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ABSTRACT

The mission of the Experimental Breeder Reactor II (EBR-II) has evolved from that of a small LMFBR demonstration plant to a major irradiation-test facility. Because of that evolution, many operational-safety issues have been encountered. This paper describes the EBR-II operational-safety experience in four areas: protection-system design, safety-document preparation, tests of off-normal reactor conditions, and tests of elements with breached cladding.

INTRODUCTION

The Experimental Breeder Reactor II (EBR-II) has been in operation since 1964, during which time it has evolved from a demonstration plant to a major irradiation facility. (A chronology of significant events is listed in Table 1.) That evolution has required considerable engineering modification to the plant as well as safety analysis to support test programs bearing on current safety issues. Emphasis has been given to ensuring that both the designer and operator understand safety margins, that the operator is provided information necessary to take emergency action, and that the best balance has been struck between system design for safety and system design for reactor availability. Such issues are relevant to an area termed operational safety.

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EBR-II experience directly related to safety may be divided into four major categories: (1) modification of the plant protection system[2], (2) preparation of limits to reactor operation, i.e. Technical Specifications[3], (3) testing of system and instrument response to off-normal conditions of flow and power[4], and (4) testing of oxide, carbide and metallic elements with breached cladding[5,6]. In addition, much safety-related analysis and testing has been conducted to assure continued reliable operation. Examples include analysis of radiation damage to in-reactor components[7,8,9], increase in the burnup limit of driver fuel[10], installation of a stainless steel reflector, testing and evaluation of steam generator performance, and development of a blanket subassembly management plan[30].

If a general characterization could be made of this experience, it would be that EBR-II is addressing a wide range of operational-safety questions appropriate to design and operation of commercial Liquid Metal Fast Breeder Reactors (LMFBR's). Two points of emphasis can be identified in early research: (i) steady-state irradiation testing of fuels and materials and (ii) testing and analysis to resolve issues associated with the hypothetical core-disruptive accident (HCDA). Steady-state testing of fuels and materials is evolving to include more attention to mild off-normal conditions, i.e. performance of breached elements, mechanical interaction between fuel elements, and response of fuel elements to mild operational transients. Research addressing the HCDA is expanding to include more emphasis upon risk-assessment and whole-plant dynamic testing, largely as a result of the "Line of Assurance" approach to safety research[11]. The significance of this evolution is that aspects of operational safety, i.e. means of preventing accidents and accommodating mild off-normal conditions, are gaining increased attention. This paper summarizes the most important results accomplished to date in EBR-II and identifies issues for future investigation.

PLANT PROTECTION SYSTEM MODIFICATIONS

Safety analysis supporting modification to the PPS has covered a wide range of safety issues. Following review and approval, more than half of the trip functions in the original Plant Protection System (PPS) were removed and converted to alarms[12] (see Table II). Most of the remainder were upgraded to improve their effectiveness. What has resulted is a simpler, more reliable and more respected PPS[2]. During this period, the "plant factor"* has steadily increased to a high of ∿ 75% in 1975. This increase can be attributed in part to a reduction in the number of spurious reactor trips from PPS instrumentation.

The goal was to identify those automatic trip functions required while maintaining as simple a protection system as possible. An example is the removal of a reactor trip function on high or increasing delayed-neutron signal (from fission products in the primary coolant). The EBR-II position was taken after it was concluded that no mechanism exists for rapid propagative failure between elements under steady-state irradiation and that for breached elements, the breach site will develop slowly enough to allow time for operator action well before significant fuel washout[6,13]. Results from subsequent testing have supported this position. The safety issue for delayed-neutron monitoring has now become one of establishing the proper alarm point for operator action.

PREPARATION OF BASIC SAFETY DOCUMENTS AND LIMITS TO OPERATION

The basic safety documents by which construction and operation of EBR-II was authorized were prepared in the late 1950's and early 1960's[14]. In 1970, the EBR-II Project undertook a major updating of safety analysis, culminating in a document defining safe limits to operation (EBR-II Technical Specifications)[3].

