed fuel element detection and location using gas tagging methods. A more complete listing of operational-safety related accomplishments is shown in Table I.

#### SAFETY ASPECTS OF OPERATION WITH BREACHED ELEMENTS

The run-beyond-cladding-breach (RBCB) testing program at EBR-II has the objective of defining the condition at which a breached element is no longer "benign". The major aspects by which benign operation is judged are (1) potential for primary circuit contamination, (2) potential for slow propagation of breach from element to element and, (3) potential for

TABLE I

## Significant EBR-II Accomplishment Operation-Safety

1. Fourteen years of successful experience with the sodium-to-water steam generator (evaporators and superheaters) with no leakage of water into sodium.
2. Fourteen years of experience with under-sodium components such as pumps, flowmeters, intermediate heat exchanger, fuel handling equipment, etc., with only minor and repairable problems.
3. Achievement of annual plant capacity factors that compare favorably with the best performance of commercial power plants--while operating as the nation's only fast-flux irradiation facility. Plant capacity factor averaged 68.3% over the past four calendar years, with a high value of 76.9% in 1976.
4. Extensive experience in maintenance of sodium components; practical demonstration that maintenance and repair of such components can be accomplished by straightforward techniques, with relatively simple equipment, and without undue hazard to personnel.
5. Operation of the plant with minimal release of radioactivity to the environment.
6. Replacement of a portion of the EBR-II radial uranium blanket with a stainless steel reflector to enhance the core environment for experimental irradiations.
7. Irradiation of 564 experimental subassemblies through run 92 (May 1965 through January 1978). These contained 9430 individual experiments, including 2756 mixed-oxide fuels, 525 mixed-carbide or -nitride fuels, 4500 metal driver fuels, 1144 cladding and structural materials, 199 absorber materials, and 306 miscellaneous specimens.
8. Improvement of EBR-II metallic driver fuel element design and increase of fuel burnup limit for ANL-produced fuel (Mark-II) from 1 at. % to 8 at. %.
9. Irradiation of mixed-oxide fuel to a heavy atom burnup of 19 at. %. Attainment of fuel element cladding temperature of 1500°F (815°C). Accumulated neutron fluence of  $1.7 \times 10^{23}$  nvt on structural material specimens.
10. Design, installation, and operation of the following experimental facilities: instrumented subassemblies (INSAT); instrumented in-core facilities (INCOT); the radioactive sodium chemistry loop (RSCL) for testing sodium-quality-monitoring instruments in primary sodium; the nuclear instrument test facility (NITF) for testing nuclear instrument sensors and cabling; and systems for testing fission-product detection instruments.
11. Irradiation of over 200 individual experiments (capsules, creep specimens, fuel elements, sensors, etc.) in instrumented in-core facilities.

TABLE I (Contd.)

12. Development and demonstration of a system of hydrogen-meter leak detectors to monitor for leakage between water and sodium systems in the EBR-II steam generator.
13. Development, demonstration, and beneficial utilization of failed-fuel element detection and location (FEDAL) techniques and equipment, including delayed-neutron monitoring in the primary sodium coolant.
14. Development, demonstration, and beneficial utilization of a failed-fuel location technique based on unique mixtures of xenon isotopes ("xenon tags") for rapidly identifying the sources of fission-product releases.
15. Design, construction, and operation of the cover-gas-cleanup system (CGCS), incorporating cryogenic distillation to remove and concentrate rare-gas fission products from the cover gas.
16. Development, prove-out, and routine use of a versatile system of computer programs for thermal-hydraulic-nuclear considerations; modeling of whole-plant thermal-hydraulic behavior; conduct of convective-flow tests in EBR-II, utilizing INSAT and INCOT facilities.
17. Training and information services for personnel of other national breeder program organizations and of the academic nuclear community.
18. Fast reactor familiarization training for FTR operating personnel.
19. Investigation, monitoring, and explanation of thermal- and fluence-related effects in fast reactor core hardware; e.g., subassembly bowing and swelling.
20. Development and application of techniques and equipment to characterize and monitor fast reactor kinetic behavior.
21. Identification and investigation of major safety and availability issues surrounding sodium-cooled fast reactor operation with breached fuel elements; conduct of a vigorous run-beyond-cladding-breach testing program.
22. Development and defense of safety philosophy and documentation supporting operation of sodium-cooled fast reactors, resulting from safety-related experimental programs and modifications in EBR-II; development of the first set of technical specifications for a fast reactor power plant in the U.S.
23. Development, testing, and application of novel sodium-sampling and impurity-measuring equipment to monitor and control sodium purity in the EBR-II systems; characterization of cold-trap performance in the practical plant environment; monitoring of fission-product tritium transport; development of a trap for  $^{137}\text{Cs}$  released to the sodium coolant from breached fuel elements.

