Con

ANL/CP--71374

DE91 011174

ADVANCES BY THE INTEGRAL FAST REACTOR PROGRAM

M. J. Lineberry, D. R. Pedersen,
L. C. Walters, and J. E. Cahalan

Argonne National Laboratory
9700 S. Cass Avenue
Argonne, Illinois 60439

ABSTRACT
The advances by the Integral Fast Reactor Program at Argonne National Laboratory are the subject of this paper. The Integral Fast Reactor (IFR) is an advanced liquid-metal-cooled reactor concept being developed at Argonne National Laboratory. The advances stressed in the paper include fuel irradiation performance, improved passive safety, and the development of a prototype fuel cycle facility.

INTRODUCTION
The Integral Fast Reactor (IFR) (Ref. 1) is an advanced liquid-metal-cooled reactor concept being developed at Argonne National Laboratory. The two major goals of the IFR program are improved economics and enhanced safety. The IFR program is specifically responsible for the irradiation performance, advanced core design methodology, safety analysis and testing, and development of the fuel cycle (including the fuel cycle facility) for metal fuel for the U.S. Department of Energy Advanced Liquid Metal Reactor Program. The basic elements of the IFR concept are: (1) metallic fuel, (2) liquid sodium cooling, (3) modular, pool-type reactor configuration, (4) an integral fuel cycle, based upon pyro-metallurgical processing and injection-cast fuel fabrication, with the fuel cycle facility collocated if so desired.

In the IFR concept, the liquid sodium coolant operates at atmospheric pressure, and maintains a design point margin to boiling greater than 400K (700°F). This eliminates the need for a pressurized primary system and thick-walled pressure vessels. With its high thermal conductivity and specific heat capacity, liquid metal cooling enables the IFR to operate at decay heat levels in natural circulation, without the need for forced flow. Liquid metal cooling permits a compact core configuration that complements the neutronic advantages of metal fuel and an enhanced fast neutron energy spectrum. These response characteristics are achieved by use of inherent mechanisms, hydraulic, and neutronic reactor system properties, which are determined by the choice and arrangement of reactor materials.

The most significant safety aspects of the IFR program result from its unique fuel design. A ternary alloy of uranium, plutonium, and zirconium, developed at Argonne, is based on experience gained through more than 25 years of the EBR-II reactor reactor with a uranium alloy metallic fuel.

The IFR safety approach for the Integral Fast Reactor (IFR) concept capitalizes on the characteristics of metallic fuels and of pool-type liquid metal reactors to provide enhanced safety margins. The fundamental safety approach guiding the IFR program is:

1. A simple, economic, high quality fuel system must be developed which will allow normal operation of reprocessed fuel with minimal fuel failures.
2. The metallic fuel system must be tolerant of fuel failures and local faults.
3. The reactor design must be such that for any system failure, including those in the balance of plant, no active systems have to operate to maintain the reactor in a safe state.
4. Even though the reactor can achieve passive shutdown for any system failure, the reactor should be provided with redundant and diverse safety grade scram systems.
5. The reactor should be provided with redundant and diverse decay heat removal systems designed such that at least one of them would be able to remove decay heat considering a full range of events such as a large secondary-side sodium fire.
6. To provide a level of safety consistent with the high level of safety achieved for protection from internal events, the reactor should be designed for a low level of risk from external events, i.e., earthquakes.
7. To provide defense-in-depth, the reactor/containment design must include features to mitigate the consequences of core melt accidents.
8. The fuel cycle design and reduction of the long term waste problem by actinide recycle must be an integral part of the safety posture in reducing risk from IFR operations.
9. The fuel cycle design must be responsive to the U.S. goals for proliferation resistance.

In this paper we will concentrate on the advances in Integral Fast Reactor Program in three areas 1) fuel irradiation performance, 2) safety, and 3) fuel cycle design.

