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Thirty years of fuels and materials information from EBR-II

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Abstract

The Experimental Breeder Reactor-II (EBR-II) was a 62.5 MWt–20 MWe sodium cooled fast reactor that was operated successfully for 30 years. Over its period of operation a wealth of fuels and materials information originated from EBR-II. Several missions were conducted in EBR-II, all of which yielded new and valuable additions to the world's knowledge base for nuclear materials. Some of the first pioneering experiments on irradiation effects in stainless steels were conducted in EBR-II. Later, practical manifestations of enhanced irradiation creep, swelling, and loss of ductility were experienced on EBR-II components. In addition, for a period of more than 15 years, the EBR-II reactor would become the primary irradiation facility for all fast reactor fuels and materials research and development. Both the initial mission and final mission for EBR-II (the Integral Fast Reactor Concept, IFR), involved the remote reprocessing and irradiation of fast reactor metallic fuels. The fuels and materials information gleaned from these missions will be summarized with the intent of portraying a sample of the valuable legacy that EBR-II contributed to the world's store of nuclear fuels and materials knowledge.

1. Brief history of EBR-II/description

The Experimental Breeder Reactor II (EBR-II) went critical in 1964. In 1951, its predecessor, EBR-I, generated the first electrical power using a nuclear reactor. Both reactors are located on the high desert near Idaho Falls, Idaho. EBR-II is a liquid sodium cooled fast reactor with the reactor core immersed in a 90,000 gallon pool of sodium. Driver fuel for the reactor has always been variations of metallic fuel. The core of the reactor was originally surrounded by a blanket of depleted uranium, both axially and radially. Later, for neutron economy, some inner rows of blankets were replaced with stainless steel assemblies. The axial blankets were also removed. The reactor generated 62.5 MW of thermal power and 20 MW of electrical power. Over the years the reactor consistently achieved a capacity factor of 70% or better despite the continual use of the reactor for experimental programs.

Of course the reactor did not stand alone. It was surrounded by all the facilities and capability required for the development of fast reactors and the accompanying fuel cycle. Fig. 1 shows the EBR-II reactor and associated facilities. Well equipped hot cells for the examination of components, an analytical chemistry laboratory for analysis of radioactive materials, and metallurgical laboratories for use in the characterization and fabrication of fuels and materials all grew around the reactor. As well, two other reactors existed at the site; one called the TREAT reactor for the overpower transient testing of fuels, and the other being the Zero Power Physics Reactor for use in generating physics data for any fast reactor core configuration. This capability in Idaho was well supported by analytical and experimental efforts at the Argonne National Laboratory site in Chicago, Illinois.

The EBR-II reactor assumed several missions over its 30 years of operation with some overlap as the transition was made from one mission to another. At the beginning, the goal of the EBR-II installation was to demonstrate the viability of a closed fuel cycle. EBR-II would generate electricity, breed plutonium from depleted uranium, and the fuel would be reprocessed and returned to the reactor in a closed fuel cycle [1]. This mission was completed by 1969 with 35 000 fuel pins (about seven cores) having been reprocessed, refabri-

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Fig. 1. The EBR-II reactor in the background with the power plant on the left and the fuel cycle hot cells on the right.

cated remotely, and returned to the reactor for further irradiation. Although the demonstration was incomplete because plutonium was never removed from the fertile blanket for reirradiation nor were waste forms developed for containment of fission products, the reprocessing demonstration was valuable. The early reprocessing demonstration provided the basis for the return of the concept some 15 years later.

At the end of the reprocessing demonstration, significant decisions were made in the Atomic Energy Commission (AEC) that set the course for all fast reactor development throughout the 1970s. Fast reactors were thought to be the power source for the future, but an interim period was envisioned where light water reactors (LWR) fueled with uranium oxide fuel would assume the nation's electrical power needs. The LWR technology emanated from the submarine reactor development. Utilities felt comfortable with water coolant, but not with sodium. Uranium oxide fuel was compatible with water, metallic fuel was not. Thus, a worldwide industry was created around the oxide fueled LWR reactors. This choice for the LWR industry influenced the direction to be taken for fast reactors. Even though little was known about the performance of oxide fuel in a fast reactor, the mixed uranium oxide–plutonium oxide fuel was chosen for fast reactors. Three reasons were given for the oxide option. First, the sodium coolant outlet design temperatures for fast reactors in the late 1960s were very high, too high for the steel structural components. Metallic fuels could not achieve these temperatures without the possibility of deleterious fuel–cladding interaction. It appeared that oxide fuels could. Second, at that time of decision in the late 1960s, metallic fuels could achieve only limited burnup due to fuel swelling. A design breakthrough was only months away from demonstration in EBR-II, but oxide fuels were believed to be capable of much higher burnup even though no irradiation data

were available to prove this. Finally, oxide fuel fabrication plants were expanding worldwide to support the LWR industry and this technology supported the mixed oxide fast reactor fuels.

