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EXPERIENCE WITH ADVANCED DRIVER FUELS IN EBR-II*

by

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Abstract

Several metallic fuel element designs have been tested and used as driver fuel in Experimental Breeder Reactor II (EBR-II). The most recent advanced designs have all performed acceptably in EBR-II and can provide reliable performance to high burnups. Fuel elements tested have included use of U-10Zr metallic fuel with either D9, 316, or HT9 stainless steel cladding; the D9 and 316-clad designs have been used as standard driver fuel. Experimental data indicate that fuel performance characteristics are very similar for the various designs tested. Cladding materials can be selected that optimize performance based on reactor design and operational goals.

1.0 Introduction

The Experimental Breeder Reactor II (EBR-II) is a complete nuclear power plant, incorporating a pool-type liquid metal reactor with a full-power thermal output of 62.5 MW, and an electrical output of 20 MW. Initial criticality was achieved in 1961, utilizing a metallic driver fuel design called the Mark-I, Walters et al. [1], Seidel et al. [2]. This design was based on the use of EBR-II as a breeder and had several features that enhanced breeding, but contributed to a very limited fuel burnup before breach of the cladding occurred. The Mark-I design was modified to obtain higher burnups, but only minor changes were initially allowed. The first major innovations were incorporated into the Mark-II design [1], [2], and allowable burnup then increased dramatically. This design performed successfully and fuel element lifetime in the reactor then became limited by subassembly hardware performance rather than the fuel element itself. Transient performance of the fuel was also acceptable and was impressively demonstrated in 1986 when EBR-II was used to demonstrate that an LMFBR using metallic fuel could survive severe upsets such as a loss-of-heat-sink without scram and loss-of-flow without scram, Seidel et al. [3], Lahm et al. [4], [5]. These tests renewed interest in metallic fuels and Argonne's Integral Fast Reactor (IFR) concept and provided the basis for a conversion from the Mark-II fuel to an IFR prototypic design in EBR-II.

The Mark-II design was used as the basis for several new designs, including the Mark-III and Mark-IV, that are described in the next section. In 1987, the Mark-III design began qualification testing to become a driver fuel for EBR-II. This was followed in 1989 by the Mark-IIIA and Mark-IV designs. The next fuel design, the Mark-V, is being planned for the IFR demonstration that will include utilization of the ternary alloy U-Pu-Zr and reprocessed fuel.

All of the advanced fuel designs tested have demonstrated the ability to exceed the exposure capability of standard subassembly hardware and have done so without breach. Irradiation in EBR-II has indicated that many design options are available to deliver high fuel burnup (>15 at.%) and reliable operating performance under either steady-state or transient operating conditions. Post-irradiation examination of these designs has provided data to support the modeling of metallic fuel performance and to assist designers in optimizing fuel designs for advanced reactors.

2.0 Basic Fuel Designs

The design parameters for a succession of driver fuels used in EBR-II are listed in Table I. The initial subassembly design for EBR-II contained 91 cylindrical fuel elements within a 5.817 cm (2.290 inch) flat-to-flat hexagonal subassembly. A spacer wire was wrapped helically around each fuel element to maintain a triangular lattice spacing and to promote mixing of the sodium coolant within the subassembly. Control assemblies were a smaller size, 4.836 cm (1.904 inch) flat-to-flat, and contained 61 fuel elements. In EBR-II, fuel is used in control rods instead of poisons to maximize utilization of neutrons for breeding. The fuel elements in the control assemblies were identical to the driver subassembly fuel elements. The fuel alloy was uranium-fissium (U-5Fs: 5 weight % Fissium where Fs is nominally 2.5% Mo, 1.9% Ru, 0.3% Rh, 0.2% Pd, 0.1% Zr, and 0.01% Nb). The fuel designs using this fuel were the Mark-I and Mark-II.

In 1984, work was initiated to demonstrate an acceptable fuel design for the IFR program based on the uranium-plutonium-zirconium fuel alloy. There were several advantages to increasing fuel element diameter, so a 61-element subassembly and a 37-element subassembly were designed for use in the experiment program. Fuel alloys irradiated in EBR-II have been based on U-xPu-10Zr, where x included 0, 3, 8, 19, 22, 26 and 28 weight percent. A variety of cladding materials have been tested including the austenitic alloys 316 SS and D9, and the martensitic alloy HT9 which is now the reference choice for ALMR's, Pahl et al. [6], [7].

