Conf-901101--73

CONF-901101--73

DE91 006463

THE ROLES OF EBR-II AND TREAT IN ESTABLISHING LIQUID METAL REACTOR SAFETY*

J. I. Sackett, W. K. Lehto, and C. W. Solbrig

Argonne National Laboratory
P. O. Box 2526
Idaho Fall, ID 83403-2528

Submitted for Presentation
at the
American Nuclear Society

1990 Winter Meeting
November 11-15, 1990

Washington, D.C.

* Work supported by the U.S. Department of Energy, Reactor Systems, Development and Technology, under Contract No. W-31-109-ENG-38.

THE ROLES OF EBR-II AND TREAT IN ESTABLISHING LIQUID METAL REACTOR SAFETY

J. I. Sackett
Argonne National Laboratory
P.O. Box 2528
Idaho Falls, ID 83403
(208) 526-7671

W. K. Lehto
Argonne National Laboratory
P.O. Box 2528
Idaho Falls, ID 83403
(208) 526-7369

C. W. Solbrig
Argonne National Laboratory
P.O. Box 2528
Idaho Falls, ID 83403
(208) 526-7064

ABSTRACT

This paper examines the role of the Experimental Breeder Reactor II (EBR-II) and Transient Reactor Test (TREAT) facilities in contributing to the understanding and resolution of key safety issues in liquid metal reactor safety during the decade of the 80's. Fuels and materials testing has been carried out to address questions on fuels behavior during steady-state and upset conditions. In addition, EBR-II has conducted plant tests to demonstrate passive response to ATWS events and to develop control and diagnostic strategies for safe operation of advanced LMRs.

TREAT and EBR-II complement each other and between them provide a transient testing capability that covers the whole range of concerns during overpower conditions. EBR-II, with use of the special Automatic Control Rod Drive System, can generate power change rates that overlap the lower end of the TREAT capability.

1. INTRODUCTION

Testing programs to support LMR development and safety research have evolved in EBR-II from the initial mission to demonstrate an LMR power plant with integral fuel processing, to its current mission as the Integral Fast Reactor (IFR) prototype with its advanced integral fuels processing scheme. Following the initial demonstration from 1964 to 1968, EBR-II was modified to include an SS radial reflector to enhance irradiation capabilities for oxide fuel testing and materials development. In this mode, the fuel for both the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor Plant were developed.

The third phase of the EBR-II testing program began in the late 1970's and expanded conventional steady-state fuels testing to include tests of the response of the fuel and plant to off-normal conditions, such as breached cladding, anticipated operational transients and overall plant transients, such as loss-of-flow and loss-of-heat sink from full power without scram. 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. These programs have led to the designation of EBR-II as the IFR Prototype. Elements of this program include metal fuel development and demonstration of the integral fuel cycle, demonstration of the safety characteristics of the fuel and overall plant, demonstration of benign response to instrument and equipment malfunctions and demonstration of easier and enhanced safety regulation compliance.

The TREAT reactor is an air-cooled, thermal, heterogeneous test reactor designed to evaluate reactor fuels and structural materials under various transient situations. The design allows experiments to replace one or more of the 10.2 x 10.2 cm fuel elements in various parts of the core (usually the center). The TREAT core acts as a driver for the experiment. Very severe accidents, including those with fuel failure and consequent fuel coolant interaction, can be simulated.

The programs in EBR-II and TREAT compliment each other. EBR-II is capable of exploring all expected transients that could occur in the IFR. Because of the inherent safety of the plant design, extreme accident conditions cannot be duplicated. TREAT, however, can be used to expose fuel to extreme conditions so that the failure limits may be investigated and safety factors established. In addition, the fuel damage limits determined in TREAT can be used to assess other reactor designs for safety during upset conditions.

2. EBR-II AS THE IFR PROTOTYPE

The designation of EBR-II as the IFR prototype follows from the demonstrated fuel behavior (>18 at.% burnup),[1] passive safety features of the metallic fuel,[2] demonstrated thermal performance during upset conditions (loss-of-flow and loss-of-heat-sink without scram tests)[3] and soon to be demonstrated fuel reprocessing and the integral closed fuel cycle.[4] EBR-II is also the test facility designated to demonstrate self consumption of plutonium and actinides,[5] thereby greatly reducing the high level waste problem and reducing the time for monitored fuel storage by several orders of magnitude.

