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THE STATUS OF ENGINEERING DESIGN AND IRRADIATION TESTING
OF THE STAINLESS CLAD-MIXED OXIDE FUEL SYSTEM
FOR FAST BREEDER REACTORS

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INTRODUCTION

Stainless steel clad plutonium oxide-uranium oxide was selected as the basic fuel system to be developed for the Liquid Metal Fast Breeder Reactor (LMFBR). This system was chosen because of the relatively advanced state of knowledge from similar water-reactor systems and the excellent potential for improved performance.

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Major progress has been made in the technology and application of stainless steel clad mixed oxide fuels since 1964. Notably, large-scale testing in fast flux and sodium environments has evolved throughout the world [1]. In the U.S., the Experimental Breeder Reactor (EBR-II) has provided the most significant capabilities for systematic testing and development of LMFBR fuels and materials.

The total development program encompasses an increasingly large number of government and commercial nuclear laboratories which are directly involved in design, fabrication, and testing of LMFBR fuels. This effort is increasingly concerned with quantitative evaluation of fuel performance. Complementary materials development programs involve irradiation and evaluation of more than 13,000 highly characterized specimens.

An unprecedented level of quality assurance is demanded throughout fabrication, irradiation testing, and postirradiation examination of fuels to assure data which are adequate for accurately predicting the performance of fuel pins for demonstration reactors and commercial power stations.

The Fast Flux Test Facility (FFTF) now under construction is a cornerstone of the LMFBR Development Program. Initial fuel cores will provide statistically significant information on fuel fabrication and performance of fuel under conditions directly applicable to demonstration reactors and commercial LMFBR's. Facilities within the FFTF will also permit testing of advanced breeder reactor fuel assemblies under precisely monitored and controlled conditions.

Companion papers describe the fabrication development program [2], cladding development [3], performance modeling [4], and supporting fundamental studies [5].

DESIGN STATUS

The overall status of the engineering design of the oxide-stainless steel fuel system is indicated by current design parameter values for the FFTF and for typical LMFBR demonstration plants (Table 1). Fuel pin design is based on current knowledge of fuel behavior in a fast flux sodium environment [6, 7, 8, 9].

The behavior of mixed oxide, stainless clad fuels depends on the complex interrelationship of a number of simultaneously occurring phenomena. The sequence of events in a fuel pin from startup to the end-of-life must be quantitatively known. The fuel cracks, restructures, expands, redistributes and interacts mechanically with the cladding. Solid and gaseous fission products coalesce, diffuse within the fuel pin and interact with fuel and cladding. The mechanical properties of the cladding are altered by irradiation, the cladding swells, and cladding-coolant interactions occur. The cladding is subjected to fuel-cladding mechanical interaction, fission gas pressure, and stresses resulting from cyclic and transient reactor operation. In the event of failure, both fission gas release and fuel-coolant interaction are potential mechanisms for propagation of failure.

Design optimization requires information relating design parameters to these phenomena. The broad objective of the irradiation testing program

described later is to provide definition of these interactions through the use of fuel pin performance models.

Performance models now under development in the U.S. include OLYMPUS, FMODEL, LIFE, BEHAVE, CYGRO-F, AND SWELL [4]. Special purpose codes which consider only certain aspects of fuel pin behavior have also been developed.

The LMFBR fuel pin design reflects some of the uncertainty associated with known and potential problem areas such as degradation of the cladding due to chemical attack by fuel and coolant and degradation of mechanical properties.

A strain limit totaling 0.2% based upon thermal creep is currently used for both the FFTF and typical demonstration reactor fuel pin cladding at the end of life at steady state (Table 1). An additional 0.1% strain is allowed for what is termed expected normal operating overpower, for a total of 0.3% total permanent strain. The failure criterion¹ for a severe transient which would still not result in cladding rupture is a total cladding strain of 0.7%. Other design criteria, including stress-rupture life, are being considered.

Design of LMFBR fuel pins using these design criteria considers fission gas release under both steady and transient conditions a major cause of cladding stress. Fission gas pressures are calculated on the conservative assumption of 100% gas release. Theories and mechanisms of fission gas release have been reviewed [10, 11].

Although the above strain limit criteria appear conservative, they are not unduly restrictive in the design of initial demonstration and FFTF cores.

