Nucl Eng Technol 47 (2015) 227–239

Available online at www.sciencedirect.com

ScienceDirect

journal homepage: http://www.journals.elsevier.com/nuclear-engineering-and-technology/

Invited Article

A REVIEW OF INHERENT SAFETY CHARACTERISTICS OF METAL ALLOY SODIUM-COOLED FAST REACTOR FUEL AGAINST POSTULATED ACCIDENTS

TANJU SOFU

Argonne National Laboratory, 9700 South Cass Avenue, Argonne, IL, USA

ARTICLE INFO

Article history:
Received 9 March 2015
Accepted 10 March 2015
Available online 19 March 2015

Keywords:
Inherent safety
Metal fuel
Passive safety
SFR

ABSTRACT

The thermal, mechanical, and neutronic performance of the metal alloy fast reactor fuel design complements the safety advantages of the liquid metal cooling and the pool-type primary system. Together, these features provide large safety margins in both normal operating modes and for a wide range of postulated accidents. In particular, they maximize the measures of safety associated with inherent reactor response to unprotected, double-fault accidents, and to minimize risk to the public and plant investment. High thermal conductivity and high gap conductance play the most significant role in safety advantages of the metallic fuel, resulting in a flatter radial temperature profile within the pin and much lower normal operation and transient temperatures in comparison to oxide fuel. Despite the big difference in melting point, both oxide and metal fuels have a relatively similar margin to melting during postulated accidents. When the metal fuel cladding fails, it typically occurs below the coolant boiling point and the damaged fuel pins remain coolable. Metal fuel is compatible with sodium coolant, eliminating the potential of energetic fuel–coolant reactions and flow blockages. All these, and the low retained heat leading to a longer grace period for operator action, are significant contributing factors to the inherently benign response of metallic fuel to postulated accidents. This paper summarizes the past analytical and experimental results obtained in past sodium-cooled fast reactor safety programs in the United States, and presents an overview of fuel safety performance as observed in laboratory and in-pile tests.

1. Introduction

Next-generation nuclear energy systems currently under consideration aim for significant advances over light water reactors (LWRs) in the areas of sustainability, economics, safety, reliability, and nonproliferation. Development of these systems is an international effort, involving collaborations under the framework of the Organisation for Economic Co-

E-mail address: tsofu@anl.gov.

http://dx.doi.org/10.1016/j.net.2015.03.004

operation and Development's Generation IV International Forum [1] and the International Atomic Energy Agency's International Project on Innovative Nuclear Reactors and Fuel Cycles [2]. Studies under these programs highlight importance of the closed fuel cycle systems using fast-neutron reactors to meet especially the sustainability goal through efficient resource utilization.

The current path relying heavily on LWRs in a "once-through" fuel cycle is not sustainable. The LWRs extract energy from only a small fraction of the natural uranium in the fuel. Used LWR fuel contains more than 95% of the original uranium, as well as heavier elements that are usable as fuel. By comparison, fast reactors can extract about 2 orders of magnitude more energy from the same amount of fuel. The once-through fuel cycle with no provision for recycling of used fuel also places a heavy burden on the spent fuel repository storage capacity. In a fast reactor with a closed fuel cycle, the transuranic elements that remain radioactive for a long time can be consumed, significantly reducing the time horizon and repository space needed for waste isolation.

Although reconsidered as part of the next-generation nuclear reactors, the fast-spectrum systems, particularly the liquid sodium-cooled fast reactors (SFRs), are not new concepts. Since the 1950s, SFR technologies have been pursued and demonstrated worldwide, leading to the construction and operation of several experimental and prototype fast reactors in the United States [U.S.; Experimental Breeder Reactor (EBR)-I and -II, FERMI, and Fast Flux Test Facility (FFTF)], Soviet Union (BR-10, BOR-60, BN-350, and -600), UK (DFR and PFR), France (RAPSODIE, Phénix, and Superphénix), Germany (KNK and SNR-300), and Japan (JOYO and Monju). These fast reactors have achieved ~400 reactor-years of accumulated operation experience. There is again a renewed interest in SFRs with the restart of Monju in Japan, completion of CEFR in China, PFBR in India, and BN-800 in Russia, and recent design efforts for PGSFR in South Korea, ASTRID in France, and BN-1200 in Russia.

2. SFR designs and their impact on safety

Fast reactor concepts are typically classified by their coolant: (1) SFRs, (2) lead (or lead–bismuth eutectic)-cooled fast reactors, and (3) gas-cooled fast reactors. All three concepts share some basic principles. They do not need neutron moderators (water or graphite), resulting in a "fast" (or "hard") neutron energy spectrum compared to "thermal reactors" (LWRs and HTGRs). They take advantage of the larger fission-to-capture cross-section ratio and the greater number of neutrons released from the fission reaction at high energies for improved neutron economy, allowing greater flexibility of material selection (such as stainless steel for cladding and structures). The fast spectrum leads to an order of magnitude longer neutron mean-free path, resulting in a much greater sensitivity to neutron leakage and minor changes in core geometry. Negligible spatial self-shielding due to longer mean-free path also implies that reactivity perturbations impact the core as a whole, not just locally. The fast neutron spectrum can be used for breeding and transmutation of transuranic waste products, allowing a long core life in "breed-and-burn" concepts.

Among the three fast reactor concepts, SFRs are by far the most common type. Their main characteristics include a high core outlet temperature (typically > 500°C) leading to a greater thermal efficiency (~40%) for energy conversion; ability to use electromagnetic pumps (with no moving parts) and electromagnetic flow instrumentation; high core power density (~5 times greater in comparison to an LWR); and an intermediate heat transport system that separates the activated coolant in the primary heat transport system from the balance of the plant. They can be configured either as a loop type (with the primary coolant pumped out of the reactor vessel into the intermediate heat exchanger located inside the containment) or a pool-type system (with the primary coolant kept within a larger reactor vessel that encompasses the core, intermediate heat exchanger, and coolant pumps). In both the loop and pool-type systems, a guard vessel surrounds the reactor vessel as a secondary barrier against primary sodium leakage.

In an LWR, water acts as both a coolant and a moderator with an optimal pitch/diameter (*P*/*D*) ratio that provides sufficient cooling capability while allowing neutrons to slow down to "thermal" energies through collisions with hydrogen and oxygen atoms. In an SFR with no neutron moderation, sodium acts only as a coolant. Because of the excellent heat transfer properties of sodium coolant (or liquid metals in general), fuel pins can be packed much tighter in a hexagonal lattice (triangular pitch), typically separated only by a thin wire spirally wrapped around each fuel pin to prevent pin-to-pin contact.

The safety advantages of SFRs include low-pressure primary and intermediate coolant systems; liquid–metal sodium coolant with about 2 orders of magnitude more effective heat transfer medium compared to water; wide margin to boiling (~400°C during normal operation); inherent safety with "net" negative reactivity feedback during accidents that lead to elevated core/coolant temperatures; dedicated systems for emergency decay heat removal to an ultimate heat sink; simpler operation and accident management providing a long grace period for corrective action, if needed.