Of primary interest in establishing safety limits is the margin available should a reactor fault occur. To evaluate available margin requires understanding of the failure mode of fuel elements and the potential for failures to propagate between fuel elements or across a subassembly duct.

*Defined as the ratio of total annual electric output to potential output from uninterrupted operation.

One question that was investigated was the effect of rapid fission-gas release from breached fuel elements upon an over-power or under-cooling transient. When a large cladding breach forms suddenly, a pressure pulse is felt by adjacent pins and the hexagonal duct. The basic safety issue at EBR-II was whether an assumed large pressure pulse could cause sufficient deformation or cracking of hexagonal cans adjacent to control rods to hinder their motion.

In order to characterize the response of irradiated hexagonal ducts to pressure pulse, a series of tests was performed[15,16]. These were of particular interest because, unlike previous tests, they utilized irradiated ducts (for which fracture is of greater concern) and were conducted in sodium at near prototypic conditions. These tests demonstrated that irradiated hexagonal cans are capable of withstanding pressure-pulses of significantly higher magnitude than those associated with either normal or off-normal conditions at EBR-II. Consequently, damage from pressure pulse is no longer of major concern.

RESPONSE TO OFF-NORMAL CONDITIONS OF POWER AND FLOW

EBR-II experience with total system response to off-normal conditions has been primarily associated with loss-of-pumping power events. Considerable attention has been given to transition into convective flow upon loss of all primary pumping power[3,17]. The results of tests conducted to address this area has lead to the following general conclusions regarding convective flow behavior at EBR-II:

1. Prediction of dynamic transition into convective flow requires system tests for accurate modeling.

2. Radial heat transfer and flow re-distribution in-core is effective in reducing locally high temperature; different hot spot factors should be employed for conditions of low flow and power than for rated conditions.

3. Heat loss from downstream primary piping can be significant under convective flow conditions and tends to reduce the thermal driving head.

4. A major contribution to convective flow is the driving head of the intermediate heat exchanger. To adequately model response of the primary system to loss of pumping power, response of the balance of plant must also be characterized.

There is presently a test program underway to provide more definitive data of response of EBR-II to loss of pumping power. These tests involve tripping of the primary pumps coincident with reactor scram without the auxiliary pump in operation. These data should be of sufficient quality to allow the accuracy of a number of available modeling codes to be checked.

The goal for system design is to avoid sodium boiling following loss of power. However, as has been suggested by tests in DFR[18], sodium boiling under conditions of low heat flux is stable and will likely not result in fuel failure. The onset of boiling with decay heat and natural convective cooling represents an upper bound on system temperature primarily because of increased heat capacity and increased convective driving head. Tests to investigate coolant boiling at low heat flux could be important in demonstrating safety margin on loss of forced flow.

RUN-BEYOND-CLAD-BREACH TESTING

A program of run-beyond-clad-breach testing was initiated in EBR-II in 1977. The major objective of this program is to establish the basis upon which continued operation of a commercial reactor with breached fuel can be judged. The task becomes one of defining the point at which a breached element is no longer "benign."

The three major concerns by which benign operation is judged are:

a. potential for primary circuit contamination,

b. potential for propagative element-to-element failure,

c. performance of breached elements on mild operational transients.

The first concern, primary circuit contamination, can have serious impact on system maintenance. Fuel elements are being tested to determine potential for fuel and fission product release, to correlate the observed delayed-neutron signal associated with such conditions, to define the mechanism by which contamination is transported in the primary system, to devise means for detecting contamination, and to devise means of sodium cleanup. A notable success to date is the verification that ^{137}Cs contamination can be effectively controlled by a graphite nuclide trap. Data from delayed-neutron detectors have shown that a rough correlation between exposed fuel area and delayed-neutron signal does indeed exist.

A particularly difficult problem, once the above questions have been resolved, is identifying an offending element for removal once it has been determined that the reactor must be shut down. In spite of the good experience at EBR-II, gas-tagging of individual elements may not be viable in a commercial system, particularly if the number of gas leakers is high. Another problem is that advanced fuels, notably carbides, may not give up their tag gas easily because of low fission gas release and sodium bonding. Emphasis must shift to using delayed-neutron (DN) techniques or individual subassembly interrogation for gas release, once shutdown has been accomplished. A major question is whether some method of sodium sampling for DN monitoring of individual subassemblies is required.