Another limiting design criterion is that fuel melting shall not occur under steady-state FFTF operation with 15% overpower. Uncertainties in thermal design parameters currently limit the maximum linear heat rating to approximately 17 kW/ft (560 W/cm) in order to meet this criterion. In lieu of direct measurements of gap conductance, values are inferred either from in-fuel measurements of temperatures or from postirradiation microstructure analysis. Whichever method is used, all experimental uncertainties of power, film coefficient, fuel thermal conductivity, and time and temperature for fuel restructuring contribute to the uncertainty in the gap conductance value.

Tests later described are being conducted to provide linear heat-rating-to-fuel melting information on prototypic LMFBR fuel pins as a function of fuel-to-cladding gap size.

Cladding wastage allowance provides for loss of material due to wall thickness variations in the initial cladding, wear and coolant corrosion on the external surface, and fuel-cladding chemical interaction on the internal surface. The end-of-life clad wall thickness of 0.008 inch (0.02 cm) for the FFTF is derived from the initial cladding wall thickness of 0.015

¹ "Failure" in this usage reflects a design basis and does not necessarily imply that failure will occur.

(0.038 cm) inch, less a cladding wastage allowance. Loss of mechanical properties due to mass transport of interstitial elements such as carbon and nitrogen is accounted for in the mechanical properties values used in design.

Testing of mixed oxide fuel pins in high temperature subassemblies with prototypic fuel parameters and irradiation conditions are providing confirmation of the cladding wastage allowance required for LMFBR fuel designs.

IRRADIATION EXPERIMENTS

Irradiation experiments in support of development of LMFBR fuel can be divided into three categories:

- . Thermal performance;
- . Confirmation of the adequacy of fuel pin design for steady-state reactor operation at bulk reactor outlet coolant temperatures up to 600°C;
- . Determination of fuel pin behavior under off-normal reactor operating conditions, including transients and fuel pin failures.

Irradiation testing is principally conducted in the Experimental Breeder Reactor-II (EBR-II) [7, 8, 12]. Additional important experiments are conducted in the Southwest Experimental Fast Oxide Reactor (SEFOR), the Materials Testing Reactor (MTR), the Engineering Test Reactor (ETR), the General Electric Test Reactor (GETR), the Oak Ridge Research Reactor (ORR), and the Transient Test Reactor Facility (TREAT).

Thermal Performance

Predictions of thermal performance of fuel pin assemblies in fast reactors have relied almost exclusively on calculational methods that are not fully verified by direct prototypic in-reactor tests.

Microstructural analysis requires knowledge of fuel melting or columnar grain growth temperatures as a function of burnup, oxygen-to-metal ratio, mixed oxide composition, and time. The temperature for initial columnar grain growth is difficult to establish accurately.

The large uncertainty in cladding-fuel gap conductance, which reduces the fuel pin power rating, is most restrictive for fuels early in life where the gap size is essentially dependent on the dimensions of the as-fabricated fuel.

A noninstrumented in-reactor test has been conducted in the EBR-II to obtain detailed information on the linear heat-rating-to-fuel melting in prototypic fuel pins as a function of fuel-to-cladding gap sizes. This test incorporated nineteen encapsulated pins of two diameters, 0.250 inch (0.635 cm) OD and 0.230 inch (0.584 cm) OD, and initial fuel-to-cladding diametral gaps ranging from 0.004 inch (0.01 cm) to 0.010 inch (0.025 cm). The heat rating to melting was defined by both the radius at which melting was observed and by the radius of columnar grain growth.

Instrumented tests play an increasingly important role in defining thermal performance capabilities of fuel assemblies. A test position in the EBR-II has been modified to accept special assemblies capable of direct measurements of coolant flow, coolant temperatures, fuel temperatures and plenum gas pressures. The first instrumented assembly, designated XX01, was irradiated to measure temperatures in UO₂ fuel pins [13].

A second instrumented assembly shown in Figure 1 was irradiated to measure coolant flow and temperatures within the assembly. Subchannel temperatures measured in this assembly were in good agreement with values calculated using the COBRA-IIA code [14]. This assembly also served to obtain information on the influence of experimental core loading configurations on coolant flow through the assembly [12].