As mentioned above, the IFR will incorporate an integral fuel cycle where the spent fuel from EBR-II will be transferred to the fuel cycle facility for reprocessing and fabricated remotely into new assemblies and returned to the reactor. The U-Pu-Zr fuel is central to the process. The initial step of the processing occurs in the electrorefiner where the chopped fuel (cladding included) is electrolytically 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, plutonium, and actinides 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. A very similar fuel cycle was demonstrated in EBR-II during the first five years of reactor operation.[6] Over 700 irradiated assemblies of all types were processed. Of these, 560 were fuel bearing subassemblies, and these 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, along with the associated Fuel Cycle Facility, has come full circle and will serve as the prototype and test bed for IFR fuel cycle development and other IFR technologies.

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 the 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.

3. METAL FUEL DEVELOPMENT IN EBR-II

EBR-II has been employed as a test bed for LMR fuel irradiations since the late 1960's. This use had included developments of the reference mixed-oxide fuel, plus designs for the Fast Flux Test Facility and the Clinch River Breeder Reactor, as well as advanced fuels to promote breeding in the form of sodium-bonded and gas-bonded carbides and nitrides. The improvements that have been steadily made in the performance of sodium-bonded U-metal driver fuel of EBR-II, over more than 25 years of operation, have also laid thc basis for the fuel design employed in the IFR concept, which is now the focus for LMR development in the U.S.

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.%.[1] 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. It is these features that 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 further the understanding of IFR fuel performance during off-normal conditions. A key question is the fuel performance during steady-state operation following cladding breach. Run-Beyond-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.[7]

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.[8] 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,[9,10] 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 restrainer dimple and was due to stress rupture with very limited fuel cladding interaction. The breach location and failure mode agreed well with the pre-test 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.% burnup and the other at 10.2 at.%. 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.

4. PASSIVE SAFETY DEMONSTRATION

During the 1980s, a number of whole 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.[3,11,12] In each case the reactor was shutdown 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[13] was also conducted to demonstrate passive 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 P/F 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. This conclusion is also true for metal fuel LMRs.

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.[14,15]

5. TREAT

TREAT has contributed to LMR safety research since its initial test program in the early 1960's. The earliest tests were directed at investigating metal fuel behavior. These tests were used in support of EBR-II and FERMI. In recent years, LMR research in TREAT has again returned to investigating metal fuel--much improved over early designs--that will be used in the IFR. Prior work was directed at oxide fuels intended for use in the Clinch River Breeder Reactor Plant and FFTF.

TREAT is equipped with a fast neutron hodoscope capable of determining fuel motion in an experiment under severe accident conditions. Hence, not only is the failure limit determined, but the failure mode and degree of failure can be observed.

The test series in the early half of the 1980's were conducted with LMR prototype oxide fuel. In the later part of the 1980's, TREAT concentrated on metal fuel tests and a major upgrade to the reactor. In general, experimental vehicles are placed in the center of the reactor in the location that one or two TREAT fuel assemblies would occupy. The usual test assemblies consist either of stagnant capsules, the single pin flowing sodium test loop, or the multiple pin (up to seven) flowing sodium Mark-III loop.

5.1 Oxide Fuel Test Results (Reference 16)

Severe accident tests were conducted for the UKAEA using the UK design (bottom plenum, annular pellet) mixed-oxide pins previously irradiated in the Prototype Fast Reactor. Tests were performed in stagnant sodium capsules, single pin test loops, and in Mark-III loops with seven pin clusters. The principal tests were Transient Overpower Without Scram with both slow (15 sec) and fast (0.15 sec) period exponential power increases for low, medium, and high initial burnup (up to 9 at.%) pins. The tests provided data on time and location of fuel motion prior to fuel failure, fuel failure modes, cladding failure, motion of fuel in the coolant channel, and blockage. Pin failures in the slow ramp tests were generally near the top of the fuel and just above the midplane in the fast ramp tests. Typically, blockage formed during fuel pin disruption. Failure was weakly dependent on burnup. In all tests, the reactivity was decreased by the fuel motion.

In support of the U.S. oxide fuel program, pins previously irradiated in FFTF from 2 to 54 MWd/KgM were tested at overpower ramp rates of 5¢/s, 50¢/s, and 1$/s in flowing sodium. At the slower transient ramp rate, it is necessary to exceed three times nominal power to induce fuel failure. At the intermediate and high rates, exceeding nominal power by eight or nine times did not cause breach. In two tests, where pins were run to failure, the mechanism of failure appeared to be due to stress rupture because of high internal gas pressure combined with reduced cladding strength at higher temperature. Failure occurred in the upper one-third of the pin. Failure in these tests again appeared to be independent of burnup.

