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Status of LMR fuel development in the United States of America

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Three fuel systems oxide, metal, and carbide are shown to be reliable to high burnup and a fourth system, nitride, is shown to have promise for LMR applications. The excellent steady state performance of the oxide and metal driver fuels for FFTF and EBR-II, respectively, supported by the experience base of tens of thousands of test pins is provided. Achieving 300 MWd/kg in the oxide fuel system through the use of low swelling cladding and duct materials and the Integral Fast Reactor (IFR) concept that utilizes metallic fuel are described. Arguments for economic viability are presented. Responses to operational transients and severe over-power events are shown to have large safety margins and run-beyond-cladding-breach (RBCB), is shown to be non-threatening to LMR reactor systems. Results from a joint U.S.-Swiss carbide test that operated successfully at high power and burnup in FFTF are also presented.

1. Introduction

This paper discusses the status of the development of U.S. Liquid Metal Reactor (LMR) fuel systems oxide, metal, and carbide/nitride, and provides some insight into fruitful areas for future consideration. There is no doubt that fast breeders can now be built and operated safely and effectively with long-life fuel systems [1–4]. There remain, of course, the questions of acceptance of nuclear energy in many countries and of economic competitiveness with other energy sources. Reliably achieving high burnup is one of the more important factors that will make the breeder fuel cycle competitive with that of the LWR. The general cost trends with burnup [5] are illustrated in fig. 1 for LWRs and LMRs for oxide fuel. It is clear that breeder fuels must achieve core average burnup levels of about 150 MWd/kg or peak burnups of around 200 MWd/kg.

2. Oxide fuel

The development of mixed oxide fuel (UO_2–PuO_2) was the cornerstone of the U.S. program for over 20 years culminating in its performance being fully demonstrated as the driver fuel for the Fast Flux Test Facility (FFTF). Mixed oxide was selected because of the excellent burnup potential of this fuel system; the relative ease of commercial fabrication, since an oxide fuel fabrication industry already existed for light water reactors; and the proven safety response (negative Doppler coefficient) of the oxide system in mitigating overpower transients [6]. Fig. 2 summarizes the number of oxide pins versus burnup irradiated in full size assemblies under prototypic conditions in FFTF. The response of pins to off normal events was also examined in the Transient Test Reactor, TREAT, facility and in EBR-II.

2.1. Steady state performance

Progress in attaining the goal of 150 to 200 MWd/kg is shown in fig. 3, which summarizes the world's experience for fast reactor mixed-oxide fuel systems. The FFTF which started up in 1980, was able to capitalize on the highly successful experience of PHENIX and PFR, and build on the extensive data base developed from an earlier irradiation test program in EBR-II and GETR [1]. The unique FFTF instrumentation, including coolant temperature and flow measurements for each in-core position, and the capability for in-situ subassembly height measurements and withdrawal loads, provided the technical data required for rapidly increasing burnup with no risk to the operating efficiency of the plant.

2.1.1. FFTF reference fuel

The FFTF reference pin design employs mixed oxide fuel pellets helium bonded to 20% CW Type 316

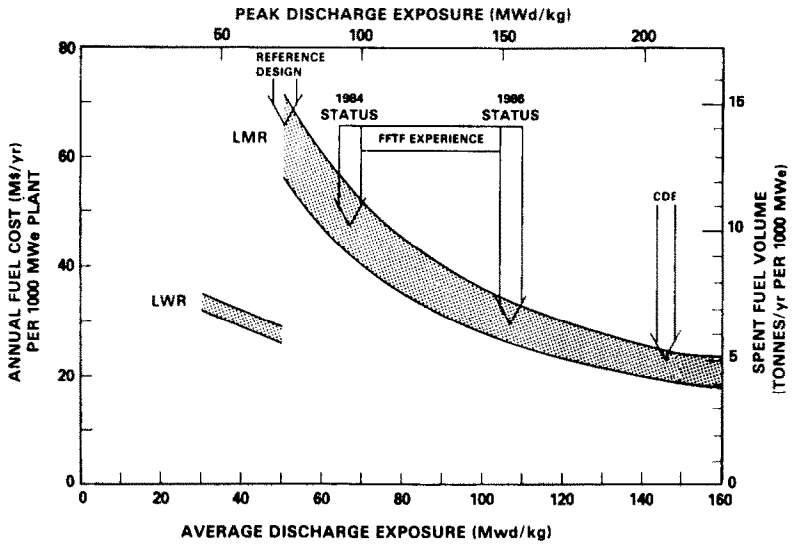

Fig. 1. Fuel economics drives toward increased burnups.

stainless steel (SS) cladding. The smeared fuel density is 85% and the fuel-to-plenum volume ratio is 1.0. The reference fuel development program was extended through the early stages of FFTF operation to qualify full size components under fully prototypic conditions and to establish a basis for extending allowable lifetime. The commercially fabricated [7], reference FFTF driver fuel has performed flawlessly [2,3] to well beyond the design goal peak burnup of 80 MWd/kgM since the reactor started routine operation in 1982. The driver fuel experienced no breaches below 100 MWd/kg and selected assemblies reached 120 MWd/kg before duct distortion forced their removal. Only one breach has occurred in an FFTF fuel pin in a standard FFTF driver fuel assembly; at burnup and fluence of 103 MWd/kg and 16×10^{22} n/cm^2, well beyond design life.