The low-pressure primary and intermediate heat transport systems eliminate the LOCA concern and the need for high-pressure coolant injection systems. Instead, the SFR designs typically include a guard vessel (and guard pipes in a loop type system) to "maintain" the primary coolant inventory. Low pressure in the primary heat transport system also results in low design pressure for the containment, mostly against heat from potential sodium fires. The large coolant temperature increase during flow through the reactor core (150°C in an SFR vs. ~30°C in an LWR) and the large thermal expansion coefficient of liquid sodium facilitate reliance on completely passive systems (driven only by natural circulation) for decay heat removal.

However, the SFRs also pose unique design challenges. Fast reactor cores are not in their most reactive configuration and, consequently, the design must ensure that a recriticality does not occur during a postulated accident that could lead to fuel failures. For designs with larger cores, sodium void worth can be positive resulting in a reactivity addition if the temperature of the sodium coolant reaches its boiling point during postulated accidents. The liquid sodium coolant also reacts with air and water, and ablates concrete: avoiding the impact of such reactions on the systems, structures, and components important to safety motivates reliance on leak-tight coolant systems, inert cells, guard pipes, and steel liner over concrete surfaces.

3. SFR fuel types and design challenges

The decision on fuel type can be based on many criteria including its irradiation performance, fabrication, safety, and fuel cycle implications. Common SFR fuel types include oxide, metal alloy, nitride, and carbide fuels, but large irradiation and safety testing experience exists only for oxide and metallic fuels. The original choice for fast reactors was high-density metal fuel to facilitate breeding. The earliest fast reactors including EBR-I (which generated first usable nuclear electricity in 1951), EBR-II (1963), Fermi (1963), and DFR in the UK (1959) all used metal fuel. However, these early designs achieved only limited fuel burnup because of greater fuel swelling in fast spectrum. Consequently, the oxide fuel type was adopted in FFTF and Clinch River Breeder Reactor projects in the U.S., and the rest of the world followed suit. Subsequent fuel testing in EBR-II from 1963 to 1994 as part of U.S. Department of Energy's Advanced Liquid Metal Reactor and Integral Fast Reactor programs demonstrated that the burnup limitation could be overcome by allowing adequate space inside the pin to accommodate initial fuel swelling.

The oxide fuel is fabricated in the form of uranium- or mixed-oxide sintered (ceramic) pellets similar in design to an LWR oxide fuel. The fuel-cladding gap and fission gas plenum is initially filled with an inert gas. The oxide fuel irradiation experience comes from the fast reactors in the U.S., France, Russia, and Japan. The metal fuel is injection cast as binary (U–Zr) or ternary (U–Pu–Zr) alloys of full-length slugs in SS (316) or advanced alloy (D9, HT9, HT9M) cladding. The gap between the fuel slug and the cladding is filled with sodium (often referred to as "bond sodium") to improve the gap conductance. The metal fuel also features a larger fission gas plenum to achieve high burnup. The metal fuel irradiation experience comes from irradiation experiments in EBR-II and FFTF reactors in the U.S.

Greater fuel swelling in the fast spectrum makes the Fuel-Cladding Mechanical Interaction (FCMI) a challenge especially for oxide fuel, limiting the burnup that can be achieved with it. The metal alloy fuel, by contrast, is prone to fail primarily because of Fuel-Cladding Chemical Interaction (FCCI). Because FCCI is a highly temperature-dependent process, it may limit the coolant outlet temperature of the metallic fuel core, but it does not impose a major burnup limit for the metal alloy fuel forms. Another important consideration between these two major fast reactor fuel types is the fuel–coolant compatibility: oxide fuel chemically reacts with the sodium coolant, imposing stricter limits on fuel pin failures to prevent potential flow blockages as a result of it.

Despite these differences, sufficient irradiation and safety testing experience exists with both oxide and metal fuel types, justifying their selection as the fuel for an SFR design. Both with oxide and metallic fuels, acceptable performance and reliability has been demonstrated up to 10% burnup, with capability demonstrated up to 20% burnup. This is in contrast to a few percent typically achieved in an LWR. Robustness of oxide fuels against over-power transients up to four times the nominal power has also been demonstrated in Transient Reactor Test Facility (TREAT) tests (well above typical primary and secondary safety system trip settings), indicating that typical performance issues are FCMI-induced creep rupture of cladding at high burnup, with failures occurring usually around the core mid-plane. Robust over-power capability of metallic fuel has been demonstrated in TREAT tests for up to five times the nominal power, with the failures occurring near the top of the fuel column due to accelerated FCCI. For metallic fuels, the performance and phenomena with the various alloy forms (U–Fs, U–Zr, and U–Pu–Zr) are found to be similar, with burnup, temperature, and cladding type being the key differentiators.

3.1. *Metal alloy fuel design*

The metallic fuels are developed at Argonne based on experience gained through 20+ years operation of EBR-II reactors and additional irradiation experiments in FFTF reactors [3]. Metal fuel is injection cast as full-length cylindrical slugs as shown in Fig. 1. The EBR-II fuel pin design is also shown in Fig. 2. Early experience with trying to restrain swelling of the metal fuel with strong cladding was not successful, and it limited the burnup that can be achieved with it. The key to overcome this limitation was the discovery that, although its soft structure allows metal alloy fuel to swell easily, the total swelling is limited to the swelling at only a few percent burnup

Fig. 1 – The full-length metallic Experimental Breeder Reactor (EBR)-II fuel in casting furnace (top) and after casting (bottom).

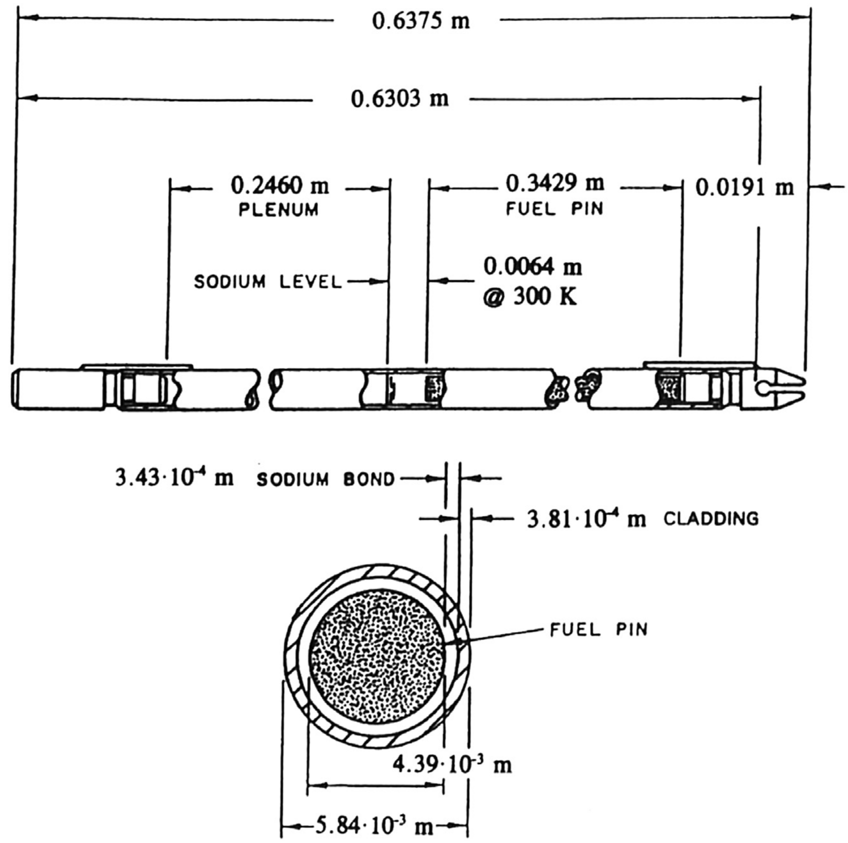

Fig. 2 – The metal alloy fuel design used in Experimental Breeder Reactor (EBR)-II reactor. *Note.* From "Performance of metallic fuels and blankets in liquid–metal fast breeder reactors," by L.C. Walters, B.R. Seidel, and J.H. Kittel, 1984, *Nuclear Technology*, 65, p. 179–231. Copyright 20XX, *Copyright Holder*. Reprinted with permission.