Another series of tests to investigate an advanced fuel design featuring larger fuel pellets, PuO_2-UO_2 fuel and HT9 cladding, showed an increase in the tolerance to pin failure of 4.5 times normal power for the low ramp rates and 16 times for the medium and high ramp rates. These pins also showed a propensity for fuel motion before failure which would help shut down the reactor. Comparison between pins with and without central voids in the fuel pellets showed those with voids had less transient-induced cladding strain and should therefore have a higher failure threshold.

Tests were also performed to investigate the behavior of molten uranium oxide fuel and cladding material during a simulated transition phase of an LMR hypothetical core disruption accident. The hodoscope measured the fuel motion and confirmed the ability of analyses to predict the onset of boilup.

5.2 Metal Fuel Tests (Reference 16)

Advances in metal fuel design since the early 1960's allowed for higher swelling, higher burnup, and greater margin to failure making much of the early database obsolete. So far, six tests on the new fuel design have been conducted in Mark-III flowing sodium loops; more are planned. Anticipated transients without scram in overpower tests have been simulated to observe failure mechanisms and associated pre- and post-failure fuel motion. By using separate flow tubes for each fuel rod with orificed inlets, the failure behavior of rods with different composition and burnups (from fresh to 9.8 at.%) was examined in each test resulting in fifteen different conditions being investigated in the six tests. All tests were conducted with an overpower exponential ramp on an 8 second period.

Failure was caused by stress rupture of the cladding due to thinning because of fuel cladding interaction. The relative importance of pressure and cladding thinning was found to be a strong function of burnup. At low burnups, it was necessary for peak cladding temperatures to reach a value at which the rate of eutectic formation increased by two or three orders of magnitude for failure to occur. Cladding failure is a function of time at temperature. The austenitic clad fuel failed consistently at 4.0 to 4.4 times IFR reference power levels. The single pin studied with the new HT9 cladding did not fail at a power level of 4.8 times the nominal power level. Further tests will be necessary to determine the failure threshold of the HT9 clad fuel.

Cladding failure consistently occurred at the top of the fuel column. Nearly all the molten-alloy (about half of the fuel) was expelled through the small breach to be dispersed in the flowing sodium. This behavior in mitigating or preventing a severe accident is considered a basic characteristic of the metal fuel. Prefailure fuel elongation was sensitive to fuel composition, preirradiation power level, and burnup. The reference U-Pu-Zr fuel expanded by only 2 to 4 percent.[17]

Additional data will be obtained in future tests on the most promising cladding type (HT9) and higher burnups to match those burnups that have been achieved in EBR-II.[18]

5.3 Computer Code Development

The details of fuel motion and clad failure provided by the hodoscope has assisted in the development and verification of models used for best estimate and safety analysis codes. Reference 16 lists many of the codes which have used the TREAT data to verify internal rod pressure, temperature, time and location of failure, failure mechanisms, elongation of solid fuel, and motion of melted fuel. In addition to the data taken during the test, destructive post irradiation examination as well as nondestruc'ive tomographic imaging is performed for details of relocated fuel and steel, flow tube blockages, and pin displacements.[19] Detailed computer codes such as SAS4A and SASSYS[20] which model fuel swelling, fission gas generation and release, pin pressurization, zirconium migration, fuel cladding contact, cladding strain, fuel chemical interaction, and cladding strain and failure have used the TREAT data to verify models. SAS4A was used to analyze the M7 metal fuel experiment in TREAT covering the complete sequence from fuel heatup and melting through pre- and post-failure fuel motion and fuel failure.[21] Good agreement was obtained between timing, characteristics and magnitude of the key fuel relocation events.

6. CONCLUSIONS

EBR-II and TREAT have, and still are, contributing significantly to LMR safety research. The two reactors have complimentary programs with EBR-II demonstrating the behavior of power reactor prototypes under normal and accident situations a d TREAT providing information of the behavior of fuel under severe accident (beyond DBA) conditions. Computer codes, such as SASSYS/SAS4A, which are rapidly becoming an industry standard, are validated against the data obtained in these two facilities so that the resultant code can be used to predict the behavior of an LMR under all conditions.

REFERENCES

1. R. G. Pahl, R. S. Wisner, M. C. Billone, and G. L. Hofman, "Steady State Irradiation Testing of U-Pu-Zr Fuel to >18% Burnup," Proceedings of the 1990 International Meeting on Fast Reactor Safety, August 12-16, 1990, Snowbird, UT.

2. D. C. Wade and Y. I. Chang, "The Integral Fast Reactor Concept: Physics of Operation and Safety," Nucl. Sci. and Eng., 100, 1988, (pp. 507-524).