The testing program showed that irradiation-induced swelling is the life limiting phenomenon in the FFTF driver fuel assemblies. For the FFTF fuel system

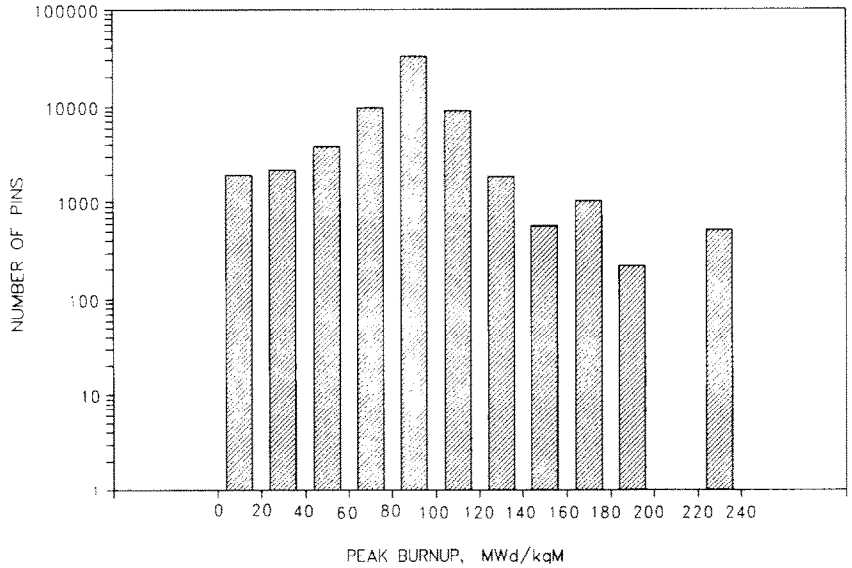

Fig. 2. Oxide fuel pins irradiated in FFTF.

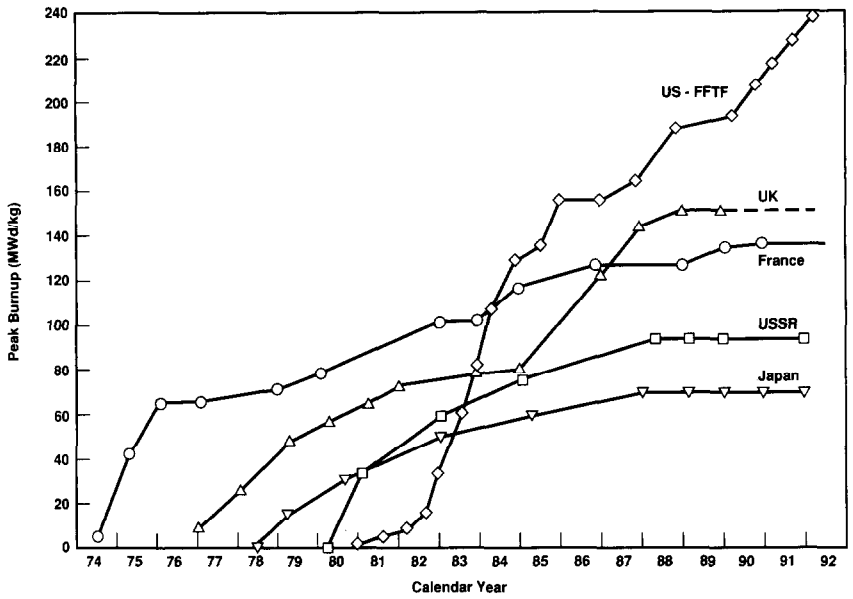

Fig. 3. Burnup achievements in mixed oxide fuel in liquid metal reactors.

using 20% CW Type 316 SS cladding and ducts, a "care-free" residence time to avoid refueling difficulties is a cumulative peak fast fluence ($E > 0.1$ MeV) of about 12×10^{22} n/cm^2 (8 to 10 at.% burnup [1] depending on fuel enrichment). Assemblies can be (and are) left in the core beyond this limit, but calculations of withdrawal loads [8,9] are made for sound core management to preclude operating difficulties. Fuel management techniques such as rotating and/or shuffling assemblies are used to considerably extend lifetime. The driver fuel assembly that developed the breached pin described above was shuffled prior to the breach. Design changes were made to the FFTF driver fuel by using low swelling materials for cladding, ducts, and wire wrap to increase lifetime and by going to larger diameter pins in response to design studies for commercial power reactors. The success in extending lifetime for a given duct elongation limit has been spectacular as illustrated in fig. 4. Elongation, of course, reflects only the swelling, as integrated over the length of the duct. Other distortions, such as bow and dilation which have a major influence on withdrawal loads [8,9] depend not only on swelling but also on irradiation creep and thermal creep. The duct distortions that are routinely calculated for each core load in FFTF are depicted in fig. 5. Each of these distortions has a specific limit [8] associated with it depending on the reactor system of which the assembly is a part.