as shown in Fig. 3. After the initial few percent burnup, the interconnection of pores in the fuel matrix allows venting of fission gas to the pin plenum avoiding further swelling. Therefore, by allowing sufficient room inside the cladding to accommodate this initial swelling, the FCMI limitation to achieve higher burnup was eliminated for the metal fuel forms.

Multiple iterations of metal fuel designs have evolved throughout the history of the various U.S. research, development, and demonstration programs. Initial metal fuel designs in the 1960s used unalloyed uranium or plutonium driver fuels with relatively high smear densities (the areal density of as-fabricated fuel within as-fabricated cladding). Early discovery of challenges associated with the fabrication and utilization of unalloyed fuel, however, led to the testing of various alloys in metal fuel to increase its mechanical strength, improve its chemical stability, and raise the liquidus/solidus temperatures. The first EBR-II driver fuel was the U–5Fs alloy. The fissium (Fs) was a composition of simulated noble metal fission products expected to result from the EBR-II fuel recycling process. By 1970, zirconium gained favor as a better metal fuel alloy, and it replaced Fs for use in EBR-II driver fuel. The FCCI was a significant contributor to fuel life reduction resulting from clad thinning owing to the formation of a low-melting point eutectic, and the use of zirconium significantly reduced the interdiffusion of fuel and clad components [4].

Although the second and third iterations of EBR-II driver fuel still utilized the fissium and Zr alloys, the U.S. Department of Energy's Advanced Liquid Metal Reactor program eventually selected ternary U–Pu–Zr alloy as reference fuel in

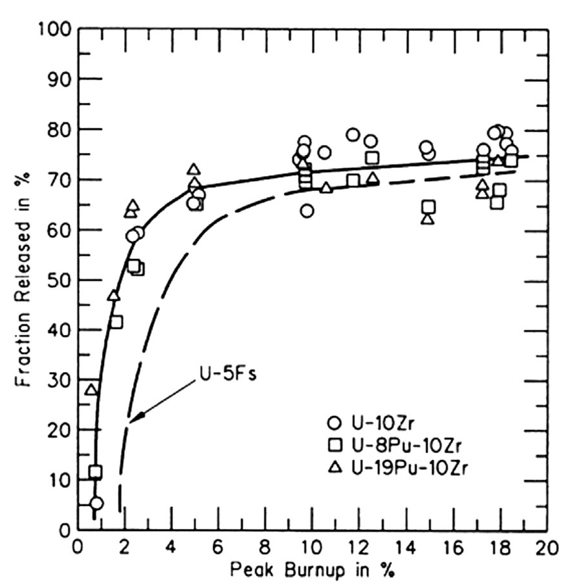

Fig. 3 – Axial and radial fuel swelling in various metal alloy fuels as a function of peak burnup.

the mid 1980s [5]. Uranium and/or plutonium alloyed with zirconium is still considered one of the most promising SFR metal fuel options, and is typically used as the reference fuel in most mature SFR designs in the U. S. and South Korea. The operating experience with metal fuel is quite substantial, and a comprehensive analysis of fast reactor fuel experience [4] suggests that the existing metal fuel database is sufficient to make a safety case for use of metal fuel in a demonstration or prototype facility, provided that the fuel composition and burnup are expected to be within the envelope of the available database.

3.2. *Comparison of irradiated metal and oxide fuel behavior*

The metallic fuel form offers some favorable neutronics properties. Specifically, the absence of oxygen atoms in the fuel leads to a harder neutron spectrum, increasing the neutron production per neutron absorbed in the pin. This occurs both because of the higher average number of neutrons emitted per neutron absorbed, and because of the enhanced fast fission in U^{238}. The combined effect increases the number of neutrons available for breeding and parasitic losses by ~20% in comparison to oxide systems. Moreover, the effective heavy metal density is higher in the metallic fuel relative to the traditional oxide fuel. For example, in an internal blanket assembly with 50% fuel volume fraction, U–10% Zr metallic pins at 85% smear density provide 35% more U^{238} atoms than do UO_2 pellets at 93% smear density. Both of these characteristics can be used to increase the core internal conversion ratio and achieve smaller burnup swing in longer refueling cycles.

A comparison of oxide and metal fuel physical properties and thermal characteristics are given in Table 1. For oxide fuel, performance and failure modes depend on various irradiation effects: (1) fuel restructuring and grain growth affect the fission gas release and fuel creep characteristics; (2) as-fabricated porosity migration is responsible for the formation of the central cavity and it affects fuel thermal conductivity; (3) fission gas retention and release affect the radial distribution of total porosity and fission gas bubble induced fuel swelling; (4) fission product swelling includes solid fission product and fission gas bubble swelling, and it affects the radial porosity profile and fuel dimensions; (5) fuel-cladding gap condition affects the fuel temperatures; and (6) irradiation-induced cladding swelling affects the cladding dimensions and density.

For transients leading to oxide fuel pin failure, the failure modes and extent of fuel disruption depend on many factors such as the fuel burnup, fuel and cladding temperatures, fuel melt fraction, molten cavity pressure, and the cladding stress. The failure modes include the plastic straining of the cladding due to internal fission gas pressure and differential expansion between the fuel and cladding, and the cladding melting due to excessive temperatures. The cladding failures result in not only the release of fission products to the primary coolant, but they could also lead to coolant channel blockages and propagation of core damage.

The important phenomena that occur during irradiation of metal alloy fuels are thermal expansion of fuel and cladding, fuel constituent radial migration, fission gas behavior, porosity formation and distribution, irradiation-induced radial and axial swelling of fuel due to solid fission products and fission gas, bond sodium migration into fuel and pin plenum, and cladding constituent migration into the fuel. As the fission gas is generated inside the metal fuel pin, it is retained within grains, in grain boundary bubbles, and in fission gas plenum above the fuel slug. The full spectrum of phenomena involved in metal fuel irradiation is summarized in Fig. 4.

Table 1 – A comparison of metal and oxide fuel properties and thermal characteristics.