Once these decisions were made for the course of fast reactor development, then a second mission emerged for EBR-II. The reactor would become the primary irradiation facility for all fast reactor fuels and materials research and development. This mission remained steady until 1978 at which time the larger Fast Flux Test Facility (FFTF) at Hanford would assume this irradiation mission. This period was a particularly interesting period. Every reactor and fuel vendor, along with all the national laboratories involved in fast reactor development, conducted irradiation programs in EBR-II. This was a time when many highly innovative test vehicles were designed and installed in EBR-II within which complex fuels and materials tests were conducted.

With no identified mission in 1978, the Energy Research and Development Association (ERDA), a successor organization of the AEC, gave Argonne a directive to shut down EBR-II by December 1980. However, before EBR-II could be shut down, a new mission was identified. Two issues existed with the performance of fast reactor fuels that could not be addressed in existing irradiation facilities. One was the behavior of fuel after cladding breach [2]. Questions existed on the rate of degradation of the fuel with the possibility of fuel expulsion and flow blockages. The other issue was associated with the behavior of fuel when subjected to overpower transients where the reactivity insertion rate is slow [3]. Such an event would occur if an operator inadvertently withdrew a control rod. These issues could not be explored with existing transient test reactors because their minimum reactivity insertion rates were too high and the fuel could not be properly 'pre-conditioned' before the transient was imposed. This sort of testing with breached fuel and overpower transient operation brought a higher risk to the facility than simply steady-state irradiations, but with FFTF assuming the steady-state work, EBR-II could reasonably be used to address these issues. A special test facility called the Breached Fuel Test Facility (BFTF) was built for EBR-II [4]. The BFTF was highly instrumented such that temperature changes and fuel movement could be measured. To prepare for transient tests on advanced fuel, the metallic fueled core of EBR-II had to be qualified for multiple overpower transient operations. This program of breached fuel experiments and overpower transient testing lasted about four years and was highly successful.

It appeared that metallic fuel would not be considered in the development of liquid metal cooled (Na) fast reactors (LMRs). A series of events beginning in the late 1970s caused a reassessment of reactor technology, including the LMR and its associated fuel cycle. The perceived threat of nuclear weapons proliferation arising from the reprocessing technology for civilian reactor fuel lead to the abandonment of all civilian reprocessing activities in the United States. Soon after, the Clinch River Breeder Reactor (CRBR) was canceled. These actions left a void in nuclear reactor research and development in the United States. In 1983, a concept called the Integral Fast Reactor (IFR) emerged at Argonne, which offered a potentially safe and economical solution to the technical and institutional issues that have prevented nuclear power from fully contributing to the world's energy demands [5,6]. Central to the concept was recognition that the world's reserve of ^{238}U must be utilized as an energy source in the centuries to come. Thus, the fuel system must be able to utilize plutonium as its principal fuel and must have the potential to simultaneously create replacement plutonium by irradiating ^{238}U.

Metallic fuel appeared to be the most suitable candidate for the integral concept, and the U–Pu–Zr system, which had been under development in the late 1960s, was chosen over other metallic fuel systems because it promised superior performance particularly in overcoming concerns over fuel/cladding interaction. Excellent neutron economy and high burnup capability had been seen to be attributes of metallic fuel prior to 1983, and additional features of metallic fuel realized during the formulation of the IFR concept made metallic fuels all the more attractive.

In the first place, compared to oxide fuel, metallic fuel has a high thermal conductivity with very significant safety benefits. These benefits were demonstrated at EBR-II when test accidents were initiated at full power, with the loss of primary flow in one test and loss of heat sink in another, each without scram; in both cases, the reactor shut itself down without operator or mechanical intervention.

In the second place, metallic fuel lends itself to straightforward recycling using a novel technique that has several inherent advantages. The key step in the processing of metallic fuel is electrorefining [7]. The cathode product contains uranium, plutonium, and the minor actinides, along with residual fission products. The bulk of the fission products are separated from the cathode product, permitting satisfactory nuclear performance of the recycled reactor fuel. The retained fission products keep the fuel highly radioactive, requiring that all recycling and refabrication steps be carried out remotely in a hot cell.