[1] 1 at.% burnup is equivalent to 10 MWd/kg.

2.1.2. Extended lifetime fuel

Full size assemblies were included in the Cycle 1 core load for FFTF that employed lower swelling, modified austenitic stainless steel alloy D9 for the cladding, wire, and duct. All other specifications (dimensions, smeared density, etc.) were identical with the reference FFTF driver fuel. A target exposure to 160 MWd/kg and 18×10^{22} n/cm^2 was demonstrated and exceeded with the lower swelling D9 [10]. Extra lifetime for "care-free" operation is obtained since the onset of duct distortion is delayed by an increment of about 6×10^{22} n/cm^2, thereby allowing higher fluence (and burnup) to be achieved before refueling constraints are encountered. Actually, mixed-oxide FFTF driver fuel with all D9 components has continued under irradiation in FFTF to a burnup of 188 MWd/kg and fast fluence of 27×10^{22} n/cm^2 showing a large performance margin in this system. For an entire FFTF-like core using D9 components, a very conservative lower exposure limit of 18×10^{22} n/cm^2 could be used for routine, "care free" operation. As with 20% CW type 316 SS components, lifetime can be extended with fuel systems using D9 components by careful core management.

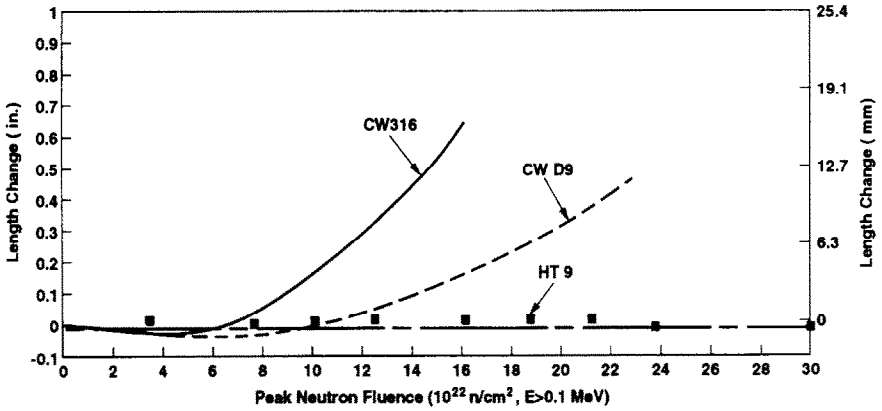

Fig. 4. FFTF subassembly length change with various duct materials.

2.1.3. Long lifetime fuel

Although D9 offers a large (50 to 100%) improvement in fuel assembly lifetime, the material eventually swells, which precludes the achievement of the desired economics of the LMR. Thus, the irradiation of a partial core loading of FFTF known as the Core Demonstration Experiment (CDE) [11] was begun as an aggressive demonstration of the performance of a fuel system using the zero swelling ferritic alloy HT9 for cladding, wire, and duct. The limits of this system

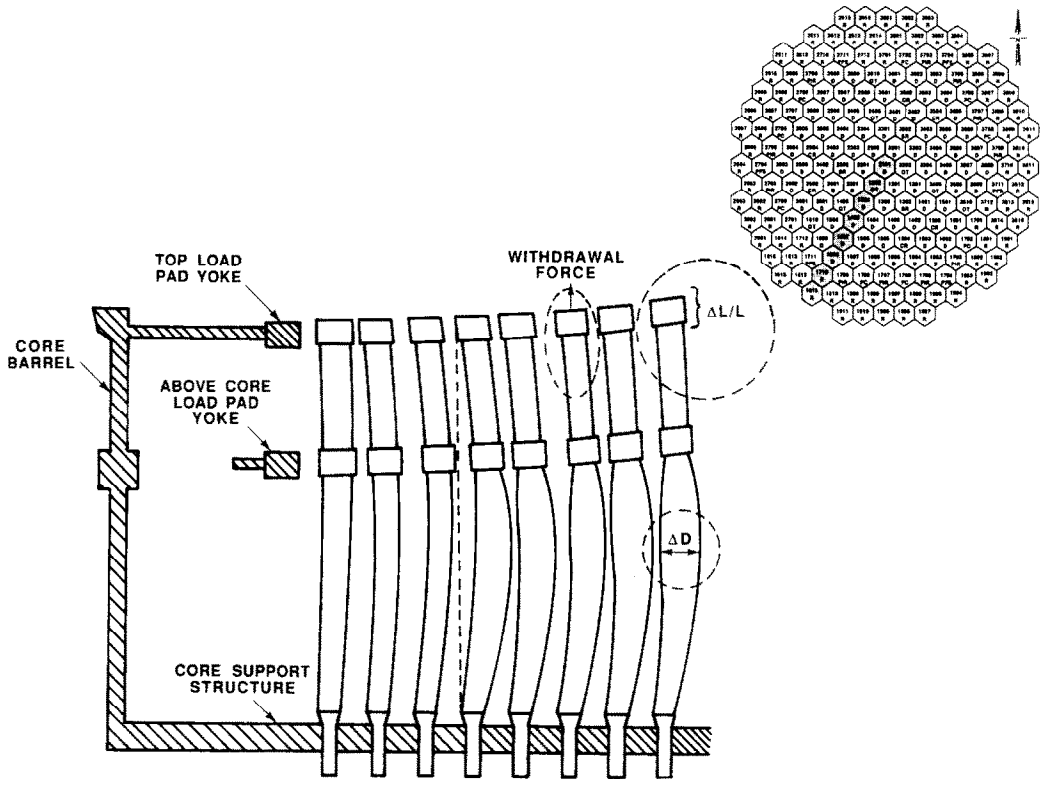

Fig. 5. FFTF duct lifetime criteria.