IFR Fuel Irradiation Advances
Metallic fuels were the first fuels used for liquid metal cooled-fast reactors (LMR's). In the late 1960's worldwide interest turned toward ceramic LMR fuels before the full potential of metallic fuel could be achieved. Development of metallic fuels continued throughout the 1970's at Argonne National Laboratory's Experimental Breeder Reactor II (EBR-II) because EBR-II continued to

DISCLAIMER

be fueled with a metallic fuel. During the period of development in the 1970's at EBR-II, the performance disadvantages of metallic fuel were satisfactorily resolved and as well additional attributes of metallic fuel were discovered (Ref. 2). Excellent breeding performance and high burnup potential were accepted attributes of metallic fuel. An aggressive program was initiated in 1983 to prove commercial feasibility of all aspects of the IFR concept including demonstrating that U-Pu-Zr metallic fuel could meet the requirements of the IFR concept.

The rationale for the choice of the alloy U-Pu-Zr and the design aspects of metallic fuel, which allow high burnup, will be summarized in the following. A fuel that contains plutonium is required for a breeder reactor using U^{238} as the fertile material; however, plutonium and uranium-plutonium alloys have low solidus temperatures that make practical reactor designs impossible. Thus, an alloy addition was sought that would increase the solidus temperature of the U-Pu alloys. Several elements (chromium, molybdenum, titanium, and zirconium) alloy well in this system and effectively increase the solidus temperature. However, zirconium was unique because it appeared to result in enhanced compatibility between the fuel and the austenitic stainless steel cladding materials (Ref. 3). Without zirconium the cladding elements, nickel and iron, readily diffuse into the fuel to form compositions that result in lowering of the solidus temperatures adjacent to the cladding. The addition of zirconium appeared to suppress the inter-diffusion of fuel and cladding components. The allowable concentrations 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 liquidus temperatures that were too high for the fabrication of the fuel. The fabrication technique used for metallic fuel is to injection cast the fuel into the quartz molds (Ref. 4). At the end of the 1960's a plutonium based fuel alloy was partially developed that had both adequate compatibility with the cladding and a high solidus temperature.

Another problem at the end of the 1960's with metallic fuel was the perceived low reactor residence time or burnup achievable with metal fuel. A simple design change, however, allowed high burnup to be achieved with metallic fuel (Ref. 5). The first metallic fuels used in EBR-I, EBR-II, FERMI, and DFR were of high smeared density (85 to 100%) with little or no gap between fuel and cladding. When the fuel swelled from fission product accumulation the cladding deformed and failed at low burnup. Attempts at the time to extend the burnup were concentrated on alloying and thermo-mechanical treatment of the fuel to suppress swelling and the use of strong cladding. The work was largely unsuccessful with peak achievable burnups of about 3 wt%. A theory was developed and applied to fuel design that greatly extended the burnup capability of metallic fuel (Ref. 6). The primary cause of fuel swelling is the accumulation of fission product gases in bubbles where the gas pressure increases with burnup. As the bubbles grow, the surface tension is overcome and the fuel matrix flows causing swelling. It is shown theoretically that when the fuel swelling reaches about 30% the bubbles must interconnect, independent of size and/or number density. Therefore, it was postulated that if the gap between fuel and cladding were large enough to allow the fuel to swell to about 30% before fuel-cladding contact then the bubbles would interconnect, release the accumulated fission gas, and thus remove, or reduce the primary cause of fuel swelling. A large gas plenum above the fuel captures the fission gas and keeps the stress reasonably low on the cladding. It was demonstrated in the late 1960's that interconnection of porosity with subsequent fission gas release consistently occurred for smeared densities less than 75% for a range of metallic fuel alloys.

The IFR concept restored interest in metallic fuel. Additional effort remained in 1983 to demonstrate that U-Pu-Zr fuel was a commercially viable option. Many of the feasibility questions associated with the performance of metallic fuel had been answered by 1983, as discussed above, and in fact additional positive attributes of metallic fuel had been discovered, such as the robust performance during transient operation. From 1969 to 1983, no U-Pu-Zr fuel had been irradiated and there was no facility available to fabricate the fuel. The data base was weak in 1983 with only 18 U-Pu-Zr fuel pins irradiated to about 4 at% burnup. In addition to the lack of demonstration that the U-Pu-Zr fuel would reach high burnup, a number of other issues required further study for complete resolution.