This recycling technology brings with it several benefits. First, undetected diversion of the fuel is virtually impossible since the material is highly radioactive; use of the material for nuclear weapons is not feasible because further PUREX-type reprocessing would still be required to separate plutonium from the uranium and

remaining fission products. Second, the process involves batch operations, and thus is easily scaled to meet local requirements. Furthermore, comparative cost analysis has shown the process to be very competitive, relative to other recycling options.

Finally, and perhaps of greatest importance, this recycling method allows essentially all actinides to remain in the fuel cycle, to be fabricated back into the recycled fuel and fissioned for useful energy. As a result, the high-level waste that emerges will decay to background in only hundreds of years, rather than tens of thousands.

Through the advantages mentioned above, metallic fuel offers solutions to a number of technical or institutional problems that, early on, had been thought to be inherent in LMRS. However, feasibility of the entire IFR concept, including safety, recycling, and fuel performance, required demonstration. An aggressive program was initiated in 1984 to prove the commercial feasibility of all aspects of the IFR concept, including a demonstration that U–Pu–Zr metallic fuel could meet all the requirements.

Therefore, during the 1980s through 1992, the mission for EBR-II and its complementary facilities was directed towards the demonstration of the IFR concept. Several tasks were pursued in parallel. The hot cell facilities were restored and equipment was installed to electrorefine the fuel and produce the ceramic and metallic waste forms. U–Pu–Zr fuel was fabricated and irradiated in EBR-II to demonstrate adequate safety and reliability margins. Design activities were initiated to produce a reactor and reprocessing facility sized to commercial interests. The IFR became the focus of the Department of Energy (DOE) advanced reactor activities during the 1980s. In addition, there was intense foreign interest in the IFR concept, principally from the Japanese.

In 1992 funding for the IFR concept was canceled due to proliferation concerns by the US Government, and shortly thereafter EBR-II was shut down. The IFR technology was redirected toward the treatment of DOE spent fuel, an activity that is ongoing today at Argonne.

The following are brief discussions of the kind of information that was generated from EBR-II during its 30 years of operation.

2. Materials problems affecting operation of EBR-II

When EBR-II was designed, irradiation phenomena that changed the mechanical properties of steel were largely unknown because no fast reactors had operated prior to EBR-II for a long enough period to notice these changes. Void swelling and irradiation enhanced creep along with irradiation induced embrittlement appeared as problems that were accommodated as operation of EBR-II continued [8,9]. Irradiation induced property changes were, of course, issues only in the primary system of EBR-II. However, interesting materials problems occurred in the secondary and steam systems of EBR-II as well. The effort to constantly improve the performance of the metallic fuel in EBR-II continued from the first day of operation to the last. A summary of the fuels work will follow in a later section.

As early as 1967 it was realized that neutron irradiation would produce rather startling effects in steel. Steel would embrittle, which was a constant concern should a fuel assembly fracture while handling, or cladding would catastrophically fail with ductility loss [10,11]. In fact, both problems appeared during operation, but never to the extent to seriously impede operation. Irradiation induced swelling of components along with enhanced irradiation induced creep led to persistent problems that constantly required operating adjustments.

During operation, the hexagonal ducts that contained fuel or reflector steel would swell and bow. This led to difficulties, if allowed to go too far, in removing and replacing assemblies in the core, and on occasion, jamming an assembly in the fuel assembly storage basket. As well, the core tightened from these effects and the reactivity of the core gradually changed [12]. Fig. 2 shows the fuel handling forces as the EBR-II core tightened due to swelling and bow of the ducts. There were early concerns that perhaps the distortion of the assembly hardware might lead to the impairment of control and safety rod motion. Thus, an intense effort was undertaken to understand and predict assembly distortion such that exposure limits could be fixed for assembly hardware. Assembly hardware could then be removed from the core and replaced before problems occurred. Fuel burnup limits were restricted not by cladding failure, but by distortion of the assembly hexagonal ducts. Limits on reflector lifetimes were directly set to given irradiation exposures which were in turn

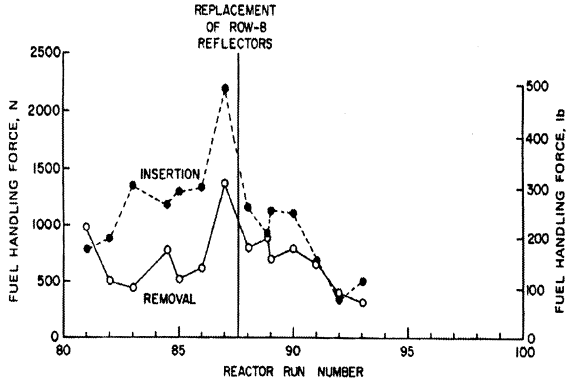

Fig. 2. Average fuel handling forces in row 7 as a function of a run number.

determined by the swelling properties of the materials. At the time, initial in-core hardware was 304 stainless steel, which was later found to be one of the most rapid swelling stainless steels.