The second concern, potential for serious propagative element-to-element failure, appears to be real only after serious primary circuit contamination would have already occurred. Therefore, it must be established that a DN signal from a potentially damaging flow blockage is much greater than the DN signal causing shutdown to avoid serious circuit contamination. This will require some flow-blockage simulation tests by which to compare the magnitude of DN signal.

The third concern, performance of breached elements on mild operational transients, is an important one. Such an element may behave benignly under steady-state condition, but may seriously contaminate the sodium on an operational transient. Likewise, the evidence from EBR-II tests on mixed-oxide fuel is that changes in power with load-following may seriously aggravate the breach site and result in contamination that would not occur during steady-state operation. Emphasis for RBCB testing must at some time shift to load-following and operational transients. It appears that this mode of operation provides the most serious constraint on survival and operability of breached elements.

To provide the capability for more extensive RBCB testing at EBR-II, a special test facility is being designed. Basically, it is a reusable device that can be placed over the top of a subassembly. Sodium from

The subassembly is directed through a thimble in a converted control rod location. Sodium is then directed to an elevation corresponding to the underside of the small rotating plug and then returned to the upper plenum of the reactor by downflow through an annulus between the central tube and the outer wall of the facility. Sodium from the subassembly is monitored for temperature, flow rate, and delayed-neutron precursor content. A removable deposition sampler is included which will measure the transport and deposition of fission products and fuel particles.

A study has recently been undertaken to evaluate capability of EBR-II to provide mild transient testing, of the type associated with anticipated operation of commercial reactors. Such transients include load-following and, in the extreme, simulation of protected over-power transients (to 130% of nominal power). Such testing could be useful in evaluating the performance of breached elements as well as studying such effects as fuel-clad mechanical interaction.

DEVELOPMENT AND USE OF DIAGNOSTIC INSTRUMENTATION

To monitor reactor and core conditions, a number of measurement techniques have been employed. An anomalous reactivity meter has been developed and used successfully[19]. It has been useful in determining the cause of local reactivity perturbations[20,22], and in evaluating unusual long-term reactivity changes[21]. It provides significant advantages over conventional monitoring and protection systems.

EBR-II is also instrumented with acoustical monitors which have been operated successfully[23]. To the present time, their use has been limited to measurement of background noise at EBR-II. These systems are potentially useful in detecting mechanical problems in the primary systems (such as the loose-fill tube in the intermediate heat exchanger in EBR-II), or for providing control for single-subassembly tests with potential for sodium boiling. (Sodium boiling tests have, in fact, been suggested for EBR-II[24].)

In conjunction with acoustical monitoring, accelerometers have been placed on important plant components to routinely interrogate vibration characteristics[25] for unusual changes. Using this system, in conjunction with reactivity meters, the source of a 10-Hz oscillation was traced to vibrating control rods[26]. The primary pumps have been found to change their vibration characteristics when lubrication in the main bearings is low.

Present efforts in the area of acoustical/vibration monitoring are to develop the diagnostic system best suited for routine use in an operating plant. Answers are being sought to the question of extent of spectral analysis required, proper alarm functions to the operator, best equipment designs, and proper use of data for maintenance or interrogation of off-normal conditions.

Characterization of the core environment has been an important feature of EBR-II operation and analysis. For this purpose, self-powered-neutron detectors have been developed and tested to monitor local flux levels[27]; thermal expansion difference monitors have been used to measure in-core temperatures; gamma expansion-difference monitors have been developed for measurement of local gamma heating[28]; extensive neutron-dosimetry measurements have been made, and in-core fuel and structural experiments have been instrumented with thermocouples and flowmeters[29].

In summary, the EBR-II experience in safety analysis has come from applying modern concepts and analysis to an operating LMFBR plant. Where appropriate, as in convective flow modeling or in run-beyond-clad-breach testing, specific tests are being conducted and planned. When operational problems arise, they must be resolved. An important point from the EBR-II experience is that any system design must provide opportunity for early identification of problems and sufficient flexibility to provide solutions.