have not been established yet, but after a burnup of 238 MWd/kg and a fast fluence of 39×10^{22} n/cm^2 (this is equivalent to about 200 dpa) no elongation of the duct occurred (see fig. 4) nor has pin breach been observed [10,12]. There are reasons to believe that burnups of 300 MWd/kg will be readily achievable with HT9 components. It may be necessary to slightly increase the pin gas plenum volume in the CDE design but whether or not this will be required will not be known until the lifetime limits of the HT9 system have been reached. It is anticipated that pin breach will be the eventual limiting feature of fuel systems using HT9 rather than assembly deformation or elongation.

While there is some concern that HT9 has insufficient high temperature strength as fuel pin cladding for all envisioned LMR duty cycles, there is general acceptance [4,9] that HT9 is the choice for a duct (wrapper) material. Furthermore, some of the lead tests for CDE started irradiation at peak cladding temperatures as high as 660°C and were progressively moved inward in the FFTF core to maintain a high cladding temperature as burndown progressed. Clearly, HT9 is a leading candidate for both cladding and ducts. Also, ongoing efforts to improve the high temperature strength of HT9 [13] may provide the designer with the needed assurance to use ferritic components.

2.2. *Off normal performance*

The steady state program has been firmly supported by fuel pin transient testing in TREAT [2,3], operational transient tests in EBR-II [2,14], and run-beyond-cladding-breach (RBCB) studies [15] in EBR-II. The latter two programs involve collaboration between U.S.-DOE and PNC of Japan.

To define transient performance of mixed oxide fuel, EBR-II pins and full-length FFTF pins were transient tested in the TREAT reactor [2,3]. The exposures ranged from 0 to 118 MWd/kgM, with fluences to 12×10^{22} n/cm^2. The overpower transient test ramp rates from 3¢/s to 3$/s demonstrated excellent failure margin of at least three times steady state power. The tests also provided valuable measurements of fuel melting and pin strain as well as breach, which were used to define performance analysis methods for application to new designs and other transient conditions. When driven to extreme transient conditions necessary to produce pin breach, the full-length FFTF pins demonstrated desirable breach characteristics. The tests all exhibited upper pin breach locations and axial extrusion of molten fuel above the fuel column; both of which are beneficial safety characteristics that can delay failure and mitigate accident consequences. Fig. 6 summarizes the margins that exist between pin failure and safety shutdown features of the FFTF [3]. While more than adequate transient margins exists for the fuel system employing austenitic stainless steel components, fig. 6 indicates these margins are increased several additional factors when the ferritic-martensitic, zero swelling HT9 (CDE) is used.

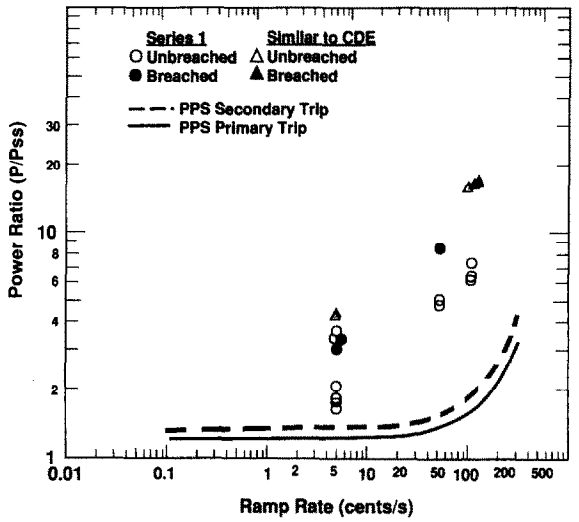

Fig. 6. Safety margins in transient tested mixed-Oxide fuel.

Both the operational transient and RBCB tests in EBR-II have greatly increased our understanding of the performance of oxide fuel under these respective circumstances, and the fuel response has been very favorable. Several RBCB programs, the most extensive of which is probably the cooperative PNC-DOE program [15] conducted in EBR-II, have demonstrated conclusively that breaches are benign events and have provided the base data for reactor operators to define their own criteria for shutting down their plant depending on their specific circumstances. This should not be interpreted to mean that the requirements for fuel pin reliability can be relaxed but rather that the operator has some latitude in operating the plant.