Because of the operations problems created from irradiation effects on the cladding and structural materials, ANL directed a strong materials effort toward the understanding of these phenomena. Early experiments were placed in the core of EBR-II where pressurized steel cladding tubes were periodically measured and replaced back in the core for further irradiation. Occasionally one would be destructively examined for immersion density measurements [13]. Other experiments followed on pressurized tubes, springs, and small stressed beam samples, where these experiments were conducted by a number of institutions. The range of alloys under investigation gradually increased.

The original designers of EBR-II had the foresight to put in special assemblies in the core with samples of all the materials used in the primary systems. There were eight assemblies irradiated in row 12 of the reactor core, as well as two in the under-sodium storage basket, the latter used for controls since they received no irradiation in that position. These assemblies, called SURV (for surveillance assemblies), would be removed periodically to examine the corrosion behavior, mechanical properties and other irradiation induced effects [14]. Information from these SURV experiments proved extremely valuable.

Data from these special experiments, when combined with measurements from actual components, gradually yielded enough information to understand and predict irradiation induced swelling and creep and loss of ductility. Once understood, exposure limits were placed on removable core components such that they could be replaced prior to problems occurring. In fact, for most of the life of EBR-II, the burnup limitations on the fuel were not dictated by fuel cladding failure, but by the distortion of the hexagonal ducts that contained the fuel pins.

Alloy development programs were carried out through the 1970s and 80s to find alloys with low neutron irradiation induced swelling rates yet with sufficient ductility. For many alloys tested, low swelling rates meant brittle alloys. However, great improvements to austenitic stainless steels were accomplished by cold-work and small alloy additions such as titanium [15]. Later, a class of ferritic/martensitic alloys emerged with excellent swelling characteristics but lower high temperature strength than the austenitic steels. When fast reactor development was terminated, ferritic/martensitic alloys with oxide particle dispersion hardening were being examined that not only had excellent swelling characteristics, but also good high temperature strength.

On the whole, the secondary and steam system of EBR-II were relatively trouble-free. Liquid sodium, when compared to water and steam, is a very benign coolant. The material used for the EBR-II steam generators was 2–1/4Cr–1Mo steel. Duplex tubes were used where the outer tube was welded to a tubesheet with a free volume above the tubesheet that separated the water/steam from the sodium. The duplex tubes were either mechanically or metallurgically bonded (brazed) to one another. Fig. 3 shows an EBR-II steam generator. Water and steam flowed on the inside of the steam generator tubes while the secondary sodium flowed counter-current on the outside of the tubes. There were two banks of steam generators with each bank having four evaporators and one superheater in a series, with the two banks being connected in parallel. After about 17 years of operation, the temperature in the superheater with mechanically bonded duplex tubes exhibited temperature fluctuations. The mechanically bonded tubes had been prestressed for a good contact such that adequate heat transfer would occur. It was suspected that the stress had relaxed over time with a consequent reduction in heat transfer. Since the temperature fluctuations varied from tube to tube, the primary concern was that axial stresses could develop of such magnitude that the welds to the tube plate would rupture. It was decided to take the superheater out of service because the remainder of the steam generating capacity was sufficient to operate at 62.5 MW and risk of tube rupture would be avoided.