REFERENCES

1. R. R. Smith, W. B. Loewenstein, C. M. Walter, "The EBR-II: A Status Report," ANL-7743 (July 1971).

2. J. I. Sackett, "Safety Philosophy in Upgrading the EBR-II PPS," Proceedings of International Meeting on Fast Reactor Safety and Related Physics, Chicago, Ill., pp. 605-614 (Oct. 1976).

3. J. I. Sackett and N. L. Gale, "Development of Technical Specifications for EBR-II: Some Considerations for Application to Commercial LMFBR's," Proceedings of International Meeting on Fast Reactor Safety and Related Physics, Chicago, Ill. pp. 411-419 (Oct. 1976).

4. G. H. Golden, J. I. Sackett, R. M. Singer, "Tests and Analyses of Normal and Off-normal Operating Conditions in EBR-II," Trans. Am. Nucl. Soc., <u>22</u> 588 (1975).

5. J. I. Sackett et al., "Preparations at EBR-II for Breached Element Operation," Trans. Am. Nucl. Soc., 27 (1977).

6. R. M. Fryer, J. I. Sackett, J. D. B. Lambert," Safety-related Considerations for Operation with Defected Elements in EBR-II," Proceedings of International Meeting on Fast Reactor Safety and Related Physics, Chicago, Ill., pp. 615-624. (Oct. 1976).

7. R. D. Phipps, "A Study of the Deformation Produced by Long Irradiation of EBR-II Control and Safety Rod Thimbles," thesis presented in partial fulfillment of the requirements for the degree of Master of Science, Major in Metallurgy, University of Idaho Graduate School, April 1973.

8. L. K. Chang and A. H. Marchertas, "Stress Analysis of a Nuclear Reactor Grid Plenum Assembly and Its Application to the EBR-II," Nuc. Engr. and Design 48, 387 (May 1977).

9. J. H. Bottcher and D. L. Johnson, "Limits on Swelling for Experimental Subassemblies," in Reactor Development Program Progress Report, July-August 1975, ANL-RDP-42, October 7, 1975).

10. B. R. Seidel and R. E. Einziger, "In-reactor Cladding Breach of EBR-II Driver Fuel Elements," Radiation Effects in Breeder Reactor Structural Materials, M. L. Bleiberg and J. W. Bennett, eds., TMS-AIME, New York, 277 (1977).

11. F. X. Gavigan and J. D. Griffith, "The Application of Probabilistic Methods to Safety R & D and Design Choices," Proceedings, International Conference on Nuclear Systems Reliability Engineering and Risk Assessment, Gatlinburg, Tennessee, pp. 22-41 (June 1977).

12. N. L. Gale, "Upgrading the EBR-II Reactor Shutdown System," Trans. Am. Nucl. Soc., 27 (1977).

13. R. M. Fryer, E. M. Dean, J. D. B. Lambert, and L. A. Reyes, "Efficacy of the EBR-II FERD System As An Automatic Trip Device," ANL-76-94, (1977).

14. L. J. Koch, et al., "Hazard Summary Report - Experimental Breeder Reactor II (EBR-II), ANL-4719 (and addendum).

15. W. E. Ruther, P. S. Chopra, and J. D. B. Lambert, "Proof Tests of Irradiated and Unirradiated EBR-II Subassembly Ducts," Fourth SMIRT Conference, San Francisco, CA (August 1977).

16. W. E. Ruther, K. J. Longua, J. A. Shields, and F. A. Clark, "Investigations of Materials Compatability Relevant to the EBR-II System -- FY 1976," ANL-76-97 (January 1977).

17. R. M. Singer, et al., "Steady-state Natural Circulation Performance of the EBR-II Primary Heat Transport Circuit," N.S. & E., 63, pp. 75-82.

18. D. C. B. Smith, et al., "DFR Special Experiments," International Symposium on Design, Construction and Operating Experience of Demonstration LMFBR's," IAEA-SM-225/49, Bologna, Italy, 10-14 April, 1978.