	Metal (U–20Pu–10Zr)	Oxide (UO_2–20PuO_2)
Heavy metal density (g/cm^3)	14.1	9.3
Melting temperature (K)	1,400	3,000
Thermal conductivity (W/cm K)	0.16	0.023
Operating centerline temperature at 40 kW/m (K)	1,060	2,360
T/T_{melt}	0.76	0.79
Fuel-cladding solidus (K)	1,000 (eutectic limit)	1,675
Thermal expansion (1/K)	17×10^{-6}	12×10^{-6}
Heat capacity (J/g K)	0.17	0.34

Postirradiation examination of metallic U–Pu–Zr fuel pins shows the formation of annular zones with considerably different alloy compositions, fuel porosities, and densities as shown in Fig. 5. Uranium migrates from the central and outer zones to the middle zone, whereas Zr and fission products tend to migrate in opposite directions. The resulting zonal densities can vary from 8 g/mL in the central zone to 16 g/mL in the middle zone. The Zr depletion in the middle zone also reduces the melting temperature significantly and impacts the thermophysical properties.

As the steady-state irradiation proceeds, fission gas is initially retained in the metal fuel matrix. Grain boundary bubbles form, producing early fuel swelling. At burnup levels higher than a few percent, the grain boundary bubbles get interlinked, opening up a path for release of fission gas to the pin plenum. As a result, fission gas-driven swelling, thus the cladding stress due to FCMI, stays limited. Although the cladding stress owing to internal fission gas pressure continues to increase with burnup, it remains as only a small fraction of the stress from FCMI in oxide fuels.

By contrast, uranium in metallic fuels interacts chemically with iron-based cladding to form a low-melting-point eutectic alloy. In equilibrium, the liquid eutectic phase has been observed at a temperature as low as 1,000 K for an alloy that is 89% uranium. This liquefied region at the fuel/cladding interface is formed only if the fuel slug is in contact with the cladding, the contact temperature is sufficient to cause eutectic alloy formation, and the temperature remains elevated long enough to sustain the eutectic formation. The primary importance of the eutectic formation is the thinning of the cladding, reducing its ability to contain the internal pin pressure from the accumulated fission gas. Although a very slow process at low temperatures, eutectic formation can lead to accelerated cladding failure at elevated temperatures.

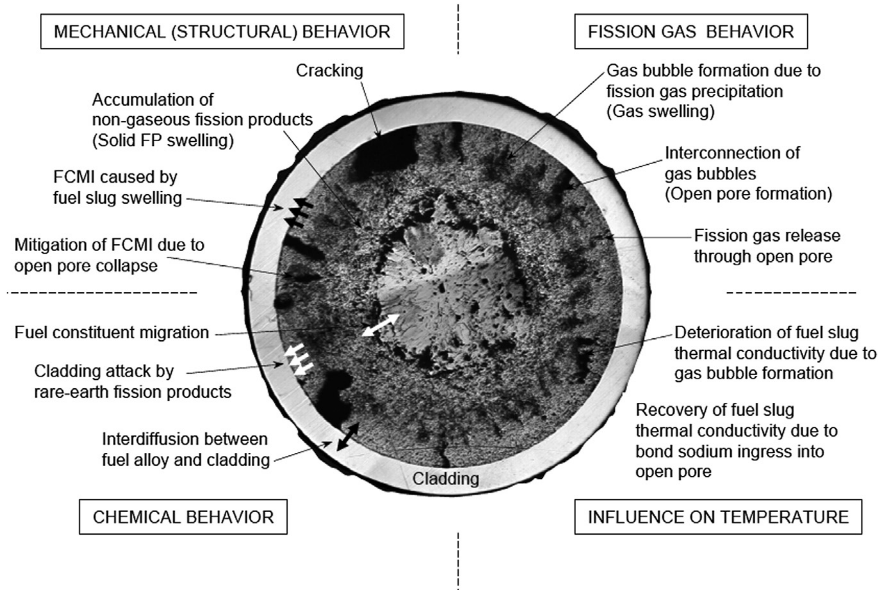

Fig. 4 – Phenomena involved in metal fuel irradiation [18].

4. Metallic fuel safety performance

The enhanced safety focus of the SFR designs aims for reliance on inherent processes to provide favorable neutronic feedback and passive systems for reactor cooling in response to accident initiators. The idea is to take advantage of the intrinsic design characteristics to maintain the balance between reactor cooling capability and power production and to prevent fuel failures in instances when engineered safety systems fail. These response characteristics are heavily influenced by the choice of reactor materials, most importantly the fuel. Performance of metallic fuel in normal and accident situations is a direct result of its favorable thermal, mechanical, and neutronic properties. As explained below, these properties assure optimal safety response in design basis accidents, anticipated transients without scram, and severe accidents, as well as in normal operation where local faults can contribute to cladding failures.

4.1. *Thermomechanical and neutronic safety aspects of metal fuel*

Many of the safety performance characteristics of the binary and ternary metal alloy fuel designs can be traced to their thermal and mechanical properties, with the most important of these being the high thermal conductivity. At operating temperatures, a typical fresh fuel has a thermal conductivity of ~0.16 W/cm K, almost an order of magnitude greater than oxide fuel. This means a much lower radial temperature rise from its outer surface to fuel centerline at operating conditions (< 200 K). As the fuel is irradiated, it swells into contact with the cladding, displacing the initial gap bond sodium and establishing even better fuel cladding thermal contact. The small temperature gradient across the fuel radius and the low operating temperature lead to a correspondingly small zero- to full-power Doppler reactivity swing, resulting in reduced control reactivity requirements

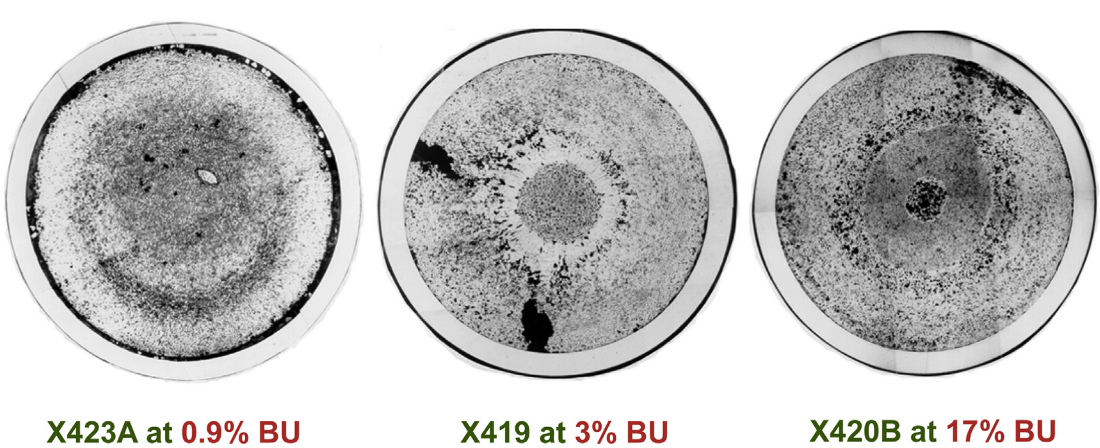

Fig. 5 – Postirradiation examination of metallic U–Pu–Zr fuel pins at different burnup levels showing formation of annular zones due to constituent migration.

and smaller external reactivity available for accidental insertion.