In 1987, the standard EBR-II driver design was changed from the 91-element design to the 61-element design using a larger fuel element and a qualification program was initiated to determine an acceptable burnup limit. This larger fuel element design was called the Mark-III. A typical fuel element design, used for both the Mark-III and Mark-IIIA is shown in Figure 1. The design characteristics are shown in Table I. The overall fuel element length was increased to 74.9 cm (29.5 inches) to employ maximum plenum volume and wire wrapped on a 15.2 cm (6 inch) pitch. The control and safety rod design could not be easily modified to accommodate the larger fuel elements, so the original 61-element design and 316 SS cladding used for the Mark-II fuel element were retained in these subassemblies and only the fuel alloy was changed. These elements were designated Mark-IIC and Mark-IICS, where S indicated a shorter element. The Mark-III manufacturing campaign, that included the equivalent of a full core loading, is now complete with the last Mark-III fueled subassemblies approaching end-of-life in EBR-II. The driver fuel operated up to 10 at.% burnup and was then removed based on subassembly hardware limitations. Although the qualification program for Mark-III fuel has been terminated, some elements in the qualification program have continued irradiation and now exceed 17 at.% burnup.

The Mark-III fuel is being replaced with Mark-IIIA fuel elements that use the same fuel composition in 20% cold-worked 316 SS cladding. This cladding was chosen based on availability and acceptable performance in EBR-II, rather than improved capabilities. Because of irradiation-induced swelling/creep characteristics, the Mark-IIIA element with 316 SS cladding exhibits more diametral strain than seen in either the D9 or HT9 clad elements, and burnup potential may be less, but performance to 10 at.% burnup has been excellent. The fluence/burnup ratio for EBR-II driver, while varying with burnup and core position, produces an average of 1 x 10^{23}n /cm^2 (E > 0.1 MeV) maximum exposure to the cladding at 10 at.% burnup. The operating parameters for the Mark-III/IIIA fuels are listed in Table II. The qualification program is expected to indicate acceptable performance in EBR-II continuing through the range of 15-20 at.% burnup.

The Mark-IV design, which added the IFR reference HT9 cladding, has been irradiated in qualification subassemblies, but has not been included as a standard driver fuel in EBR-II. The initial procurement of HT9 tubing for the Mark-IV elements has been reserved for use in the Mark-V fuel elements and Mark-IIIA fuel elements are being used until the Mark-V element production can be initiated. Due to similar performance characteristics of U-10Zr and U-20Pu-10Zr fuel designs, data from the performance of Mark-IV fuel will be used to support qualification of the Mark-V design (U-20Pu-10Zr/HT9).

3.0 Irradiation Results

To date, over 13,000 Mark-III, IIIA, IV, Mark-IIC, IICS and other special test elements have been irradiated as part of the IFR demonstration program. A part of these were irradiated in qualification subassemblies for each fuel design. These subassemblies are typically irradiated in a test group of four; three to cover various operating conditions in the core, and the fourth irradiated under 2-σ peak cladding temperature conditions. Post-irradiation examination data at intermediate burnup is now available for Mark-III, Mark-IIIA and Mark-IV fuel. Because the fuel alloy is U-10Zr in all of these designs, the only differences in performance have been due to the different cladding materials. Several performance characteristics have been measured including fuel swelling, fission gas release, and cladding diametral strain.

3.1 Fuel Swelling

Fuel swelling has long been recognized as an important feature of metallic fuels, impacting both core neutronics and fuel performance, and has been investigated for IFR fuel, Hofman et al. [8]. To accommodate fuel swelling, the fuel slug is designed to provide a 75% smeared density when it comes into contact with the cladding. This occurs after approximately 2 at.% burnup.

The axial growth of the fuel depends significantly on Pu content and has been found to be very low (< 6%) for Pu concentrations of 19 wt % and above, but for the binary fuels can be between 6 and 13 %. Note, the axial growth of the driver fuel elements compares well with that measured for U-10Zr experimental elements irradiated under a variety of conditions (see figure 2). The axial growth of the fuel can produce a significant loss of reactivity that requires compensation by control rod motion. Therefore, the choice of fuel alloy should be considered when attempting to minimize control rod worth.

Various means are available to limit this axial growth, if required. The use of zirconium molds has been tested as a means of eliminating the large quantity of glass mold waste in the manufacturing process, and has been shown to significantly limit axial growth, Crawford, et al. [9]. Other methods can also be used, but this has not been a significant problem in EBR-II and may not be a significant issue in other reactor designs.

3.2 Fission Gas Release

Fission products include a significant quantity of noble gases (~25%) that contribute to the swelling of the fuel and tend to increase the internal pressure in the fuel element throughout its life. If the fuel is allowed to swell enough, fission gas bubbles that form in the fuel can interconnect allowing the gas to be released to the fuel element plenum. After irradiation, the pressure and volume of gas in the plenum are measured by laser-puncturing the cladding using a laser to puncture the cladding and collecting the gas released. Pressure and the derived gas release for driver fuels are shown in Table III. Gas release characteristics have been found to be independent of fuel alloy (Pu concentration). The pressure created by the gas is usually the major life-limiting phenomenon, so similar lifetimes are expected for fuel elements of like design, relatively independent of Pu concentration.