A third instrumented assembly, with thermocouples in mixed oxide pellets (Figure 1), is designed to measure central fuel temperatures in eight of 37 pins operating at 460 W/cm. Measured fuel temperatures, with accurate estimates of linear heat rating, will provide a detailed insight into gap conductance values used in fuel pin heat transfer calculations. Central fuel temperatures are measured in annular pellets of pins with fuel-to-cladding gaps ranging from 0.004 inch (0.01 cm) to 0.010 inch (0.025 cm). Sibling fuel pins with annular pellets or solid pellets are to be irradiated concurrently in this assembly. Coolant inlet and outlet temperatures are measured to provide data on fuel pin heat rating. Six of the fuel pins are spaced by thin walled hypodermic needle tubing containing iron flux monitor wires. Activation data from the iron wires will assist in establishing the linear heat rating of the test fuel pins.

Steady-State Irradiations

Current irradiation testing in EBR-II emphasizes testing in assemblies of 37 and 61 pins. High risk experiments are irradiated in assemblies containing 19 encapsulated pins and in fail-safe vehicles that minimize the risk of failure propagation and excessive fuel loss.

The number of mixed oxide fuel pins under test in the EBR-II increased rapidly during the past year. Irradiation was completed on a total of approximately 700 pins by March 1971 (Figure 2).

Figure 3 shows important fuel, cladding and irradiation parameters of more than 1200 fuel pins included in the current overall EBR-II irradiation test program. Annealed and cold worked stainless steel cladding of AISI Type 304, 321, 347 and 316 are tested as well as fuel in the form of sintered and annular pellets, and compacted powder or microspheres. The oxygen-to-metal ratio of test fuels is principally in the range of 1.95 and 2.00 although some hyperstoichiometric fuels have been tested. Fuel planar smear density, an important design variable, has varied from 76% to more than 95%. Fuel pin linear heat ratings have varied from 160 to 660 W/cm. Low power - high fluence conditions expected to produce maximum cladding swelling and overpower conditions to deliberately test the effect of fuel melting are included.

Peak cladding temperatures which have ranged from 450 to 730°C, are to be extended to 760°C using special EBR-II fuel assembly hardware. The fuel parameters and irradiation conditions of two high temperature tests are shown below. Subsequent tests will include fuel with O/M ratios from 1.94 to 2.00.

	Test No. 1	Test No. 2
Subassembly Type	EBR-II Mk-J	EBR-II Mk-J
Number of Pins	37	37
Fuel	Pellets	Pellets and powder
O/M	1.95 - 1.99	1.96 - 1.97
Smear Density	85.5 $\pm$ 2.5% TD	80.9 to 88.9% TD
Stainless Steel Cladding Type	316, 20% cold worked	304, 316, 321, 347 annealed, 20% cold worked, and carbide agglomerated
Peak Cladding ID Temp.	760° C	680° C
Peak Linear Pin Power	426 W/cm	525 W/cm
Goal Exposure	8.4 at .%	10.5 at .%

The highest burnup attained in the EBR-II is 12.4 at .% for the GE F2 test group of encapsulated mixed oxide fuel pins [15]. Many unencapsulated fuel pins have reached approximately 5.3 at .% burnup.

A typical multiple pin assembly for irradiation of mixed oxide fuel in EBR-II is illustrated in Figure 4. Replicate pins with the same variables provide a statistical basis for performance analysis [12, 15].

Experiments conducted in various reactors provide data on fuel-cladding compatibility [16, 17, 18], thermal conductivity [19, 20, 21, 22], fuel restructuring [23, 24, 25, 26, 27], fuel swelling, fuel mechanical properties, fission gas release and fuel-cladding mechanical interactions to support the interpretation and analysis of the results from the EBR-II irradiation tests.

Postirradiation measurements on 76 fuel pins irradiated to approximately 6 at .% showed 0.4% $\Delta D/D$ for the Type 316 stainless clad and 0.8% $\Delta D/D$ from Type 304 stainless clad related to cladding swelling. Cladding density measurements on other encapsulated pins [31] irradiated to 4.7 at .% burnup with smeared densities of 87-89% correlate directly with the pin diameter measurements, indicating that all of the diameter increase of 0.25% for these fuel elements resulted from cladding swelling.

The results of the high burnup irradiation [28, 32] may indicate that the fuel-clad gap opens at burnups higher than 6% for annealed clad, which suggests that the fuel swelling contribution to cladding deformation may diminish with burnup. The high burnup results also indicate that fuel swelling could be restrained by thicker cladding (30 mils) and forced into gas porosity and the central void [32].