A fuel performance demonstration program was designed in 1983 to gain the information required to eventually license metallic fuel. A number of assemblies were irradiated to establish the burnup potential of the U-Pu-Zr fuel, the performance of fuel pins with alternative cladding materials, the performance of fuel pins with a range of design choices such as smear density, plenum to fuel ratio, operating temperature, and linear power and finally, a series of tests to help develop the fuel fabrication specifications. In parallel to the irradiation experimentation, an analytical and out-of-pile testing program was established to fully understand the fuel performance. Fuel pin modeling codes for steady-state and transient analyses called LIFE-METAL and FPIN2; respectively, are under development (Ref. 7). Fuel-cladding compatibility is being studied by use of several techniques including an apparatus called the Whole Pin Furnace Test Apparatus, (WPT) (Refs. 7 & 8) and the Fuel Behavior Test Apparatus (FBTA). In the FBTA, small sections of irradiated fuel and cladding are heated to high temperature to determine the onset of melting at the fuel-cladding interface and the subsequent cladding penetration kinetics. The WPT subjects entire irradiated fuel pins to high temperature, with prototype axial temperature profiles, to the point of cladding breach to understand fuel failure under the combined effects of cladding creep and the fuel-cladding eutectic interaction. Property studies of the fuel system were initiated to more completely establish the phase diagrams, thermal conductivity, and redistribution of the alloying elements.

The lead assembly that contained U-Pu-Zr fuel began its irradiation in early 1985 in EBR-II and has reached a burnup of 18.4 at%. The assembly contained 61 fuel pins of the three fuel compositions; U-10Zr, U-8Pu-10Zr, and U-19Pu-10Zr, where compositions are given in weight percent (Refs. 9 & 10). The fuel pins were clad with an austenitic stainless steel alloy, D-9. Three such assemblies began their irradiation concurrently in EBR-II in 1985. Two of the three assemblies were interrupted during the irradiation cycles, and fuel pins were removed for post-irradiation testing. The third assembly was irradiated until the first cladding breach. Later in 1985, an identical assembly to the first three, with the exception that the cladding was the martensitic alloy HT-9, began its irradiation in EBR-II reaching a burnup currently of 17.0 at% burnup without cladding breach.

A great deal of information was accumulated from the post-irradiation testing of these initial assemblies. It was found that the quantity of gas released to the plenum as a function of burnup and the burnup where interconnected porosity occurred was consistent for all the fuel alloys irradiated even though the microstructure of the alloys was strongly dependent on the alloy composition. 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 being more than a factor of two larger than the axial component. Still axial growth of the fuel slug occurred where the extent of axial growth was a function of alloy composition and irradiation conditions. As expected from the earlier irradiation results from the 1960's, radial redistribution of the alloying elements was observed, specifically, U and Zr, with the Pu radial concentration profile was largely unchanged. As the radial concentration of Zr and U changed, a radial porosity distribution developed that revealed distinct rings or zones that were evident on a macroscopic scale. Up to the burnups examined, the cladding diameter changes for the austenitic cladding could be attributed primarily to irradiation induced swelling and creep with the source of stress being the gas plenum pressure in the fuel pin. Up to a burnup of 18.4 at% it appeared that any contribution to the cladding strain from fuel-cladding mechanical interaction (FCMI) was insignificant. Cladding strain data are available for the martensitic cladding up to a burnup of 16.4 at%. At that burnup no swelling is expected and the observed cladding strains were small. A number of other irradiation tests were conducted to investigate specific design variables.