The superheater was removed and sectioned such that the duplex tubes could be removed for analysis. Although the water side of the system showed some corrosion and deposit buildup, the sodium side of the system looked as though it had just been manufactured. In fact, the original chalk marks from the assembly of

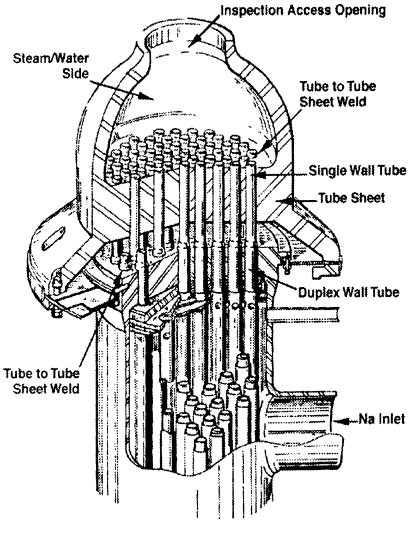

Fig. 3. Steam-outlet region of evaporator EV-702.

parts were still clearly visible. Sections of the mechanically bonded tubes were strain gauged on either the inner surface or outer surface of the tubes [16]. The surface without the strain gauges was incrementally milled away and strain measurements recorded. In this manner, the stress distribution across each tube could be measured, and as well, the interfacial pressure between the two tubes could be calculated. It was found that the tubes had relaxed and the magnitude of relaxation correlated with the temperature fluctuations.

3. EBR-II's role in the development of fast reactor fuels

Once the decision was made to develop mixed oxide fuels for fast reactors, very little national or international attention was given to the further development of metallic fuels. Any advanced fuel work, beyond the mixed oxide fuel system, was directed toward variations of mixed carbide, and later mixed nitride fuels [11,17]. The irradiation programs for mixed oxide fuels in the 1970s and early 1980s in EBR-II were extensive. In addition to the understanding of the fuel pellet behavior during irradiation, all the previously described irradiation effects on the cladding impacted fuel pin performance. Pellet–cladding interactions and fission gas retention and release also were important phenomena that required understanding. As fuel modeling codes matured, the need for additional experiments seemed to necessarily follow. Of course, new irradiation data suggested design variations that would lead to improved performance such that there was continually a new generation of fuel being tested.

In-reactor experiments became more complex, while at the same time the post-irradiation equipment necessary to extract the information in a remote environment became more sophisticated. One of the more important features of EBR-II that evolved during this time period was the technique of gas-tagging experimental fuel pins. Obviously, one of the most important pieces of information that can be gained from an in-reactor experiment is the exact time of cladding failure. The time of failure can be correlated to any number of combinations of design parameters, material conditions, and operating conditions such as temperature, neutron flux, number of shutdowns and startups, etc. A technique was developed where unique combinations of xenon isotopes would be included in the gas plenum of the fuel pins prior to irradiation [18]. When the cladding failed, the gas in the fuel pin would quickly find its way to the reactor gas plenum where this gas was continuously monitored with a mass spectrometer. The particular fuel pin that failed could then be quickly identified. In fact, multiple fuel pin failures could be simultaneously identified. The rate of gas release also gave clues regarding the nature of the cladding failure.

Instrumented tests were carried out in assemblies designed to be located in control rod positions that were no longer required because high worth control rods were put in EBR-II. Thermocouples embedded in the fuel pins were connected to the monitoring station on the floor of the reactor. In addition, special non-instrumented assemblies were designed such that temperatures could be adjusted well above the surrounding temperature of the flowing sodium. A number of temperature monitors were developed with one type being used routinely in most experiments [19]. It was called the thermal expansion difference monitor (TED). The TED was simply a small piece of tubing that was completely filled with sodium at a given standard temperature. When the temperature of the TED was elevated in-reactor, the sodium would expand and deform the tubing. By measuring the immersion density of the TED before and after irradiation, the in-reactor temperature would be deduced from a calibration curve. In addition, most experiments contained a complete set of flux wires. These wires were made from several different elements. Upon neutron activation the neutron fluence and energy spectrum could be determined.

As the more heavily funded research and development on mixed oxide fuels progressed, the development on metallic fuels moved forward as well. Justification for continued development of metallic fuel hinged on the fact that if metallic fuel could be developed with higher burnup, then the capacity factor for EBR-II would improve with fewer defueling shutdowns. These meant, of course, improved irradiation conditions for the experiments in EBR-II. Further, with a higher burnup, less fuel assemblies would be used with a subsequent reduction in operating costs of EBR-II. Fig. 4 shows metallic EBR-II pins being fabricated by injection casting.