3.3 Cladding Strain

The amount of time that fuel elements can now spend in the reactor has increased to the point where thermal and irradiation-induced strain in the cladding (swelling and creep) are significant. The overall effect is measured as cladding diametral strain. Swelling of austenitic stainless steels is well characterized, and has led to the use of martensitic stainless steels because they do not exhibit significant irradiation-induced swelling. Creep effects become significant when burnup is increased and high stresses are created, and operating temperatures are high. For driver fuels in EBR-II, operating temperatures are relatively low, and differences in the performance of different cladding materials are largely due to irradiation-induced deformation. The deformation characteristics of the various cladding materials are compared in Table IV. When operating temperatures are increased, the ferritic HT9 cladding loses strength much sooner than the austenitic alloys, and reactor life-time can be significantly decreased, Pahl et al. [7], [10].

3.4 Off-Normal Performance

Behavior of the driver fuel under off-normal conditions (loss-of-cooling, transient overpower) is also tested to demonstrate that the fuel will survive these events without cladding breach. A series of ex-reactor overheating tests are used to qualify the fuel for loss-of-cooling events [11]. In addition, a recent test has shown that Mark-IIIA and Mark-IV fuel, after irradiation to 9 at.% burnup (near the burnup limit) could survive a 0.1%/s overpower transient to 40% overpower without breach of the cladding.

4.0 Conclusions

Advanced driver fuels tested in EBR-II have all performed well under both steady-state and transient conditions. Post-irradiation examination of driver fuel at the current EBR-II burnup limit has indicated that performance is consistent with previous experimental data. There have been no fuel element failures detected in the Mark-III, Mark-IIIA, and Mark-IV driver fuel programs, and a significant number of Mark-III and Mark-IIIA fuel elements have now been operated to 10 at.% burnup, under various core conditions, and performed well. Although the IFR demonstration program will require conversion to Mark-V fuel elements containing U-Pu-Zr fuel, the performance of the binary Mark-III and Mark-IIIA fuels in EBR-II has exceeded all requirements for the driver fuel and would be acceptable choices for the future.

4.0 Acknowledgements

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Table I

Design Parameters (Nominal) of EBR-II Driver Fuel Elements

Item	Mark-IA	Mark-II	Mark-IIC	Mark-IICS	Mark-III	Mark-IIIA	Mark-IV
Fuel alloy, wt %	U-5Fs	U-5Fs	U-10Zr	U-10Zr	U-10Zr	U-10Zr	U-10Zr
Enrichment weight, % ^{235}U	52	67	78	78	66.9	66.9	69.6
Fuel-slug mass, g	64	52	47	47	83	83	78
Fuel smeared density, %	85	75	75	75	75	75	75
Cladding-wall thickness, cm	0.023	0.030	0.030	0.030	0.038	0.038	0.046
Cladding-wall OD, cm	0.442	0.442	0.442	0.442	0.584	0.584	0.584
Length, cm	46.0	61.2	63.0	53.6	74.9	74.9	74.9
Cladding material	304L	316	316	316	CW D9	CW316	HT9
Spacer-wire diameter, cm	0.124	0.124	0.124	0.124	0.107	0.107	0.107

Table II

Mark-III Operating Conditions*

Reactor Row	Subassembly Flow l/s	Coolant Outlet Temperature, °C	Peak Pin Power, kW/m	Peak Inside Cladding Temperature, °C
1	6.3	477	49	547
2	6.1	478	50	549
3	5.6	485	48	557
4	4.5	508	47	555
5	4.0	514	44	592
6 (Normal Flow)-Corner	3.1	523	37	596
6 (High Flow)-Flat	3.5	518	40	595

*Maximum for hottest operating conditions in a nominal core.

Table III

Average Plenum Pressures and Fission Gas Release

Type	Burnup at. %	Pressure*	Gas Release
		MPa (psi)	%
Mark-III	9.2	2.4 (350)	83
Mark-III	14.4	3.6 (520)	80
Mark-IIIA	9.2	2.2 (320)	76
Mark-IV	8.7	2.1 (310)	74

*Pressures are reported for room temperature conditions ($\approx$300°K).