The peak operating temperature for metallic fuel is ~1,000 K. This comparatively reduced system heating allows increased time for operator action to correct flow or cooling deficiencies. Although the melting temperature for metallic fuel is relatively low (~1,400 K), the over-power margin to fuel melting point is approximately the same as the oxide fuel as shown in Table 1. The phenomena depending on diffusional rate processes, such as creep and fission gas release, are also similar for the two fuel types.

Under accident conditions, transient heating of metallic fuel produces cladding loading dominated by the plenum pressure. Typically, the larger fission gas plenum in metal fuel pins plays a major role in delaying the cladding failures because it limits the internal pin pressure during postulated accidents. The similarity of the fuel and the cladding thermal expansion, and the compliance of the porous and soft fuel matrix lead to negligible FCMI-induced cladding damage. Although the FCMI stresses the cladding early in the transient, little or no plastic strain accumulates before the fuel creep relaxes the cladding load to a state that follows only the increase in pin plenum pressure.

If an accident sequence proceeds to fuel melting, the high fuel porosity, low gas retention, and small fuel density decrease during melting help avoid overpressurization of the pin. When melting at the top of the fuel column allows molten fuel to expand into the plenum region, in addition to delaying the fuel failure (as demonstrated during the TREAT overpower transient tests), this in-pin molten fuel extrusion into pin plenum can provide a significant source of negative reactivity feedback.

Metallic fuels interact metallurgically with iron-based cladding materials, making FCCI the dominant failure mode. During normal operations, the rate of solid-state interdiffusion is no greater than the wastage in ceramic oxide fuel pins due to fission product attack of the inner cladding wall. At elevated temperatures, however, cladding penetration by liquid fuel-cladding eutectic becomes a major contributor to cladding failure. The low melting point eutectic weakens the cladding by thinning the wall, but the small pin pressure expected during anticipated operational occurrences and most design basis accidents limits the impact.

The harder neutron spectrum with the metallic fuel form has two important effects on reactivity feedback coefficients. The negative Doppler reactivity coefficient is reduced by about a third relative to oxide systems. The positive sodium density coefficient also becomes more positive by about a third. Because the radial temperature gradient is smaller, however, the component of the power coefficient vested in the coolant temperature rise is larger than what is vested in the fuel temperature rise for metal fuel. This partitioning of the power coefficient components (which is opposite to that of oxide fuel) contributes to the favorable inherent response attainable in an SFR with metal fuel. During the anticipated transients without scram, the smaller Doppler feedback with metal fuel also permits other naturally occurring negative reactivity feedbacks, such as axial and radial core thermal expansion, to overcome the positive Doppler component at reduced temperatures, resulting in self-adjustment of the reactor core power to match the available decay heat removal capacity as explained below.

4.2. *Response of metal fuel to design basis events and anticipated transients without scram*

The design basis events have routinely been evaluated for the metal-fueled SFR concepts to demonstrate that the consequences of such events are well within conservatively interpreted acceptance guidelines. The passive safety capability of the pool configuration provides large margins during design basis accidents. In pool systems, the large primary system heat capacity buffers the primary system so that no reactor scram is required for an array of balance-of-plant faults.

In the “Anticipated Transients Without Scram” spectrum, three specific scenarios serve as quantifiers of safety margins: (1) unprotected loss-of-flow (ULOF) accident, in which power to the coolant pumps is lost; (2) unprotected transient overpower (UTOP) accident, in which an inserted control rod is accidentally withdrawn; and (3) unprotected loss-of-heat-sink (ULOHS) accident, in which feed-water supply to the steam generators is lost.

For all three scenarios, it is assumed that the plant protection system fails to insert the control rods. The key to successful prevention of core disruption under these conditions is to limit the mechanisms leading to reactor damage, and to promote the mechanisms responding to the upset condition and acting to restore the reactor power production/cooling balance.

In all three classes of unprotected accidents, the key to avoidance of short-term core disruption is to maintain the coolant outlet temperature below its boiling point. Under normal operating conditions, the core inlet temperature is ~600–650 K, and the average coolant temperature rise through the core is ~150 K. To keep the core coolant temperatures below the sodium boiling point at ~1,200 K, the power-to-flow ratio must be typically kept below ~4.

In the long term, the net negative reactivity feedback tends to bring the reactor power into equilibrium with the available heat rejection rate, and the system approaches an asymptotic temperature distribution. To avoid core disruption in the long term, therefore, it is necessary that the peak asymptotic temperatures in strategic components (reactor vessel, core support structure, fuel, and cladding) are maintained below the levels at which creep could cause failures.

Avoidance of both the short- and long-term fuel failures during the unprotected events depends on; (1) providing sufficient negative reactivity feedback to overcome the power-to-cooling mismatch and return the system to equilibrium in a subcritical state and (2) reducing the positive reactivity feedback components (such as Doppler feedback) acting to resist the transition to system equilibrium. In this second respect, metallic fuel provides better inherent safety performance than oxide fuel owing to the reduced Doppler reactivity feedback that turns positive as the system approaches equilibrium.

For the ULOF accident, the assumed initiator is loss of power to the primary and intermediate coolant pumps without scram. As the flow decreases, the core temperature rises and the expansion of the core radially and axially causes negative reactivity feedbacks that reduce the reactor power. As the power falls, the coolant outlet temperature also begins to decrease with some delay. For an optimally designed metal-fueled pool-type SFR with adequate pump coast-down characteristic, coolant boiling can be avoided with a substantial margin. With sufficient emergency decay heat removal capacity, system temperatures should also remain below levels at which load stress-induced creep could result in structural failures as the system approaches the equilibrium state.

For the UTOP accident, the assumed initiator is an uncompensated withdrawal of a single, maximum-worth control rod. In a metallic-fueled core with a small burnup reactivity swing, the withdrawal of a single rod should amount to an insertion of smaller amount of reactivity in comparison to an oxide-fueled core. In the resulting transient, the reactor power rises above nominal levels, followed by heating of the core and coolant that introduces sufficient negative reactivity to return the reactor power gradually to equilibrium with the rate of heat rejection. For an optimally designed metal-fueled pool-type SFR with sufficient decay heat removal capacity, the low control rod worth results in manageable overheating of the primary coolant system with no fuel failures.

For the ULOHS accident, feed-water supply to the steam generators is lost, yielding a gradual heating of the intermediate and primary coolant systems and an increase in the core inlet temperature. Heating of the core support grid spreads the core radially, introducing negative reactivity that reduces the reactor power. In the long term, the reactor power equilibrates to any available heat sink with the inlet temperature elevated above the initial state. For an optimally designed metal-fueled pool-type SFR with sufficient emergency decay heat removal capacity, the negative reactivity feedbacks reduce the reactor power as the core inlet temperature rises, with peak temperatures only slightly elevated above nominal conditions.

The unprotected ULOF and ULOHS transients from full power have been carried out in EBR-II, confirming the capability of the metal-fueled pool-type SFR concept to respond to unprotected accidents to avoid any core upset (coolant boiling or fuel failures) or system damage [6].