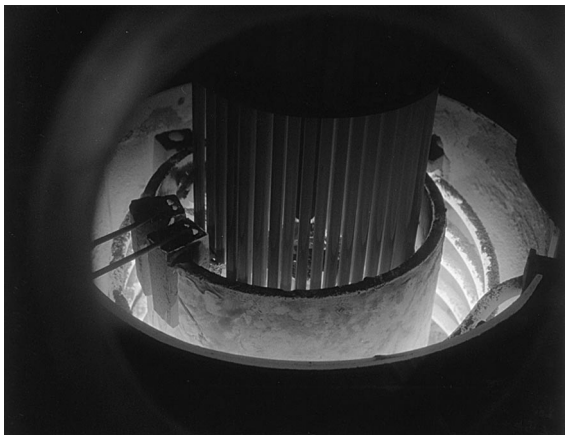

Fig. 4. EBR-II fuel pins fabricated by injection casting.

The original fuel in EBR-II was a uranium–fissium [1] alloy with a high smeared density of 85%. A series of evolutions resulted in the standard fuel for EBR-II being a lower smeared density of 75% with a uranium–zirconium alloy. In parallel, and beginning in the 1980s, development of U–Pu–Zr alloys was once again initiated. The choice of zirconium as an alloying element arose because an additive was needed that would increase the melting temperature of U–Pu alloys. Several elements that alloy well in this system were explored. Chromium, molybdenum, titanium, and zirconium all resulted in an adequate increase in melting temperature over a satisfactory range of plutonium content in the alloy. However, zirconium was unique in that it enhanced compatibility between the fuel and austenitic stainless-steel cladding materials by suppressing the interdiffusion of fuel and cladding components [20]. Without zirconium, the cladding elements nickel and iron readily diffuse into the fuel to form compositions that result in a lower solidus temperature (the temperature at which melting starts) adjacent to the cladding. Should the solidus temperature at the fuel/cladding interface be exceeded during an off-normal event, the cladding could fail due to penetration by the liquid front. The concentration of zirconium in the U–Pu–Zr alloys was limited to about 10 wt% for plutonium concentrations of up to 20 wt%, because too much zirconium would result in a liquidus temperature (the temperature at which melting is complete) that would exceed the softening point of the fused-quartz molds in the injection-casting fabrication equipment used for metallic fuel [21]. By the end of the 1960s, a plutonium-based fuel alloy had been partially developed that had both adequate compatibility with the cladding and a high solidus temperature.

Raising the melting temperature solved only part of the difficulty – there remained the need to achieve high burnup and long residence time in the reactor. Eventually, that was accomplished by a simple change in design, based on theoretical developments in the early 1960s [22,23]; the initial smeared density was reduced [24].

The metallic fuels first used in the experimental breeder reactors EBR-I and EBR-II, in the Fermi nuclear power station, and in the Dounreay Fast Reactor (DFR) had high smeared density (initially 85–100%), with little or no gap between fuel and cladding. Even at low burnup, the cladding deformed and failed when the fuel swelled from accumulation of fission products. Attempts at that time to extend the burnup concentrated on alloying, on thermomechanical treatment of the fuel to suppress swelling, and on the use of strong cladding to resist deformation and the onset of swelling. That work was largely unsuccessful; peak burnups of about 3 at.% being the best achievable.

In fuel without fabrication texture, the primary cause of swelling is the accumulation of fission-product gas in bubbles that grow as gas pressure increases with burnup and overcomes the gas-bubble surface tension, causing the fuel matrix to expand. It was known theoretically that when fuel swelling reaches about 30% the bubbles must begin to interconnect, independent of size and number density. Therefore, it was postulated that if the gap between fuel and cladding were large enough to allow the fuel to swell about 30% before fuel/cladding contact, the bubbles would interconnect and release the accumulated fission gas, thus removing or reducing the primary cause of swelling; a large gas plenum above the fuel would capture the fission gas and keep the stress on the cladding reasonably low. By the time the metallic fuel development program was terminated in the late 1960s, it had been demonstrated that interconnection of pores and subsequent release of fission gas occurred consistently when the smeared density was less than 75% for a range of metallic fuel alloys. Fig. 5 shows the fission gas release as a function of fuel-volume increase [25].

Thus, by the close of the 1960s, solutions of performance problems associated with metallic fuels had been proven feasible – but had not been demonstrated with a large number of fuel pins irradiated to high burnups; only about 18 U–Pu–Zr fuel pins had reached a burnup of as much as 4 at.% in a fast reactor without failure before the metallic fuel development program was ter-

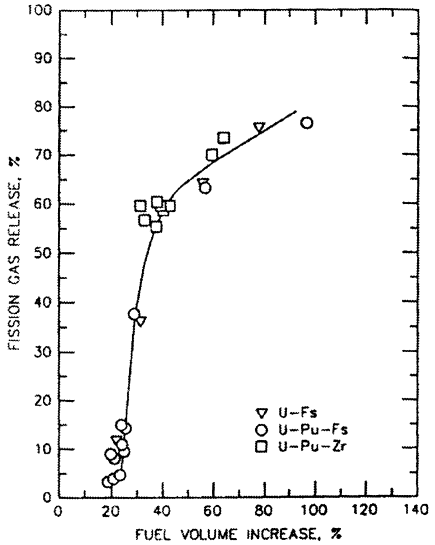

Fig. 5. Fission-gas release versus fuel-volume increase.