Evidence of wire-wrap/cladding mechanical interaction has been found at burnups ranging from 3 to 10% [32]. This interaction manifests itself as an apparent spiral ovality of the cladding.

Recent data [28] on fuel pins irradiated to a peak burnup of 12.4 at .% with smeared density of 84, 87 and 93% theoretical, indicate that cladding plastic deformation is related to the smeared density of the fuel. The strain in excess of swelling was 0.0., 1.6 and 3.7% respectively (Figure 5).

Evidence is increasing that loss of cladding thickness through fuel-cladding chemical interaction may impact fuel pin design more than previously thought. Thermal and fast flux irradiation tests reveal potential for attack of stainless steel cladding by irradiated fuel [16, 18, 29, 32, 33, 34 and 35]. This attack involves either a cladding matrix reaction or intergranular penetration. Significant reaction is observed only at threshold temperatures above 500 to 600°C, and apparently increases with temperature [16]. The effect of cladding composition on the reaction is not fully understood. Differences have not been detected between Type 304 and 316 stainless steels. These data also indicate that the location and distribution of carbides and nitrides within the cladding have a significant effect. Hastelloy-X appears to be resistant to attack at temperatures 50 to 100°C higher than for stainless steels [34].

Thermal flux irradiation [17] tests reveal dependence of depth and type of cladding grain boundary [18] attack on initial oxygen-to-metal ratio. Fuel pins with an initial ratio of 1.94 showed only deposits of fission products on the inside surface of the cladding, without indications of grain boundary attack. Fuel pins containing stoichiometric oxide fuel showed cladding grain boundary attack to a depth of 0.004 inch (0.01 cm), with Mo, Cs, and Te present in the cladding grain boundaries.

Fission gas release from mixed oxide fuel depends on fuel operating temperature and burnup. Gas release of 80 to 90% can be expected from high power pins (12-15 kW/ft: 390-490 W/cm) at burnups of approximately 10 at .% [32]. Releases of 100% from three pins which operated at high linear heat ratings, 16 kW/ft (525 W/cm) to burnups of approximately 10 at .% have been reported [36].

The irradiation testing program currently underway will provide additional information on the relationship of phenomena discussed above to fuel pin design.

Transient and Fuel Failure Tests

A large number of mixed oxide or UO_2 fuel pins, unirradiated and pre-irradiated in both fast and thermal neutron fluxes, have been transient tested in the TREAT facility [9, 37, 38, 39]. The results have indicated that differential thermal expansion, transient fission gas release and pressure, and molten fuel dynamics play an important role in pin failure [40, 41, 42, 43].

A program which will provide early information on the cause of fuel pin failure is being conducted in the EBR-II in special fail-safe hardware. Fuel pins previously irradiated to burnups as high as 10 at .% have been inserted into fail-safe assemblies for irradiation until failure occurs.

The behavior of failed fuel pins is explored in several series of reactor tests, as well as in out-of-reactor experiments. These tests indicate that sodium logging is not a major problem, but the oxygen content

of the fuel does influence progression of fuel failure [44, 45]. The most rapid reactions observed (corresponding to high oxygen availability in the fuel material) caused 0.060 in. (1.5 mm) diametral increase in a period of one month for fuel of typical LMFBR geometry. Deformation was reduced to approximately one-third of this value by reducing the oxygen-to-metal ratio in the fuel.

Tests to determine the behavior of fuel pins after failure are underway with the objective of permitting continued plant operation with as many as 1% of the fuel pins being nonhermetic.

CONCLUSIONS

More than 1200 test pins achieving significant exposures with essentially flawless performance in a fast flux provide growing reassurance that the mixed oxide-stainless steel fuel system is adequate for demonstration reactor and FFTF initial cores.

Exposures of more than 5.3 at .% with cladding temperatures of 625°C and resonable extrapolations of these levels appear to be technically feasible. This assures adequate performance for initial demonstration reactors and FFTF, and at least marginal performance for advanced reactor cores. Substantial incentive exists for two-fold or greater increase in exposure and coolant temperatures to approximately 600°C. The broad base of experiments now in EBR-II, and later in FFTF, provides reasonable confidence these goals can be achieved.