The performance of mixed-oxide fuel pins during operational transients has been adequately demonstrated [14] in the joint DOE-PNC program with aggressively designed pins and the response of the fuel is described by computer codes (e.g., LIFE) that were calibrated to actual performance data. The program includes mixed-oxide fuel pins using first-generation cladding (20% CW type 316), second-generation claddings (D9 and PNC-316) and advanced claddings

(ferritic-martensitic alloy, 20% Ni austenitic stainless steel, and oxide-dispersion ferritic stainless steel). The results demonstrate the capability of second-generation fuel pins to survive a wide range of duty-cycle and extended overpower irradiations. Fuel pins with advanced alloy claddings have been irradiated in EBR-II under steady state conditions and are available for off-normal testing.

3. Metal fuel

3.1. History

During the late 1960s, plutonium bearing metallic fuels were being developed worldwide in an effort to find an acceptable fast breeder fuel [16]. Excellent progress was made toward solution of the problems associated with metallic fuels, e.g., high swelling and eutectic limitations. Experiments underway would eventually demonstrate that high burnup could be achieved with a reduction in smeared fuel density. Further, a U–Pu–Zr alloy possessed not only an adequate solidus temperature with the zirconium additions, but also appeared to exhibit excellent compatibility with stainless steel cladding alloys [17]. At the end of the 1960s, high burnup had not yet been demonstrated for metallic fuels and oxide fuel was selected as the fuel system to be used in the next generation of breeder reactors. Available irradiation performance data for U–Pu–Zr pins were promising, however, and a small effort continued at Argonne National Laboratory.

Interest in metallic fuels for liquid metal reactors (LMR) was revitalized when the Integral Fast Reactor (IFR) concept was introduced in 1984. For a number of reasons, metallic fuel as opposed to ceramic fuels was an essential feature of the IFR concept [18]. Since metallic fuels have high thermal conductivity, fuel pin temperatures are low. Thus, during a loss-of-flow event, the doppler reactivity feedback is small and readily compensated by negative reactivity insertion from thermal expansion of core materials. This inherent safety feature of metallic fueled cores was demonstrated in 1983 in EBR-II [19].

Further, metallic fuels lead to an economical, and proliferation resistant fuel reprocessing scheme. Metallic fuels are readily dissolved and electrorefined in a molten salt electrolyte. The removal of fission products during electrorefining is sufficiently complete for reintroduction of the refined uranium and plutonium into the reactor, but yet the refined product remains highly radioactive [20]. The process is thus accomplished remotely in a hot cell, whereby diversion of the product is difficult. In addition, unless further expensive refining is carried out, the product is useless for weapons.

Finally, if breeding is important, metallic fuels are by far the best choice for a high breeding efficiency because of the high energy neutron spectrum.

The IFR concept is dependent upon a plutonium bearing metallic fuel alloy because of plutonium breeding. When the IFR concept was introduced in 1984 a great deal was known about the U–5Fs [2] metallic fuel that had been the standard fuel for EBR-II since it went critical in 1964. The performance of U–5Fs fuel has recently been extensively reviewed [21]. During the course of utilizing the U–5Fs fuel in EBR-II, significant but simple design changes were made that allowed a ten-fold increase in burnup and robust performance during off-normal events. The smeared fuel density was decreased from 85% to 75% to allow interconnection of fission-gas bubbles and subsequent gas release to a much enlarged gas plenum. High burnup and excellent off-normal performance, including benign run-beyond-cladding-breach (RBCB) performance were thus assured with the large data base as a result of the irradiation of many EBR-II cores between 1964 and 1984. It was then necessary to demonstrate that U–Pu–Zr fuel possessed similar performance characteristics.

3.2. Restoration of fuel fabrication capabilities

Before any of the performance issues could be addressed, a facility was required to fabricate U–Pu–Zr fuel. The facilities used for this purpose in the 1960s had long since been dismantled. Fuel was needed for irradiations, for basic property studies and, further, the facility was required to gain fuel casting experience. A glovebox facility was constructed at the Argonne National Laboratory in Idaho. The large, three compartment glovebox, contained a small injection casting furnace, a cladding closure-weld apparatus, and fuel pin characterization equipment. The injection casting furnace could contain a 1 kg charge and cast as many as six fuel slugs up to 50 cm in length. The construction of

[2] Fissium (FS) is an equilibrium concentration of fission product elements left by the pyrometallurgical reprocessing cycle designed for EBR-II and consists of 2.4 wt% molybdenum, 1.9 wt% ruthenium, 0.3 wt% rhodium 0.2 wt% palladium, 0.1 wt% zirconium, and 0.01 wt% niobium of the total fuel mass.

Table 1
Number of IFR metallic fuel elements irradiated as a function of burnup

Subassembly type	All (> 10 at%)	Pu (> 10 at%)	All (< 10 at%)	Pu (< 10 at%)	Total
Experiment	1611	273	1014	329	2625
Standard core	NA	NA	11484	NA	11484
Total	1611	273	12498	329	14109

Note: The standard core has a very conservative design limit of less than 10 at.% burnup and does not contain plutonium bearing fuel.

the facility, called the Experimental Fuels Laboratory (EFL) was completed in the fall of 1984 and the first fuel elements placed under irradiation in the early spring of 1985 in EBR-II. Since that time, valuable fabrication information on casting parameters and closure-welds on the cladding have been accumulated.