4.3. *Metal fuel response to local faults*

Loss of cladding integrity of a fuel pin during normal steady-state full-power operation is not expected during the design lifetime of the fuel. However, stochastic fuel element failures can be hypothesized owing to the random cladding defects that could go undetected during the manufacturing process and inspection, or because of random and localized unfavorable neutronics (fuel loading or enrichment errors), thermal, hydraulic, or mechanical conditions within the fuel assembly. Such conditions are often referred to as local faults. The random loss of fuel element cladding integrity can lead to mass transport, releasing fission gas, bond sodium, and/or fuel and solid fission products from the fuel element into the coolant, or permitting ingress of primary sodium into the fuel element. Local fuel failure implies a failure that is initiated within a single fuel pin.

Owing to compatibility with sodium, low operating temperatures, predictable irradiation performance, and low FCMI-induced cladding loading, the metallic fuel elements offer a greater tolerance to local fuel failure events. Although there may be some limited interaction with trace oxygen in the coolant, this is significantly different from the chemical reaction that occurs between oxide fuels and sodium coolant. The characteristics that give the metal fuel good local fuel failure performance (both in terms of reduced failure frequency and diminished failure consequences) have been demonstrated during the run-beyond cladding breach tests in EBR-II.

Postirradiation examination of fuel pins after run-beyond cladding breach tests for an oxide and metal fuel pin is shown in Fig. 6. In these tests, an area of cladding was machined down to 25–50 μm, leaving < 10% of the original cladding thickness intact. After a short period of irradiation, cladding failure occurred at the machined spot for both types of fuel pins. As metallic fuel is compatible with sodium coolant, failures due to local faults can be tolerated for an extended period with proper monitoring of fission gas release. The metal fuel shown in Fig. 6 was irradiated for 169 days after failure (before the PIE was performed), and its posttest examination indicated no fuel loss into coolant or liquid or solid fission product escape from fuel pins. By contrast, oxide fuel chemically reacts with sodium coolant, and local faults can lead to the formation of reaction products with fuel loss into coolant and potential fuel coolant channel blockages. Therefore, oxide-fueled SFR designs require a rigorous fuel failure detection program.

4.4. *Metal fuel response to severe accidents*

The possibility of widespread core melting and disruption in SFRs can be significantly reduced by designing and constructing essential equipment to be highly reliable, and by providing redundant and diverse scram systems. In addition, because of their unique reactivity feedback characteristics, metal fueled SFR systems can be designed to avoid core meltdown under anticipated transient conditions (loss of flow, loss of heat sink, and transient over-power) even without scram. Furthermore, because of the low stored energy in metallic fuel, these systems can survive a sudden and complete rupture of any part of the core coolant system if a normal scram is accomplished. The probability of core meltdown in SFR systems, therefore, can be made exceedingly small.

Historically, the strategy for demonstrating a low risk of core disruption in the SFR design concepts during the multiple failure events has been: (1) to provide reactor design features to enhance passive safety response and to mitigate the consequences of core disruption accidents, (2) to perform analyses of accident scenarios with experimentally validated analysis techniques to quantify margins to core disruption, and (3) to account for the uncertainties associated with the frequency of accident initiators and the reliability of passive safety mechanisms.

Despite all possible design measures taken, a theoretical possibility of a core disruption (e.g., from a complete and sudden loss of flow without scram or from complete, long-term loss of all decay heat removal systems) remains. Work to date has revealed three characteristics of particular

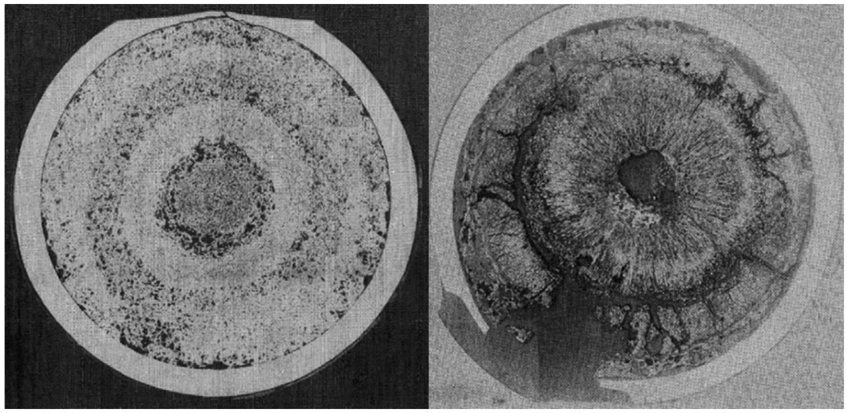

Fig. 6 – Postirradiation examination of fuel pins after run-beyond cladding breach tests. Left figure is for a metal fuel pin at 12% burnup; right figure is for an oxide fuel pin at 9% burnup.

importance to reduce the consequences of such extreme scenarios that fall under the "residual-risk" category: (1) molten metallic fuel have favorable dispersal characteristics giving rise to a powerful reactivity shutdown mechanism and (2) resolidified molten metal fuel debris is highly porous, leading to coolable configurations by natural circulation and in-vessel retention. The energetic potential of a metal-fueled SFR meltdown accident is expected to be much more modest than that of an oxide core.

Because the metal alloy fuel melting point is well below the cladding melting point, in many cases fuel can be expected to melt almost entirely before the cladding fails. Once the molten fuel comes into contact with the cladding, however, it leads to rapid eutectic penetration and cladding failure. Owing to the high fuel thermal conductivity, the axial profile of the fuel-cladding interface temperature closely follows the coolant temperature profile. Therefore, the FCCI-induced cladding failures are consistently near the top of the fuel column, where the coolant and cladding temperatures are the highest. The metal fuel also first melts well above the core midplane, facilitating in-pin molten fuel motion into the fission gas plenum, sometimes even before the cladding fails. This in-pin molten fuel motion can be a significant negative reactivity contribution.

When the cladding fails, the internal fission gas pressure forces the eutectic mix into the coolant channel (again, near the top of the fuel column) and out above of the core. Because the eutectic mix temperature is close to the sodium coolant temperature (and it could be less than the sodium boiling point in some transients), it does not refreeze or create blockages, and exits the core upward resulting in a net reactivity loss. The upward motion of the eutectic mix (out of the active core region) has been demonstrated in transient overpower tests at TREAT as explained in the next section. A similar upward motion is also predicted at reduced flow rates under ULOF conditions because of fission gas expansion, but there are no tests to validate this prediction yet. As metal fuel is chemically compatible with the sodium coolant, the phenomenon of eutectic mix entering into the coolant channel is fairly benign. It does not lead to energetic fuel–coolant reactions as oxide fuel does, and the damaged fuel assemblies remain in a coolable configuration.

5. Metal fuel experiments

Laboratory experiments and in-pile tests provide the phenomenological basis for understanding the metallic fuel safety performance. The knowledge obtained from these experiments forms the basis for theoretical models used in analyses and, therefore, plays a central role in metallic fuel safety assessment.

5.1. *Fuel behavior test apparatus experiments*

The Fuel Behavior Test Apparatus (FBTA) experiments tested the response of short irradiated fuel pin segments to thermal transients [7,8]. The apparatus had the capability of heating the segments by direct electrical heating or by external radiant heating. The FBTA tests were designed to study the effects of retained fission gas, and the metallurgical interaction of fuel and cladding. External heating was adequate because the small radial temperature gradients from volumetric heating of metallic fuels were not important to the phenomena studied.