A significant component of the out-of-pile test and analysis program has been directed toward the understanding of fuel-cladding compatibility. The FBTA apparatus has developed into a very effective means to address fuel-cladding compatibility. To date, a large number of irradiated fuel-cladding specimens have been tested over a wide range of cladding types, fuel compositions, burnup, temperature, and hold time at temperature. Over all conditions of test and specimen conditions the onset of melting temperature has remained above 725°C. There appear to be three different types of cladding attack. At high temperature, but below the onset of melting, there is accelerated intergranular interdiffusion of the lanthanide fission products with nickel in the austenitic cladding and much less of this effect with the martensitic cladding. This same phenomenon occurs at a lesser extent during steady-state operation at lower temperatures. At higher temperatures, above 725°C, a second phenomenon occurs when liquid metal forms at the interface between fuel and cladding, and the cladding thins along a fairly uniform front with little grain boundary attack ahead of the front. At the same time a liquid metal front moves inward into the fuel as iron from the cladding moves through the liquid phase to the fuel and deceases the solidus temperature of the fuel. Measurement of the depth of cladding penetration as a function of time at temperature allows a quantitative description of the penetration kinetics. It was found that the penetration was linear or less with time with no tendency toward a break-away characteristic. Further, the penetration rates vary exponentially with temperature, such that for temperatures near the melting temperature, the penetration rate is low. A third phenomenon that contributes to cladding wastage and seems associated with the austenitic cladding is a grain boundary attack of the cladding by the lanthanide fission products. As the austenitic cladding swells away from the fuel at high burnup and leaves a gap between the fuel and cladding, the lanthanide fission products accumulate in blocky phases. When the specimen reaches high temperature, the lanthanides tend to penetrate the grain boundaries ahead of the intergranular diffusion front. This phenomenon appears to be uniquely associated with regions on the fuel pin where the cladding has swollen away from the fuel.

Although the FBTA tests are a relatively expedient way of gaining on set-of-melting temperatures and cladding penetration data over a wide range of test variables, it is important to test entire irradiated fuel pins in order to understand the combined effects of stress and cladding attack. To date, three irradiated metallic fuel pins have been tested in the WPT apparatus at a peak temperature of 800°C. Prior to the tests, the time of cladding breach was predicted by both the steady-state modeling code LIFE-METAL and the transient code FPIN2. Both of the codes did a satisfactory prediction of the time of cladding stress failure.

IFR Safety Advances

Many of the superior safety performance characteristics of the IFR ternary alloy fuel design can be traced to its thermal and mechanical properties. A low temperature gradient across the fuel gives a correspondingly small zero power-to-full power Doppler reactivity swing, resulting in reduced control reactivity requirements and less external reactivity available for accidental insertion. The low operating temperature also yields a smaller positive Doppler reactivity input in unprotected transients on power reduction. This permits other reactivity feedbacks such as axial and radial core thermal expansion to overcome the small positive Doppler input associated with power reduction, resulting in self-adjustment of the reactor core power to equal available decay heat removal capacity in loss-of-heat-sink (LOHS) and loss-of-flow transients (LOF).

Anticipated Transients Without Scram

In the full spectrum of unprotected (unscrammed) accidents, three specific initiators have emerged to serve as quantifiers of safety margins. They are: (1) the loss-of-flow (LOF) accident, in which power to the coolant pumps is lost, (2) the transient overpower (TOP) accident, in which one or more inserted control rods are withdrawn, and (3) the loss-of-heat-sink (LOHS) accident, in which feedwater supply to the steam generators is lost. For all three initiators, it is also assumed that the plant protection system fails to insert the shutdown control rods. The events are generally classed as anticipated transients without scram (ATWS).

Avoidance of both short- and long-term core disruption in ATWS events depends on (1) providing sufficient negative reactivity feedback to overcome the power-to-cooling mismatch and return the system to equilibrium at slightly elevated system temperatures, or alternatively, (2) reducing the positive reactivity feedback components acting to resist the transition to system equilibrium. In this second respect, metallic fuel provides superior inherent safety performance in ATWS events, due to the reduced positive Doppler reactivity feedback associated with the small radial temperature gradient in the fuel (high thermal conductivity).

Full scale unprotected LOF and LOHS transients have been carried out in EBR-II (Ref. 11). These tests have confirmed the capability of the metal fueled IFR concept to respond to unscrammed accidents without core (coolant boiling or fuel failures) or system damage.