Table IV

Peak Diametral Strain ($\approx$9 at.% Burnup)

Type	Average of Peak Strain	Maximum Strain
	%	%
Mark-III	0.81	1.17
Mark-IIIA	1.07	1.26
Mark-IV	0.42	0.70

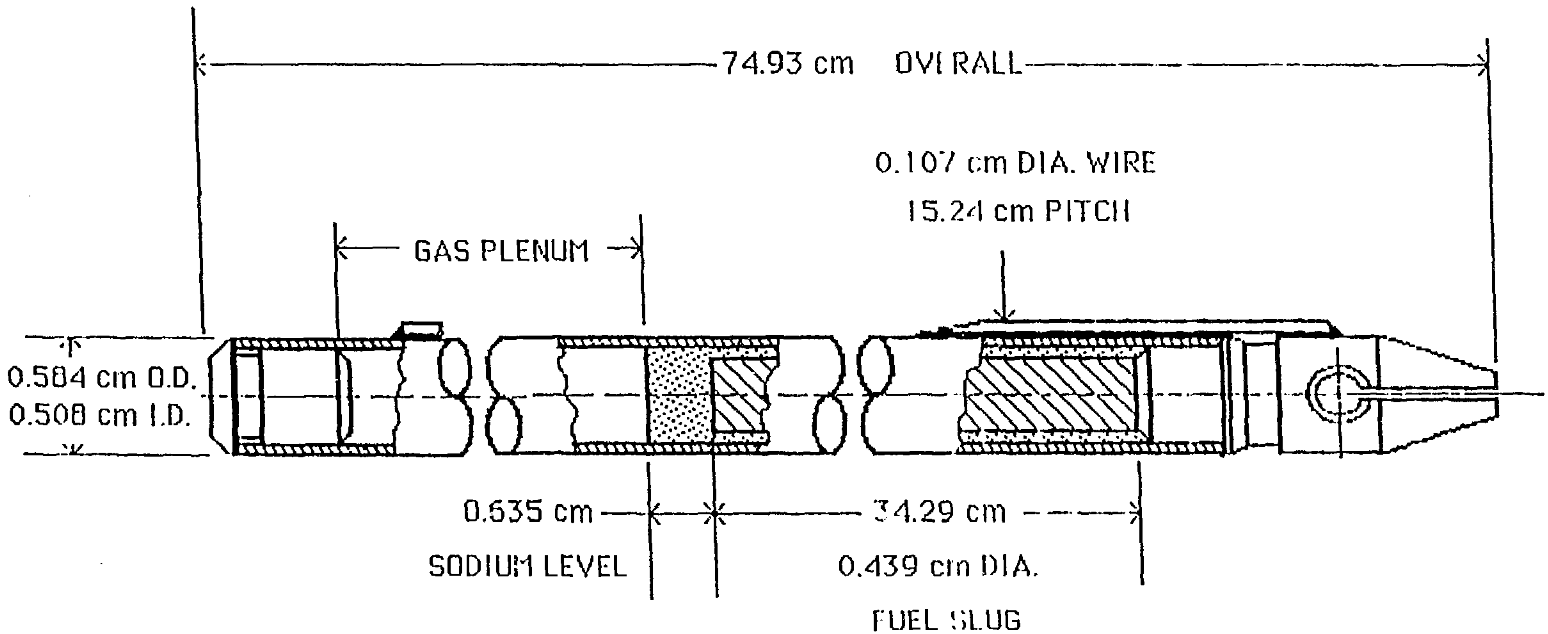

Figure 1. Typical Mark-III or Mark-IIIA Fuel Element

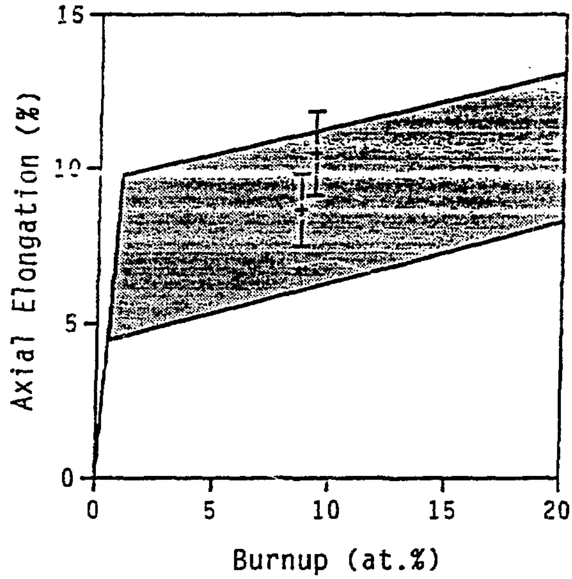

Figure 2.

Axial Elongation for U-10Zr.
Range of experiment data shown shaded; MK-IIIA, IV
Data range plotted.