[1] 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.

minated [17]. ANL proposed that the core of EBR-II be converted to U–Pu–Zr fuel, clad with austenitic stainless steel, at a smeared density of 75%, with a large gas plenum at the top of the fuel pin. This would replace the uranium–fissium (U–5 wt% Fs) fuel, designated as MK-IA, beginning in 1970. The MK-IA fuel was clad with austenitic stainless steel with an 85% smeared density and had a very small gas plenum. Because of the low burnup achievable with the MK-IA fuel, the decision was made by the US Atomic Energy Commission (AEC), that mixed-oxide fuel would be developed as the nation's fast reactor fuel. However, it was convenient to continue to use metal fuel for EBR-II, which would continue operations as a test reactor for mixed-oxide fuel and advanced cladding material. It was economically desirable to convert the core of EBR-II from the high smeared density MK-IA fuel to a low smeared density fuel that could go to higher burnup, but to keep the fuel composition U–5 wt% Fs instead of U–Pu–Zr [26]. As a result of that decision, a MK-II fuel design emerged in 1970 for use in EBR-II. The MK-II was U-5 wt% Fs at a smeared density of 75%, first clad with Type 304 stainless steel and later with Type 316 stainless steel, with a plenum-to-fuel volume ratio of about 0.6. By 1974 it was clear that the new design was successful. Cladding breach did not occur until about 10 at.% burnup, more than a factor of 3 better than the MK-IA fuel. Fig. 6 shows the steady increases in burnup limit for the EBR-II driver fuel. Ultimately the burnup limit would likely have achieved 15 at.% or better with MK-IIA fuel.

The most common breach mode for the MK-II fuel was a small intergranular crack in the cladding at the restrainer dimples – three small, sharp-bottomed indentations that were placed 120° apart, about 2 cm above the fuel column [27]. Their purpose was to prevent the metallic fuel pin from somehow ratcheting upward inside the cladding, then dropping back down at an inopportune time and creating a reactivity insertion. However, post irradiation examination of a large number of MK-II fuel pins showed only a slight upward motion in a small number of them; thus future designs eliminated any type of restraining device. Later, MK-IIA fuel without the restrainer dimples and a slightly larger plenum volume achieved substantially greater burnup.

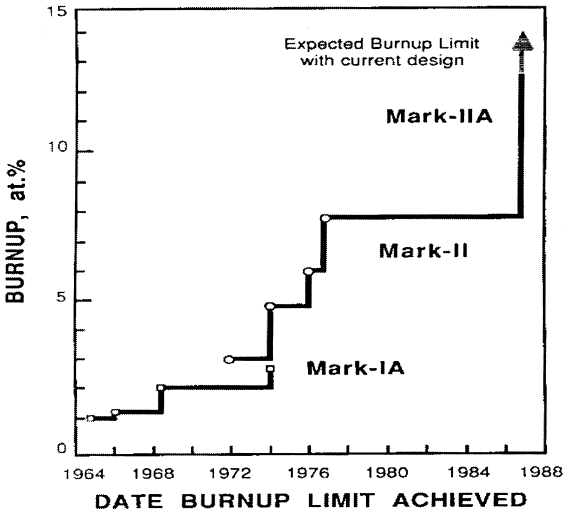

Fig. 6. Steady increases in burnup limit of the EBR-II driver fuel.

More than 30 000 MK-II fuel pins have been irradiated in EBR-II as standard driver fuel, with consistently excellent results [28,29]. The administrative burnup limit for the fuel remained at 8 at.%, even though consistently no cladding breach occurred below 10 at.%. The burnup limit of 8 at.% was chosen for two reasons. First, the 'hex' (hexagonal) ducts on the fuel assemblies were initially made of Type 304 stainless steel. At fuel burnup of 8 at.% (about 8×10^{22} n/cm^2 total), the diameter of the hex ducts had increased from radiation-induced swelling until they could not be handled through the in-vessel EBR-II storage basket. Second, 8 at.% was far enough below the ultimate burnup capability of 10.5 at.% to assure that the probability of in-reactor breach was very low during steady-state operation (<1 failure per core loading) and to provide a wide margin for containing all anticipated effects of off-normal events. With these safeguards in place throughout the 1970s, EBR-II continued to demonstrate that metallic fuel is capable of high burnup.