3.3. Irradiation experiments

The first three subassemblies of IFR fuel irradiated in EBR-II all contained the same complement of fuel elements using advanced cladding alloy D9. The only variable in the experiments was the plutonium concentration in the fuel, where two plutonium concentrations were chosen, 8 and 19 at% along with a U–10Zr alloy with no plutonium. The zirconium concentration was the same for all the fuel at 10 wt% with the balance being uranium. These three subassemblies were used to gain a wealth of performance information with the first breach occurring at 18.4 at% burnup [3] [22].

Soon after these three subassemblies were put into EBR-II, an assembly was fabricated for irradiation in FFTF. A series of issues were raised that could best be addressed by irradiating IFR fuel in FFTF. The primary issue involved the question of extrapolating fuel performance results from EBR-II with a fuel length of 34 cm, to the longer cores of proposed commercial reactors of 90 cm or more. Extent of axial fuel growth, fraction of fission gas release to the plenum, migration of fuel constituents, and the stress from fuel-cladding mechanical interaction are all aspects of fuel performance that could be fuel length dependent. Since it is necessary to highly enrich with ^{235}U for EBR-II irradiations, a secondary issue involves the potential for variation in fission product effects depending on U or Pu fission.

As a first step in addressing these issues, an assembly for FFTF was fabricated that duplicated several parameters of the first three subassemblies irradiated in EBR-II. The cladding was D9 and several ternary fuel compositions of 8 and 19 wt% plutonium were included with the binary U–10Zr alloys. The ^{235}U content was adjusted in the binary alloy and lower Pu alloy to provide operating conditions (power and temperature) approximating those in the first three subassemblies in EBR-II. The assembly was irradiated to 10.2 at% burnup in FFTF without breach. Postirradiation examination is currently in progress. Thus far, observed fuel performance results have been consistent with the EBR-II information [23].

A number of subassemblies have subsequently been irradiated in EBR-II to meet various objectives, including tests to examine design options, experiments on prototype designs, and tests to validate the fuel specification [22]. In addition, a series of tests was initiated to study the run-beyond-cladding-breach (RBCB) performance of metallic fuel [24]. The highest burnup achieved to date is 19.3 at% on fuel elements still under irradiation. Table 1 shows the number of IFR fuel elements that have been irradiated to date.

Transient overpower tests on metal fuel pins also have been performed in the TREAT facility [25]. Results showed robust overpower capabilities with cladding failure threshold about 4 times nominal power.

Additional testing of long metal-fueled pins has been conducted in the FFTF, as discussed in a companion paper [26]. Seven full size fuel assemblies containing U–10Zr fuel slugs loaded in non-swelling ferritic/martensitic HT9 cladding have been irradiated under aggressive conditions [high power (548 w/cm) and temperature (640°C)] without failure to burnup levels ranging from 38 to 143 MWd/kg (fast fluence to 20×10^{22} n/cm^2). The duct and wire-wrap material in these test assemblies is also HT9. These assemblies originally supported the use of U–10Zr as a potential driver fuel for FFTF but the results will also support the design of the U.S./DOE Advanced Liquid Metal Reactor (ALMR), which utilizes long fuel pins containing ternary (U-Pu-Zr) metal fuel in HT9 cladding and

[3] 1 at.% burnup is equivalent to 10 MWd/kg.

ducts. Postirradiation examination of these tests will provide data in fuel column growth, fuel and sodium bond performance, cladding strain behavior, fuel-cladding mechanical interaction, and other performance attributes. Should the FFTF continue to operate, ultimate burnup capabilities and breach mode in long metal fuel pins would be determined.

3.4. Property studies and performance analysis

A complete understanding of the U–Pu–Zr fuel system was not achieved when the program was terminated in the 1960s. Also, the interaction between fuel and candidate cladding materials required further evaluation. It was essential to understand phase relationships in the U–Pu–Zr system in order to know the fuel solidus temperature for reactor design purposes, and to know the liquidus temperatures for fuel casting needs. Good progress has been made in both experiment and thermodynamic analyses of the fuel system [27,28]. A complementary series of tests and analyses to understand the radial migration of fuel constituents is also well underway [29]. Therefore, the performance of U–Pu–Zr fuel is reaching an advanced state of understanding.

Fuel-cladding compatibility is being studied by use of the fuel-performance-test apparatus (FPTA). Sections of irradiated fuel and cladding are heated along prescribed temperature ramps and held at peak temperatures for various times. The information gained from these tests includes the temperature where a liquid phase first forms between fuel and cladding, and the rate of penetration of the cladding once a liquid phase forms. These tests are complemented with experiments where entire irradiated fuel elements are subjected to temperature ramps and hold times representative of possible off-normal events. These tests are called whole pin tests (WPT). The combined testing program and subsequent analyses provide a strong base to understand and predict fuel performance during off-normal situations [30,31].

The focal point for much of the ex-core analyses is the development of the LIFE-METAL steady-state performance code and the F-PIN transient code for metallic fuel [30,31]. These codes have been under development for a number of years and have now reached a point where reliable predictions can be made. For example, both codes are used for predictions of the WPT tests with remarkably good results.