TABLE I (Contd.)

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24. Development, testing, and application of diagnostic techniques and instrumentation for characterization of sodium-cooled fast reactor operating conditions (reactivity meters, thermal-expansion difference thermometers, gamma expansion difference monitors, self-powered neutron detectors, pulsed-neutron activation flow measurement, etc.).

aggravation of the breach site by mild overpower operational transients. In terms of test objectives, these may be recast into the issues of survivability, detectability and coolability as shown in Table II; the first two have been addressed by testing [2] and the last by analysis [8].

Irradiation of breached fuel elements at EBR-II has revealed that changes in power and flow incident to routine reactor operation can severely impact the lifetime of breached oxide elements. A common experience is that, so long as the reactor power is held constant, little change in breach condition occurs. Shutdown with subsequent startup will generally reveal an increased delayed neutron signal from an enlarged breach site. It seems likely that ingress and egress of sodium from the fuel element is a major contributing factor. Sodium drawn into the element will react chemically with mixed oxide fuel upon reaching operating temperatures. (The reaction product has a lower density than the fuel and will stress the cladding.) Unreacted sodium that is trapped in the element may be forced out under pressure, enlarging the breach site. Fuel-clad mechanical interaction on reactor restart and thermal stress may also contribute to breach-site enlargement.

The significance of this experience is that safety-related aspects of breached element operation may be dominated by performance on reactor transients. Reactor transients in this context include those incident to normal reactor operation (startup, shutdown, scram), and protected reactor faults. Subjecting breached fuel elements to programmed transients would provide information on (1) the likelihood of gas leakers to "open up" and expose fuel, (2) the increase in area of exposed fuel, and (3) potential for slow propagation of breach. The goal of such testing is to provide element designs that, if breached, will remain as gas leakers for extended operation without significant exposure of fuel to the coolant.

TABLE II

Tests Comprising a Safety/Reliability Case for RBCB Operation

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Survivability

1. Show that breached elements can survive extended steady-state operation without exposing fuel.
2. Show that breached elements can survive normal startup, shutdown and change in power without enlarging breach site.

TABLE II (contd.)

3. Show that fuel and fission-product loss from an enlarged breach site does not lead to serious contamination.

#### Detectability

1. Show that breached elements can be detected and identified early.
2. Show that delayed-neutron detection can bound the area of exposed fuel, and that shutdown limits can be established.
3. Show that if fuel (and cladding) collects to form a flow blockage, it can be detected and shutdown accomplished before extensive propagation.

#### Coolability

1. Show that if shutdown is not accomplished, a flow blockage of fuel and cladding will not fail all elements in a subassembly.
2. Show that enlarged breached sites can survive a mild upset (protected TOP and TUC) without leading to flow blockage.

#### RESPONSE OF CORE AND PROTECTION SYSTEMS TO PLANT TRANSIENTS

The EBR-II has experienced a large number of plant transients, both planned and unplanned [7]. Much useful data has been obtained, and it is planned to develop the capability to undertake more severe transients in the future program of operational-safety testing. The objective of this program is to verify protective margins and shutdown-reliability. This requires a capability to simulate anticipated overpower transients, rapid startup, power cycling, reactor shutdown and loss of forced cooling.

In the process of reactor licensing, it is necessary to identify those transients requiring protection. When setpoints for reactor trip are established, it is also necessary to demonstrate that fuel will not fail for conditions at the trip setpoints and that adequate margin exists to allow for overshoot in power and uncertainty in prediction of temperature. In principle, this can be demonstrated either by subjecting elements to single transients severe enough to cause failure, or to repeated transients at the predicted limits of power and temperature.

Because most "anticipated" transients occur from a condition of partial or full power, initial fuel conditions must be representative of conditions obtained at partial or full power. This is especially true of breached oxide elements as discussed previously. For unbreached elements the fuel will crack on cooldown and for a mild power transient, may load the cladding differently than would "annealed" fuel.