The goal of studying the effects of retained fission gas was to understand its role in affecting short-term fuel swelling and fuel column elongation as they relate to fuel integrity and reactivity feedback. The specific objectives of the test program were as follows: (1) to assess the amount of fission gas retained in SFR fuel as a function of fuel structure established by the original fuel composition, constituent migrations, and axial location in the fuel rod; (2) to determine the mechanisms influencing fission gas behavior in the fuel with respect to local fuel swelling and gas release; and (3) to provide empirical correlations and the mechanistic basis for modeling the effects of retained fission gas.

The goal of the fuel-cladding compatibility tests was to develop a reliable predictive capability for the FCCIs in the time–temperature regime corresponding to off-normal conditions. The specific objectives of the test program were as follows: (1) to determine the rate at which irradiated fuel and cladding interact as functions of time, temperature, and fuel composition (burnup) and (2) to use these data to determine a kinetic relationship for such interactions that might lead to a reduction in fuel life or to severe fuel damage.

5.2. *Whole pin furnace tests*

The Whole Pin Furnace experimental facility was established to perform out-of-reactor tests on irradiated intact fuel pins to determine cladding failure margins during their exposure to elevated temperatures for up to 24 hours [9,10]. These conditions are typical of the later stages of protected or unprotected loss-of-heat-sink events when the power is at or near decay heat levels and fuel temperatures are nearly isothermal. The period of 24 hours is based on a reasonable time to manually scram the reactor and/or to establish auxiliary cooling. Fuel failure tests of this duration complemented the TREAT tests (discussed later) by extending cladding failure data into lower temperature regimes. These data are particularly important for HT9 clad metallic fuel pins, because the ferrite-to-austenite phase change at ~1,175 K precludes the extrapolation of high-temperature TREAT test results. The whole pin furnace tests were also designed to complement the FBTA eutectic penetration rate tests by investigating cladding failure at burnups where the plenum pressure is sufficient to rupture the cladding by itself or in conjunction with simultaneous wall thinning by eutectic penetration.

In the whole pin furnace tests, single, whole, EBR-II-irradiated fuel pins were heated in a radiant heater system. The furnace was capable of establishing a suitable axial temperature profile along the cladding length. Test parameters were peak cladding temperature (in the 975–1,225 K range), burnup, and different fuel-cladding material combinations. Most of the tests resulted in cladding failure as expected to occur in the range from a few minutes to hours. The tests typically used a ramp-and-hold temperature history with a fixed ramp to temperature followed by a hold at temperature until failure is detected. Some tests were stopped short of failure to better understand the condition of the pins prior to failure. Posttest examination involved cladding strain measurements and posttest metallography.

The computer-controlled radiant heating system was also able to simulate various loss of decay heat removal conditions during which the fuel heats gradually over a period of 24–36 hours, and considerable eutectic formation between the fuel and cladding materials can be expected. In-pin fuel motion, cladding failure times, and postfailure fuel motion are important issues under such conditions. The configuration of the furnace test facility was able to address the first two of these issues through a series of tests that were terminated at failure and at selected times prior to failure.

5.3. *TREAT metal fuel tests*

A series of pin-disruptive tests with metal alloy fuels were conducted during the Integral Fast Reactor program in Argonne's TREAT facility using flowing sodium loops [11–15]. Beginning in 1985, six experiments were performed with irradiated EBR-II metal fuel, designed to provide data pertaining to anticipated over-power transient without-scram conditions (UTOP). The objective was to study the behavior of fuel and cladding near the cladding failure threshold, for a range of burnup and fuel-cladding combinations. Achieving this objective required, for some fuel pins, stopping the power transient at the brink of failure and, for other pins, stopping the transient immediately after failure. Specific goals for the tests also included accurately measuring the time-dependent prefailure axial growth of the fuel slugs using a hodoscope system designed to collimate and detect fast neutrons born by fissions in the test fuel.

The sodium loop used in TREAT tests was a thick-walled stainless steel pipe through which liquid sodium is circulated. The first three tests (M2, M3, and M4) each accommodated three EBR-II driver pins that contained U–5% Fs fuel in stainless steel cladding. In subsequent tests (M5, M6, and M7), U–Pu–Zr and U–Zr reference metal alloy fuels were studied, but only two pins per test could be accommodated because of their larger cladding diameter. In all six tests, each pin was located in a separate stainless steel flow tube with lateral separation of the pins as wide as possible in order to maximize the hodoscope's ability to distinguish the fuel in one pin from that in another, and to minimize the azimuthal power gradient in the test fuel due to neutron shielding of one fuel pin by another. The cross section view of the test trains is shown in Fig. 7. Coolant temperatures at the outlet and along the fuel zone were measured using thermocouples attached to the outer surface of each thin flow tube. The flow tubes remained intact in all five instances in which fuel pins failed, except in one case (M7) in which sodium was found outside the flow tubes but the flow tube breach was scarcely detectable.

All test fuel pins in the M series were subjected to similar over-power conditions: full coolant flow and an exponential power rise on an 8-second period as the slowest possible transient within the energy deposition limitations of the TREAT reactor. Baseline thermal conditions in the test fuel were referenced to nominal operating conditions in EBR-II (peak linear power rating of 40 kW/m, an inlet temperature of 630 K, and a 150 K coolant temperature rise). The power rise was rapidly terminated upon detection of cladding breach or just prior to failure. Out of 15 metal fuel pins tested in six TREAT tests, five were overheated to cladding breach. In every case, the cladding breach occurred near the top of the fuel slug. Over-power levels achieved in each case were about four times the nominal power. A summary of peak over-power conditions and fuel performance results achieved in the test series is given in Table 2.

The high thermal conductivity of metal fuel assures peak cladding temperatures, hence likely failure sites, near the top of the fuel column. Temperatures key to the failure threshold analysis (pin plenum, peak cladding midwall, and cladding inner-surface temperatures) are close to or easily correlated to the measured whole-pin coolant temperature rise. The rate of this temperature rise is sufficiently rapid that, except at the highest possible burnups, failure would not be expected until the temperature of the fuel cladding interface exceeds a threshold value of 1,350 K corresponding to the temperature at which eutectic penetration into the cladding becomes very rapid.

The calculations for cladding failure predict that nearly total eutectic penetration would be required to fail cladding at low burnup, partial penetration would be required at mid-range burnups, and almost no penetration would be required

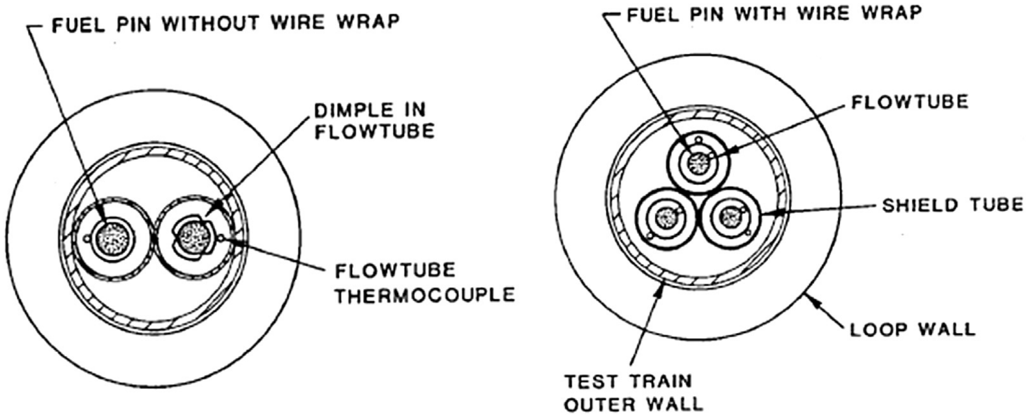

Fig. 7 – Two-pin and three-pin test train cross sections used in Transient Reactor Test Facility (TREAT) metal fuel tests.

at high burnup. At the heating rates of these TREAT tests, the margin to failure was not expected to depend strongly on the particular metal fuel or cladding type. In all but the three highest burnup pins, eutectic penetration plays a major role in computed cladding damage. With the noticeable exception of the single U–Zr pin, observed cladding failures were in reasonable agreement with pretest predictions. Melt fraction profiles from the posttest examinations for the unfailed pins reveal melting of approximately one-half of the total fuel inventory encompassing > 90% of the pin cross section near the fuel top.