Within the framework of a cooperative effort among European and US liquid metal-cooled reactor (LMR) research and development organizations, a comparative study of the safety performance of metal and oxide-fueled core designs has been performed. Technical specialists from Germany, France, and United Kingdom, and the United States joined together in a team to analyze the response of the two fuel types during a range of accidents in a large (3500 MWt), pool-type, liquid-metal-cooled reactor design. The emphasis in the analysis was to quantify the safety margins available with oxide and metal fuels when used in conjunction with design features that supply self-limiting reactivity responses to transient power-to-flow mismatches in accidents with failure to scram. The analyses considered three accident sequences: the unprotected loss-of-flow (ULOF) sequence, in which power to all pumps is lost; the unprotected transient overpower (UTOP) sequence, in which one or more inserted control rods are inadvertently withdrawn; and the unprotected loss-of-heat-sink (ULOHS) sequence, in which all normal heat removal capability is lost. The initial conditions for all transients were taken to be the normal operating conditions (full power and flow) at the end of equilibrium burnup cycle, and all analyses were conducted on the basis of best-estimate phenomenological modeling assumptions.

The results of this comparison study reflect the basic thermal, mechanical, and neutronic performance differences of oxide and metal fuels. Metal fuel has an effective thermal conductivity that is nearly an order of magnitude larger than that of oxide fuel. This results in much lower operating temperatures in metal fuel and a lower stored energy during operation. In addition, the lower metal fuel temperature results in a lower Doppler reactivity to overcome on startup, yielding a reduced control reactivity requirement, and a reduced positive feedback to be overcome on power reduction, either upon scram or in an unprotected accident. Because metal fuel gives a harder (higher energy) neutron spectrum due to the absence of light-weight moderating isotopes, the fertile-to-fissile conversion is increased compared to oxide fuels. This further reduces the control reactivity worth requirement. The harder spectrum also leads to an increased positive coolant density reactivity and a decreased reactor Doppler feedback coefficient.

Analyses (Ref. 12) show that in unprotected loss-of-flow and loss-of-heat-sink sequences, metal-fueled liquid-metal-cooled reactors with pool-type primary systems provide larger temperature margins to coolant boiling than are available with oxide fuels. Eutectic interactions are not anticipated to cause serious problems for these transients. Thus metal fuel provides enhanced safety margins for all sizes of reactors.

Transient Fuel Performance

Under accident conditions, transient heating of metallic fuel produces cladding loading dominated by the fission gas pressure. The similarity of the fuel and the cladding thermal expansion and the compliance of the porous fuel lead to negligible Fuel-Cladding Mechanical Interaction (FCMI) cladding damage. Fuel melting in a metallic fuel element does not result in a significant clad loading because of the available porosity and small fuel density decrease on melting. The high thermal conductivity of metal fuel results in the hottest fuel being located near the core exit. Six experiments M2 to M7 have been performed in the TREAT transient reactor to determine margins to fuel pin failure, failure location, associated mechanisms and consequences and to characterize pre- and post-failure fuel relocation. A full range of fuel burnup and fuel and clad compositions are to be investigated. Tests M2, M3, and M4 were carried out using EBR-II driver fuel pins with U-5 Fissium fuel. Nine such pins were treated under slow overpower transients, with burnup and peak heating conditions being the key test parameters. Three of the pins were tested to cladding breach.

Three similar transient overpower tests (M5, M6, and M7) have been performed, using five D9-clad U-19 Pu-10Zr fuel pins with burnups up to 10 at.% and one low-burnup HT9-clad U-10Zr fuel pin. Two of the ternary fuel pins were tested to failure. Posttest analyses and examinations of the test pins from those tests have been completed. Additional TREAT tests will be performed to expand the database for IFR reference fuels to higher burnups, HT9-cladding, and to evaluate the impact of high Pu fuel.

The general results of the tests are that metal fuel has a large margin to pin failure (about 4 times nominal power in an 8 second period overpower transient), and significant molten fuel extrusion into the plenum region. In the experiments where pin failure occurred, considerable sweepout and fuel dispersal was observed without blockage formation. Fuel extrusion can provide a significant source of negative reactivity feedback in preventing severe core melt accidents (Ref. 13).