Indeed, the failure mechanism for fuel elements may be quite different for repeated mild transients initiated at a condition of partial or full power than for single, severe transients initiated from an initially cold condition.

A minimum transient capability has been established and preliminary analysis has shown that plant capability limits are satisfied.

#### Minimum Transient Capability

Mild TOP to 130% of Nominal Power;  
Maximum Rate of Power Increase  $\sim 12$  %/s;  
Maximum Reactivity Insertion Rate 10 c/s.

Establishment of a minimum transient capability is related to typical setpoints of a plant protection system. Generally, protective systems for overpower protection have setpoints in the range of 15 to 25% overpower. The maximum reactivity insertion rate is defined by operational transients utilized in design basis analysis. For anticipated protected transients, more energy is deposited at slow insertion rates than at higher insertion rates ( $< 1$  \$/s) associated with hypothetical insertion events. This implies that slow ramps may be more damaging.

Locally higher power increases are possible by modifications to the subassembly environment (i.e., shifting subassembly position, increased enrichment, flux-tailoring).

A typical testing sequence would include a series of cyclic transients simulating slow power increases to just below typical plant protection system setpoints and then returning to nominal power levels. After a predetermined number of cycles have been completed, a protected overpower transient would be conducted.

Subassembly testing will be undertaken primarily in open-core positions to reduce cost and maximize the quantity of data obtained. Two open-loop test facilities are being considered for those experiments in which significant failure is expected. These test facilities would be similar to the breached-fuel-test-facility (BFTF) [3] being installed for the run-beyond-cladding-breach (RBCB) program. They will provide flow and temperature monitoring, flow control, fission product/fuel containment and delayed neutron monitoring.

Protected transient overpower testing in EBR-II will provide a link between steady-state element/component testing in EBR-II and FTR, and the more severe transient testing provided by TREAT and SLSF. Such testing would support long-range FBR design and licensing efforts and aid in demonstrating the insignificant risk to the public from operating fast reactors.

#### WHOLE-PLANT HEAT TRANSPORT SYSTEM TESTING FOR DECAY HEAT REMOVAL

The program of decay heat removal testing proposed for EBR-II is an extension of testing accomplished with instrumented subassemblies XX07 [4] and XX08 [5]. These were driver-fuel subassemblies instrumented to measure flow, coolant and fuel temperature. A similar subassembly, designated XX09, is proposed for the follow-on program. A unique feature of this phase is that a structural subassembly instrumented for coolant flow and temperature (XX10) will also be included.

This program offers to extend data from the previous tests in three ways. First, more severe plant transients can be simulated. This is important to reduce the amount of extrapolation that must be made to qualify safety codes predicting response to the most severe events. (Dynamic transition into natural convective flow can lead to high temperatures for short times, and is complicated.) Second, only with instrumented subassemblies of two different types, can basic information be obtained for flow redistribution in the reactor core. Third, with improved plant instrumentation, more data can be obtained for whole-plant transient response.

The basic objective of this testing program is to better qualify safety codes to predict plant thermal-hydraulic response to a range of upset events (pipe break, loss of power, sodium-water interaction, etc.). At issue are the basic phenomena that dominate dynamic response. For example, it has been found that radial heat transfer between subassemblies, heat loss from downstream piping, thermal stratification in large sodium pipes, and transition from turbulent to laminar flow are all important factors which must be accurately modeled in dynamic codes. Better understanding of these factors will assist in plant design, and when a design is available, specific predictions can be made. Operation of a real plant must involve some convective flow and dynamic testing to check the model, but no large plant can be subjected to transients severe enough to fully simulate response under accident conditions. We are attempting to approach this capability at EBR-II.

#### CONCLUSION

Much has been learned from EBR-II operation and testing to better ensure safe and reliable operation of LMFBR plants. When the FTR is operational, EBR-II will be able to accept higher risk operation to more clearly address operational safety issues. Table III shows the major objectives of a coordinated operational-safety test program. It is seen to be a three-by-three matrix. Across the top are the three groups to whom information will flow: the fuel element and subassembly designers, the plant and system designers, and the safety/licensing analysts. Each must communicate with the other for a successful design. Likewise, down the left column are shown the three major elements of the operational-safety program. Each provides data for input to the other.