Specific areas of design uncertainties such as fuel-cladding chemical interaction and loss of cladding constituents in flowing sodium have been recognized and are the subject of specifically oriented programs using specially designed tests to provide directly applicable information. Current assessment of available data indicate the combined effects of these phenomena are fully within the allowances made in the fuel pin design.

Cladding strain resulting from fuel cladding mechanical interaction has been observed in high exposure tests. The role of this phenomena is acknowledged in fuel pin and assembly design, but its nature is not totally understood. Run-to-failure tests are underway to provide a better assessment of this and other factors contributing to ultimate fuel pin failure.

The degree of sophistication of special test assemblies and the quality of the data from these experiments and statistically planned irradiation tests are continuously being improved. Data from instrumented fuel assemblies and materials tests, including irradiation experiments with more than 13,000 highly characterized material specimens for mechanical properties and electron microscopy studies, form the basis for the longer range technical and economic potential of this fuel system.

Equally important to the long-range optimization of this fuel system is the intense commitment and focus of efforts of both national laboratories and industrial organizations to supporting integrated and disciplined engineering based development programs to develop fuel for commercial breeder reactors.

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TABLE I

FUEL ASSEMBLY ENGINEERING PARAMETERS

PARAMETER	UNITS	DESIGN VALUE FOR FFTF	DESIGN VALUE FOR TYPICAL DEMONSTRATION REACTOR
<u>THERMAL DATA</u>			
Core Power	MWt	400	800-1200
Fuel Assembly Power	MWt	5-7	4-8
Fuel Pin Nominal Linear Power	W/cm		
Average		230	230-260
Maximum		425	425-490
Core Coolant Temperature	°C		
Inlet		315-425	370-425
Bulk Outlet		480-595	540-595
Maximum Cladding Temperature	°C	640-750	650-705
<u>DESIGN DATA - FUEL ASSEMBLY</u>			
Fuel Assemblies Per Core	Each	73	125-200
Driver Fuel Pins Per Core	Each	15,841	30,000-40,000
Driver Fuel Pins Per Assembly	Each	217	217
Hexagonal Fuel Duct Width Across Flats - OD/ID	cm	11.621/11.011	12.192/11.938
Residence Time, Total In-Core	Full Power Days	306(1)	550-900
Refueling Cycle	Full Power Days	102(1)	180-360
<u>DESIGN DATA - FUEL PIN</u>			
Length of Fuel Pin Overall	cm	238	228-279
Triangular Pitch, Pin to Pin	cm	.726	.762-.889
Cladding Tube	cm		
OD/ID		.584/.508	.635/.559
Wall Thickness Beginning of Life		.038	.038
Wall Thickness End of Life		.020	.020-.025
Composition		SS-316, 20% CW	SS-316, 20% CW
Fission Gas Plenum, Volume	cm ³	18.52	16.39-20.48
Maximum Pressure	kgf/cm ²	52.73	49.21-56.24
Cladding Strain, (2) Steady State End-of-Life	%	0.2	0.2
Minor Incident		0.3	0.3
Major Incident		0.7	0.7
Peak Cladding Fluence	n/cm ²	2 X 10 ²³	2-3 X 10 ²³
Pin Spacing Method		Wire Wrap	Wire Wrap or Grid
Fuel Pellet			
Diameter	cm	0.24	0.54-.546
Form	--	Solid, Cylindrical	Solid, Cylindrical
Composition	Mixed PuO ₂ -UO ₂	22-26 wt% PuO ₂	12-20 wt% PuO ₂
Oxygen to Metal	--	1.96	1.95-1.99
Density	% TD	90.4	85-95
Fuel Pellet to Cladding Gap	cm	.014	.013-.018
Planar Smear	% TD	85.5	80-90
Pellet Stack Height	cm	91.44	76.2-101.6
Number Fuel Pellets Per Pin	--	150	100-200
Goal Burnup Average Pin	at.%	4.7	7.9-10.5
Peak Pin	at.%	8.4	10.5-13.2

(1) For 480°C bulk outlet conditions.

(2) These values are exclusive of irradiation induced swelling and creep.

Figure 1. Instrumented Fuel Assemblies

Figure 2. Scope of Stainless Steel Clad Mixed Oxide Fuel Irradiations

Figure 3. Mixed Oxide Irradiations in EBR-II

Figure 4. 61-Pin Test Assembly

Figure 5. Comparison of Measured and Predicted Pin Deformation Irradiated to 11 at.% Burnup