As mentioned earlier, the IFR concept restored interest in metallic fuel. However, as of 1983 the commercial viability of U–Pu–Zr fuel remained undemonstrated, even though many of the feasibility questions associated with the performance of metallic fuel had already been answered. In fact, additional positive attributes of metallic fuel had been discovered, such as robust performance during off-normal transients. Nevertheless, from 1969 to 1984 no U–Pu–Zr fuel was irradiated and there was no facility available to fabricate the fuel.

With only 18 U–Pu–Zr fuel pins irradiated to about 4 at.% burnup, the data base was weak, although these fuel pins did exhibit the performance that would have allowed them to reach high burnup. Moreover, in addition to the lack of demonstration that the U–Pu–Zr fuel would, in fact, reach high burnup, a number of other issues required further study for complete resolutions.

In 1984, as a result of a broad reassessment of the constraints inhibiting nuclear power deployment, Argonne initiated work on the IFR. In conjunction with this, capability to fabricate ternary fuel was established,

and a fuel program to demonstrate performance was initiated to gain the information that would be needed to eventually obtain a license for metallic fuel from the Nuclear Regulatory Commission (NRC). A number of assemblies were irradiated to establish the burnup potential of the U–Pu–Zr fuel, and how the fuel pins would perform with alternative cladding materials and with a range of design parameters such as smeared density, plenum-to-fuel volume ratio, operating temperature, and linear power. Finally, there was a series of tests to help develop the fuel fabrication specifications. In parallel with the irradiation experimentation, analytical modeling and out-of-core testing were undertaken to improve understanding of fuel performance.

For the initial set of three tests of U–Pu–Zr fuel, the irradiation began in EBR-II in early 1985, and a burnup of 18.4 at.% was reached [30,31]. The test contained fuel of three compositions: U–10Zr, U–8Pu–10Zr, and U–19Pu–10Zr (compositions given in weight percent). The fuel was clad with an austenitic stainless steel alloy, D9. Later in 1985, a test with fuel identical to the first three, but with cladding of the ferritic/martenistic alloy HT9, began its irradiation in EBR-II. Fuel elements clad with HT9 reached 19.9 at.% burnup without cladding breach.

A great deal of information was accumulated from the postirradiation examination of these initial assemblies. It was found that although the microstructure of the alloys was strongly dependent on their composition, the quantity of gas released to the plenum as a function of burnup was consistent for all fuel alloys irradiated, as was the burnup at which the pores became interconnected. Further, it was observed that the initial swelling of the fuel, up to the point of fuel/cladding contact, was anisotropic, with the radial component more than twice as large as the axial component [32]. Still the fuel slug had appreciable axial growth as a function of alloy composition and irradiation conditions. As expected from irradiation results in the 1960s, radial redistribution of the alloying elements was observed, particulary uranium and zirconium, although the radial concentration profile of plutonium was largely unchanged [33]. As the radial concentration of zirconium and uranium changed, a radial distribution of porosity developed, with distinct zones that were evident on a macro scale. Up to the burnups examined, the changes in the diameter of the austenitic cladding could be attributed primarily to radiation-induced swelling and creep, the source of stress being the plenum pressure in the fuel pin.

Up to a burnup of 18 at.%, it appeared that any contribution to the strain in austenitic cladding from fuel/cladding mechanical interaction (FCMI) was insignificant. Cladding-strain data are available for fuel pins with martenistic cladding, up to a burnup of 20 at.%. At that burnup no swelling is expected; indeed the observed cladding strains were small, and could be attributed largely to creep due to plenum-pressure stress alone.

4. Summary

The EBR-II reactor and its companion facilities have had a proud history. While achieving a capacity factor of up to 80%, which is comparable to the best commercial plants operating today. EBR-II contributed greatly to the advancement of reactor design and safety, reprocessing technologies, and fuels and materials development. Every researcher involved in fast reactor development for more than three decades depended in some manner on information generated from EBR-II. For many years the EBR-II site hosted engineers and scientists from around the world. Without a doubt the inevitable utilization of nuclear power in the next millennium will be traceable to information generated from this versatile irradiation facility.

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