<u>IFR Fuel Local Faults Tolerance</u>

Loss of cladding integrity of a fuel element during normal steady-state full power operation should not occur during the design lifetime of the fuel because of the margins included in the design of the fuel and cladding. However, stochastic fuel element failure must be anticipated, due to a random cladding defect which goes undetected during manufacture and inspection or due to random localized thermal, hydraulic or mechanical conditions within the fuel assembly.

Metallic fuel elements have a range of features that enhance their tolerance to local fuel failure events. These features include:

a. Fuel compatibility with sodium - no chemical reaction products.
b. High thermal conductivity of metal fuel - This results in very low fuel centerline temperatures, and reduced hot-spot-temperatures for distorted geometries.
c. Low fuel clad mechanical interactions - reduction in clad loading.
d. Easy to fabricate fuel-allows easy attainment of high quality reprocessed fuel.

A significant component of the recent irradiation testing program was aimed at the performance of metallic fuel after the cladding had breached. Six Run Beyond Clad Breach (RBCB) experiments with predefected metal fuel have been completed with breach time of up to 223 days without observable fuel loss or opening of the breach site (Ref. 14). The first indication of breach is a small, but distinct peak in the DN signal as the cesium in the bond sodium is forced out of the fuel pin from the fission gas pressure. After the DN signal disappears, within a short period, some tag gas is released and continues to be released for some longer time in short bursts.

<u>IFR Fuel Cycle Development</u>

The waste management potential of the IFR concept is promising but has yet to be demonstrated. The key technical elements of the IFR fuel cycle technology are based on metallic fuel and pyroprocessing. Pyroprocessing is radically different from the conventional PUREX reprocessing developed for the LWR oxide fuel. Chemical feasibility of pryoprocessing has been demonstrated. The IFR program is developing a particularly simple fuel cycle technology called the <u>pyroprocesses</u>, so named because the three key steps are conducted at relatively high temperatures. These steps are <u>electro-refining</u>, used to separate the useful fuel materials from the radioactive fission products; <u>cathode processing</u>, which further purifies the metal product of electrorefining; and <u>injection casting</u>, which is a technology widely used to form metals and plastics into desired shapes and is used in the IFR to form new fuel rods.

Electrorefining is a chemical process that uses an electrical current to drive the chemical reactions. In the electrorefiner, electricity is used to dissolve the metal fuel into molten salt and then to transport the uranium, plutonium, and transuranics to a cathode separating them from the fission products which are left in the salt. This process has several advantages over other reprocessing schemes. One key advantage is that the transuranic fission products, which can be harmful for millions of years, are not separated from uranium and plutonium. Therefore, they automatically return to the reactor where, in the IFR, they fission to produce power, rather than become troublesome waste. The fission products, which do become waste, are only harmful for hundreds of years.

Cathode processing separates the electrorefiner process fluids (i.e., salt and cadmium) from the uranium and plutonium product. This process uses high temperatures to vaporize the cadmium and salt and separate them before melting the uranium and plutonium into metal ingots. The salt and cadmium are condensed, collected and recycled back to the electrorefiner.

The ingots from the cathode processor are combined with zirconium in an injection casting furnace. The casting furnace melts the metal and injects it into molds. After cooling, the metal is removed from the molds, inspected, and reassembled into new fuel elements which are then bundled into fuel assemblies and transferred to the reactor.

The next major step in the IFR development program will be the full-scale pyroprocessing demonstration to be carried out in conjunction with EBR-II. IFR fuel cycle closure based on pyroprocessing can also have a dramatic impact on the waste management options, and, in particular, on the actinide recycling.

For discussion of high-level waste management, it is convenient to categorize the nuclear waste constituents into two parts: fission products comprised of hundreds of various isotopes, and actinides comprised of uranium and the transuranic elements--neptunium, plutoniuim, americium, curium, etc.