3. H. P. Planchon, et al., "Implications of the EBR-II Inherent Safety Demonstration Tests," Nucl. Eng. and Design, 101, 1987, (pp. 75-90).

4. L. Burris, M. Steindler, and W. Miller, "A Proposed Pyro-metallurgical Process for Rapid Recycle of Discharged Fuel Materials from the Integral Fast Reactor," Proceedings of the Fuel Reprocessing and Waste Management Meeting, Vol. 2, August 26-29, 1984, Jackson, WY, (pp. 2-257).

5. D. C. Wade and Y. I. Chang, "The Integral Fast Reactor (IFR) Concept: Physics of Operation and Safety," International Topical Meeting on Advances in Reactor Physics, Mathematics and Computation, April 27-30, 1987, Paris, France, (pp. 311-326).

6. C. R. Stevenson, "The EBR-II Fuel Cycle Story," American Nuclear Society, La Grange Park, IL, 1987.

7. B. R. Seidel, et al., "Recent Progress in Development of Metal Fuel," Proceedings of the Conference on the Safety Status and Future of Noncommercial Reactors and Irradiation Facilities, September 31 - October 4, 1990, Boise, ID.

8. Y. Liu, et. al., "Whole-Pin Furnace System: An Experimental Facility for Studying Irradiated Fuel Pin Behavior Under Potential Reactor Accident Conditions," Proceedings of the 1990 International Meeting on Fast Reactor Safety, August 12-16, 1990, Snowbird, UT.

9. C. E. Lahm, J. F. Koenig, and B. R. Seidel, "Consequences of Metallic Fuel-Cladding Liquid Phase Attack on Fuel Element Lifetime During Over-temperature Transients," Proceedings of the 1990 International Meeting on Fast Reactor Safety, August 12-16, 1990, Snowbird, UT.

10. W. K. Lehto, et al., "Safety Analysis for the Loss-of-Flow and Loss-of-Heat Sink Tests Without Scram," Nuc. Eng. and Design, Vol. 101, No. 1, 1987.

11. D. Mohr, et al., "Loss-of-Primary-Without-Scram Tests: Pretest Predictions and Preliminary Test Results," Nucl. Eng. and Design, Vol. 101, No. 1, 1987 (pp. 45-56).

12. E. E. Feldman, et al., "EBR-II Unprotected Loss-of-Heat-Sink Predictions and Preliminary Test Results," ibid (pp. 57-66).

13. W. K. Lehto, E. M. Dean, and R. M. Fryer, "Reactor Safety Implications of Pump Run-Up Tests in EBR-II," Nucl. Eng. and Design, Vol. 110, 1988, (pp. 47-53).

14. L. K. Chang, et al., "Demonstration of EBR-II Power Maneuvers Without Control Rod Movement," Proceedings of the International Topical Meeting on Safety of Next Generation Power Reactors, Seattle, WA, 1987, (pp. 394-401).

15. J. I. Sackett, et al., "EBR-II Test Programs," Proceedings of the 1990 International Meeting on Fast Reactor Safety, August 12-16, 1990, Snowbird, UT.

16. A. E. Wright, D. S. Dutt, and L. J. Harrison, "Fast Reactor Safety Testing in TREAT in the 1980's," ANS International Fast Reactor Safety Meeting Transactions, Vol. II, 1990, Snowbird, UT, (p. 233).

17. T. H. Bauer, et. al., "Behavior of Metallic Fuel in TREAT Transient Overpower Tests," Proceedings, International Topical Meeting on Safety of Next Generation Power Reactors, 1988, Seattle, WA, (p. 857).

18. D. R. Pederson, et al., "Overview of Fast Reactor Safety Research and Development in the United States Since Guernsey," ANS International Fast Reactor Safety Meeting Transactions, Vol. I, 1990, Snowbird, UT, (p. 1).

19. J. A. Morman, et al., "Tomographic Imaging of Severely Disrupted Fuel Assemblies Tested in TREAT," ANS International Fast Reactor Safety Meeting Transactions, Vol. 1, 1990, Snowbird, UT, (p. 441).

20. J. E. Cahalan and T. Y. Wei, "Modeling Developments of the SAS4A and SASSYS Computer Codes," ANS International Fast Reactor Safety Meeting Transactions, Vol. II, 1990, Snowbird, UT, (p. 123).

21. A. M. Tentner, B. Kalimullah, and K. J. Miles, "Analysis of Metal Fuel Transient Overpower Experiments with the SAS4A Accident Analysis Code," ANS International Fast Reactor Safety Meeting Transactions, Vol. III, 1990, Snowbird, UT, (p. 141).