19. H. A. Larson and J. I. Sackett, "An Anomalous Reactivity Meter," Nucl. Tech. Vol. 33, p. 2[illegible]3 (April 1977).

20. J. I. Sackett, "Detection and Analysis of Small Reactivity Perturbations in EBR-II Associated with An Experimental Subassembly Irradiation," Trans. ANS, 17 (1973).

21. L. C. Walters, et al., "Investigation of Reactivity Anomalies in EBR-II," ANL-76-27 (June 1976).

22. E. M. Dean and J. I. Sackett, "Response of EBR-II and the Plant Protection System to Off-Normal Primary Flow," Trans. ANS, 22 (1975).

23. W. M. Carey, et al., ANL-CT-77-29 (Dec. 1977).

24. S. A. Wright et al., "Designs for In-Reactor Sodium Boiling Detection and Generation," Proceedings, International Meeting on Fast Reactor Safety and Related Physics, Chicago, Illinois, CONF-761001, pp. 635-645 (Oct. 1976).

25. C. C. Price and J. R. Karvinen, "EBR-II Noise Signature Analysis During the First Half of Fiscal Year 1971," ANL/EBR-036 (1971).

26. C. C. Price et al., "Correlation of Mechanical Noise with Neutron Noise in EBR-II," Trans. ANS, 16 (1973).

27. C. C. Price and J. R. Karvinen, "Performance Testing of Self-Powered Detectors in EBR-II Instrumented Subassemblies XX01, XX02 and XX04," ANL-8030 (1973).

28. C. C. Price et al., "A Passive Calorimetry System for Measuring Gamma Energy Deposition Rates in An LMFBR," Trans. ANS, 22 (1975).

29. J. L. Gillette, et al., "Design and Fabrication of the EBR-II Environmental Subassembly: Test XX07," ANL-76-78 (1977).

30. J. F. Koenig, et al., "Effects of Power and Irradiation Behavior on EBR-II Blanket Management," Trans ANS, 27 (November 1977)

TABLE I

EVOLUTION OF EBR-II

- 1963 STARTUP AND OPERATION OF EBR-II AS A SMALL DEMONSTRATION POWER PLANT INCLUDING A CLOSED-FUEL CYCLE
- 1965 SUCCESSFUL DEMONSTRATION OF POWER OPERATION AND METALLIC FUEL REPROCESSING (30-DAY RECYCLE TIME FOR FUEL)
- 1969 CORE CONTAINED ~20 EXPERIMENTAL SUBASSEMBLIES. PLANT FACTOR (PERCENT OF TIME AT FULL POWER) WAS 42.4%.
- 1975 CORE CONTAINED ~50 EXPERIMENTAL SUBASSEMBLIES. PPS UPGRADING EFFORT COMPLETED. PLANT FACTOR WAS 66.1%.
- 1977 CORE CONTAINED A PEAK OF 65 EXPERIMENTAL SUBASSEMBLIES

  IMPLEMENTED A PROGRAM OF RUN-BEYOND-CLAD-BREACH TESTING

  INCREASED IRRADIATION OF ADVANCED FUEL (CARBIDE, NITRIDE)

  CONTINUED TO MODIFY PPS

  INSTITUTED A STUDY TO SUPPORT CONVERSION OF EBR-II TO A TRANSIENT TEST FACILITY

Table II. PPS Modifications at EBR-II

Trip Parameter Modified	Action
Reactor Power High	Nine detectors in the original system replaced by three wide-range detectors for protection in both fuel-handling and reactor-operate modes
Instrument-thimble-cooling temperature high	Deleted
Any control rod unlatched	Deleted
Reactor-coolant-flow low	Performance criteria established for the four flowmeters. An additional flowmeter qualified for PPS application
Reactor-outlet-coolant temperature-high	Deleted
Bulk-sodium-temperature high, level high, level low	Deleted
Primary pumps: winding-temperature high, clutch reference voltage low, clutch cooling-water pressure low	Deleted
Argon cover-gas temperature or pressure high	Deleted
Reactor trip on isolation of containment building	Deleted
Redundancy in shutdown string	A second shutdown string provided to ensure redundancy of instrument channels between common sensors and shutdown mechanisms
Delayed-neutron-signal high (FERD)	Deleted