In a time span of the order of 200 years, the fission products decay to a sufficiently low level that their radiological risk factor drops below the cancer risk level of their original uranium ore. Actinides, on the other hand, have long half-lives and their radiological risk factor remains orders of magnitude higher than that due to fission products for tens or hundreds of thousands of years. From this point of view, therefore, there is a strong incentive to separate actinides and recycle them back into the reactor for in-situ burning.

The benefit is in the fact that the effective lifetime of the nuclear waste is reduced from millions of years to about 200 years. This would have an enormous impact on assuring the integrity of high-level waste for its lifetime and ultimately on the public acceptance of the nuclear power. But even if the actinides are removed and the lifetime of the high-level waste is reduced to hundreds of years, the need will remain for a geological repository.

Summary

The IFR program can be summarized as follows. Nuclear power must emphasize safety during: 1) fuel manufacture, 2) power production, and 3) eventual long term waste management. The IFR program responds to the demands in all these areas.

Acknowledgements

This work was performed under the auspices of the U.S. Department of Energy under contract W-31-109-Eng-38.

References

1. C. E. Till and Y. I. Chang, "The Integral Fast Reactor," Advances in Nuclear Science and Technology, 20 (1988).
2. L. C. Walters, B. R. Seidel, J. H. Kittel, "Performance of Metallic Fuels and Blankets in Liquid-metal Fast Breeder Reactors," Nucl. Technol., 65, 179 (1984).
3. C. M. Walter, G. H. Golden, N. J. Olson, "U-Pu-Zr Metal Alloy, A Potential Fuel for LMFBR's," ANL-76-28 (1975).
4. D. B. Tracy, S. P. Henslee, N. E. Dodds, K. J. Longua, "Improvements in Fabrication of Metallic Fuels," Trans. Am. Nuc. Soc., 60, 314 (1989).
5. G. L. Hofman, "Irradiation Behavior of Experimental Mark II Experimental Breeder Reactor II Driver Fuel," Nucl. Technol., 47, 7 (1980).
6. R. S. Barnes, "A Theory of Swelling and Gas Release for Reactor Materials," J. Nucl. Mater., 11, 135 (1964).
7. M. C. Billone, Y. Y. Liu, E. E. Gruber, T. H. Hughes, J. M. Kramer, "Status of Fuel Element Modeling Codes for Metallic Fuels," Proc. Int. Conf. Reliable Fuels for Liquid Metal Reactors, Tucson, Arizona, Sept. 7-11, 5-77 (1986).
8. H. Tsai, "Fuel/Cladding Compatibility in Irradiated Metallic Fuel Pin at Elevated Temperatures," 1990 International Fast Reactor Safety Meeting, Snowbird, Utah (1990).
9. R. G. Pahl, R. Wisner, M. C. Billone, G. L. Hofman, "Experimental Studies of U-Pu-Zr Fast Reactor Fuel Pins in the Experimental Breeder Reactor II," Met. Trans. A., 21A, 1863 (1990).
10. R. G. Pahl, M. C. Billone, G. L. Hofman, "Steady State Irradiation Testing of U-Pu-Zr Fuel to > 18 at.% Burnup," 1990 International Fast Reactor Safety Meeting, Snowbird, Utah (1990).
11. H. Planchon et al., "Implications of EBR II Inherent Safety Demonstration Test," Nucl. Eng. Des., 101, 75 (1987).
12. J. Cahalan et al., "Performance of Metal and Oxide Fueled Cores During Accidents in a Large Metal Cooled Reactor," 1990 International Fast Reactor Safety Meeting, Snowbird, Utah (1990).
13. J. W. Holland, A. E. Wright, T. H. Bauer, and W. R. Robinson, "Post-Test Examination of TREAT Experiments on Metal and Oxide Fuels," 1990 International Fast Reactor Safety Meeting, Snowbird, Utah (1990).
14. G. Batte and G. E. Hofman, "Run-Beyond-Cladding-Breach (RBCB) Tests Results for the Integral Fast Reactor (IFR) Metallic Fuels Program, 1990 International Fast Reactor Safety Meeting, Snowbird, Utah (1990).