3.5. Future direction

Since the restart of metallic fuel development in 1984, most of the performance questions that remained from the 1960s have been answered. It is now apparent that metallic fuel is a viable commercial option, and furthermore, the use of metallic fuel is central to the IFR reprocessing technology. Two main tasks remain to be completed. The first is to demonstrate the same performance characteristics on metallic fuel that has been remotely reprocessed. This demonstration will begin in about a year when the Fuel Cycle Facility begins operation at Argonne National Laboratory in Idaho Falls, Idaho. At that time, the entire core of EBR-II will be converted to U–Pu–Zr fuel, which will be irradiated, reprocessed, and reirradiated.

The second main task is the completion of the compilation of all the fuel performance information in a form suitable for the defense of licensing questions and in a form adequate for the design of new LMR's. Several efforts are underway to accomplish this task. A Metallic Fuels Handbook is in preparation that will not only include fuel and cladding property data, but also irradiation performance information. In addition, a computer data base is nearing completion where all the fuel performance information generated is stored for ready recall for an anticipated wide variety of analyses. Finally, a fuel specification has been developed that will be revised and refined as experience is gained on the performance of reprocessed metallic fuel.

4. Carbide / nitride fuel

The physical and irradiation performance characteristics of carbide and nitride fuel have much in common and the information from one fuel system frequently (in a general sense) applies to the other.

Carbide fuel was selected as the driver fuel for the Indian Fast Reactor and has performed well [32] but the only work on carbide fuel in the U.S. since that reported on in Tucson in 1986 [2] has involved a joint U.S.–Swiss irradiation test in FFTF [33,34] that irradiated (UPu)C pellets and microspheres in D9 cladding. The 91 pin assembly operated successfully at an initial 83 kW/m peak linear power to a burnup of 85 MWd/kgM. Postirradiation examinations [35] show excellent performance of both types of fuel. These examinations included measurements of pin length, fuel column length, pin diameter profiles, gamma radiation profiles, and fission gas release, and detailed metallography. All observations were within the established data base for mixed-carbide fuel and demonstrate the capability of carbide fuels for providing reliable operation with high power densities for fast reactor application.

The adoption by the Space Power Program of UN fuel [36,37] and the recent world-wide interest in (U,Pu)N has sparked some limited evaluations [38] in the U.S. of the potential for application of this fuel system to LMR's. The earlier EBR-II program with mixed nitride fuel [39] suggested a burnup potential of 20 at.% at very high (100 kW/m) linear heat ratings. Interestingly, nitride fuel exhibits many of the same desirable characteristics of metal fuel, i.e., high heavy metal atom density, good thermal conductivity, and excellent compatibility with sodium. It has the added advantage of being compatible with existing fabrication and reprocessing methods established for oxide fuels. The excellent performance of nitride fuel in the Space Power program, the expected similarity in behavior to carbide fuel and foreign irradiation programs will continue to be monitored closely for potential consideration in the future U.S. LMR program.

5. Conclusions

The U.S. program has determined:

(1) Bundle and duct distortion due to neutron induced swelling limits the lifetime of fuels systems using austenitic type 316 SS and the advanced austenitic D9 alloy to about 12×10^{22} n/cm^2 and 18×10^{22}, respectively or about 100 to 150 MWd/kg.

(2) The oxide fuel system is capable of burnups in excess of 200 MWd/kg using very low swelling HT9 as the cladding, duct and wire wrap as was demonstrated with the Core Demonstration Experiment (CDE) in FFTF which consisted of ten fuel and six blanket assemblies plus four other fuel assemblies of the same design.

(3) The Integral Fast Reactor (IFR) concept employing the ternary U–Pu–Zr metal fuel alloy, currently the mainline U.S. program for LMR's, has been shown to meet the fuel performance needs of a commercially viable fast reactor.

(4) Mixed-carbide fuel has the capability to provide reliable operation to high burnup at extremely high linear powers.

(5) Mixed-nitride fuel shows promise to meet the capability required for future LMR application.

References

[1] American Nuclear Society, Int. Conf. on Fast Breeder Reactor Fuel Performance, Monterey, CA, 1979.
[2] American Nuclear Society, Int. Conf. on Reliable Fuels for Liquid Metal Reactors, Tucson, AZ, 1986.
[3] American Nuclear Society, Proc. LMR: A Decade of Progress and Promise, held during the ANS Winter Meeting, 1990, Washington, D.C.
[4] BNES, Fast Reactor Core and Fuel Structural Behavior, Inverness, Scotland, 1990.
[5] R.D. Leggett and R.P. Omberg, Mixed-Oxide Fuel Development, Experience Gained and Path to Economical Power Generation, ANS, 1987, Richland, WA.
[6] J.O. Arterburn et al., ASME Nuclear Engineering Conf., Palo Alto, CA, 1971.
[7] J.F. Williams and L.H. Rice, Trans. ANS, vol. 34 (1980) p. 221.
[8] H. Sutherland, IAEA IWGFR Specialists Meeting, Mottram Hall, South Manchester, UK, 1984.
[9] S.L. Hecht and R.G. Trenchard, BNES, Fast Reactor Core and Fuel Structural Behavior, Inverness, Scotland, 1990.
[10] R.B. Baker, F.E. Bard, R.D. Leggett and A.L. Pitner, in these Proceedings (Proc. Int. Symp. on Fuels for Liquid Metal Reactors, Chicago. IL, 1992), J. Nucl. Mater. 204 (1993) 109.
[11] J.J. Laidler and R.J. Jackson, Trans. ANS, vol. 47 (1984) p. 190.
[12] A.E. Bridges, R.D. Leggett, R.B. Baker, A.E. Waltar, and B.C. Gneiting, Proc. Int. Conf. on Fast Reactors and Related Fuel Cycles, Kyoto, Japan, 1991.
[13] S. Nomura, S. Shikakura, S. Ukai, I. Seshimo, M. Harada, I. Shibahara and M. Katsuragawa, ibid. ref. [12].
[14] A. Boltax, L.A. Neimark, H. Tsai, M. Katsuragawa and S. Shikakura, ibid. ref. [12].
[15] R.V. Strain, J.H. Bottcher, K.C. Gross, J.D.B. Lambert, S. Ukai, S. Nomura, S. Shikakura and M. Katsuragawa, ibid. ref. [12].
[16] J.H. Kittel et al., Nucl. Eng. Des. 15 (1971) 373.
[17] C.M. Walter, G.H. Golden and N.J. Olson, U–Pu–Zr Metal Alloy: A Potential Fuel for LMFBR's, ANL-76-28, Argonne National Laboratory.
[18] C.E. Till and Y.I. Chang, Proc. Int. Symp. on Fuels for Liquid Metal Reactors and Related Fuel Cycles, Kyoto, Japan, 1991.
[19] D. Mohr, L.K. Chang, E.E. Feldman, P.R. Betten and H.P. Planchon, Nucl. Eng. Des. 101 (1987) 11.
[20] L. Burris, R.K. Steunenberg and W.E. Miller, AICHE Symp. Series 83 (1987) 135.
[21] L.C. Walters, B.R. Seidel and J.H. Kittel, Nucl. Tech. 65 (1984) 179.
[22] R.G. Pahl, D.L. Porter, D.C. Crawford and L.C. Walters, J. Nucl. Mater. 188 (1992) 3.
[23] H. Tsai and L.A. Neimark, Proc. Int. Conf. on Design and Safety of Advanced Nuclear Power Plants, Tokyo, Japan, 1992.
[24] G.L. Batte and G.L. Hofman, Proc. of 1990 Int. Fast Reactor Safety Meeting, Snowbird, Utah, 1990.
[25] T.H. Bauer et al., Nucl. Tech. 92 (1990) 325.

[26] A.L. Pitner and R.B. Baker, in these Proceedings (Proc. Int. Symp. on Fuels for Liquid Metal Reactors, Chicago, IL, 1992), J. Nucl. Mater. 204 (1993) 124.
[27] L. Leibowitz and R.A. Blomquist, J. Nucl. Mater. 184 (1991) 59.
[28] L. Leibowitz, R.A. Blomquist and A.D. Pelton, J. Nucl. Mater. 167 (1989) 76.
[29] M.C. Petri, A.G. Hins, J.E. Sanecki and M.C. Dayananda, submitted for publication in J. Nucl. Mater.
[30] J.M. Kramer, Y.Y. Liu, M.C. Billone and H.C. Tasi, in these Proceedings (Proc. Int. Symp. on Fuels for Liquid Metal Reactors, Chicago. IL, 1992), J. Nucl. Mater. 204 (1993) 203.
[31] Y.Y. Liu, H.C. Tsai, M.C. Billone and J.M. Kramer, in these Proceedings (Proc. Int. Symp. on Fuels for Liquid Metal Reactors, Chicago. IL, 1993), J. Nucl. Mater. 204 (1993) 194.
[32] S.B. Bhoje et al., Int. Conf. on Fast Reactors and Related Fuel Cycles, Kyoto, Japan, 1991.
[33] H.P. Alder et al., ANS Winter Meeting, San Francisco, CA, 1989.
[34] R.W. Stratton, G. Ledergerber, F. Ingold, T.W. Latimer and K.M. Chichester, in these Proceedings (Int. Symp. on Fuels for Liquid Metal Reactors, Chicago. IL, 1993), J. Nucl. Mater. 204 (1993) 39.
[35] R.E. Mason, C.W. Hoth, B.J. Makenas, R.W. Stratton and F. Bota, presented at the Int. Symp. on Fuels for Liquid Metal Reactors, Chicago, IL, 1992.
[36] C.M. Cox, D.S. Dutt and R.A. Karnesky, Proc. 1st Symp. on Space Nuclear Power Systems, Albuquerque, NM, 1984, vol. 2, p. 301,
[37] R.B. Matthews, Trans. 7th Symp. on Space Nuclear Power Systems, Albuquerque, NM, 1990, CONF-900109.
[38] W.F. Lyon, R.B. Baker, R.D. Leggett and R.B. Matthews, Proc. Int. Conf. on Fast Reactors and Related Fuel Cycles, Kyoto, Japan, 1991.
[39] A.A. Bauer, P. Cybulskis, R.L. Petty, N.S. Demuth, Proc. Int. Conf. on Fast Breeder Reactor Fuel Performance, Monterey, CA, 1979.