The survival of the U–Zr pin tested to about 4.8 times nominal power was unexpected because computed temperatures far exceeded the expected threshold (1,350 K) for rapid eutectic melt penetration. To obtain the expected relationship for melt penetration, it is necessary that a molten phase rich in uranium be in contact with the cladding. It is possible that the high solidus temperature (~1,500 K) of binary fuel prevented or delayed the onset of this condition necessary for rapid melt penetration with HT9 cladding.

The measurements of peak prefailure elongation made by the fast neutron hodoscope for each test pin vary between 2% and 20%, significantly beyond an approximate 1% attributable to pure thermal expansion. In most cases, the peak expansion persisted during cooldown and was evident in posttest examinations. Measured expansions of irradiated EBR-II driver fuel showed strong dependence on burnup and were especially large at low burnup. By contrast, expansion of the oxide fuel is typically less and does not show large burnup dependence. In irradiated fuel, the underlying mechanism is believed to be the expansion of fission gas that is initially confined within solid fuel but freed to expand when fuel approaches its melting point and the fuel matrix softens.

When cladding failed, postfailure events were characterized by rapid fuel dispersal, a sudden but temporary reversal of inlet coolant flow, and rapid coolant voiding. Different fuel types tested behaved similarly. Pressure spikes were minor (< 2 MPa) and were correlated to the plenum pressure of the failed pin. In each case, about half of the molten fuel inventory was ejected through a small breach at or near the top of the fuel column. Quantitatively, the amount of disruption observed seemed correlated to the amount of pin depressurization following failure, driven either by expansion of

Table 2 – Peak over-power conditions and metal fuel performance in TREAT tests.

Fuel/cladding	Axial peak burnup (at.%)	Normalized peak over-power	Peak pressure (MPa)	Maximum axial expansion (%)
U–5%Fs/316 SS	Fresh	3.8	0.6	4
	0.3	4.1	0.6–0.8	16
	0.3	4.1	0.6–0.8	18
	2.4	4.1 (failed)	2–6	7
	4.4	4.2 (failed)	7–9	Unclear
	4.4	4.0	7–9	4
	4.4	3.8	7–9	4
	7.9	4.1 (failed)	17–20	3
	7.9	3.4	17–23	4
U–19%Pu–10%Zr/D9 steel	0.8	4.3	1	1
	1.9	4.3	3	3
	1.9	4.4	3	5
	5.3	4.4 (failed)	10	4
	9.8	4.0 (failed)	19	2
U–10%Zr/HT9	2.9	4.8	6	3

TREAT, Transient Reactor Test Facility.

trapped fission gas or sudden boiling of the liquid sodium bond within the fuel.

5.4. *Molten metal fuel quench tests*

The breakup of jets and drops of molten metals in various liquids other than uranium alloys in sodium are historically studied on a gram scale. A test series was therefore carried out to study the breakup and interaction behavior of kilogram quantities of U alloys in sodium. In addition, data on heat transfer, solidification, and impingement heat flux were obtained.

The molten fuel quench tests were conducted in the apparatus that consisted of a furnace/injector for melting the pour stream metals, an interaction vessel containing sodium, and an overall containment vessel [16]. The uranium metals were melted inductively with a 30-kW 10,000-Hz generator. A ZrO plug was removed pneumatically from a MgO crucible to initiate the downward pour of the fuel melt. The sodium was contained in heated interaction vessels of variable length with argon cover gas. The instrumentation for indicating conditions in the interaction vessel consisted of a bundle of thermocouples, sodium level indicators of the spark plug type, and a pressure transducer.

No vapor explosions resulted from these interactions because the conditions of the pour stream and sodium were far from satisfying vapor explosion criteria [17]. The quenched particulate material was primarily in the form of filaments and sheets. Particle size decreased with increased duration of the hydrodynamic action on the pour stream prior to freezing. The largest particles were obtained from the tests with low melt temperature and, in the test with high injection velocity of 10 m/s, the pour stream was dispersed into smaller particles (with a mean size of 0.6 mm) and a lower void fraction than the low-velocity tests in which the pour stream was accelerated by gravity to ~2 m/s. Size distributions of particles and mean particle sizes were measured after each test.

It was evident from calculations based on typical bed conditions observed in these tests that the debris from a meltdown of a metal-fuel pool reactor would be largely coolable by conduction alone without considering enhanced heat removable by convection. For 10% porosity and no stainless steel in the bed, 1% decay heat (0.145 MW/kg thermal full power) could be removed by conduction in bed depths up to ~0.12 m before boiling is initiated. Several correlations for the maximum boiling heat flux before dryout of deep beds were compared for a representative particle size of 10 mm, and all consistently showed that, for 0.9 voidage, bed depths are significantly high. Even for a bed with 50% porosity, the entire core with no stainless steel (0.3 m depth) at the bottom spherical surface of a reactor with a 6 m radius would be coolable. It was concluded from this study on metal fuel pour stream breakup and coolability that in-vessel retention would be the likely outcome.

6. Conclusion

A survey of metallic fuel safety performance characteristics is presented along with a summary of experiments from laboratory and in-pile tests. A characteristic of the metal fuel that inhibited the early development of high burnup pins is the extensive fuel swelling early in life. Later on, a more complete understanding of the nature of this swelling led to higher burnup performance when enough volume was included to allow the fuel to reach the state where significant interconnected porosity developed allowing fission gas release to the plenum. This eliminated FCMI, and the fission gas pressure in pin plenum determined the cladding loading. A unique aspect of the metal fuel pins, by contrast, is the formation of a low melting point eutectic intermetallic between the uranium and iron at the fuel-cladding interface. If transient temperatures are sufficiently high for an extended period, the potential exists for a thinning of the cladding and subsequent breach. To develop significant cladding thinning, an abundance of molten fuel is required to drive the reaction. When zirconium is used as a component in the metal fuel alloy, this eutectic penetration is delayed and reduced. The zirconium raises the eutectic temperature and provides a protective region that reduces the migration of uranium to the cladding surface.

One of the major characteristics of metal alloy fuels is the high thermal conductivity. The metal fuels have thermal conductivity almost an order of magnitude greater than the oxide fuels. When combined with the use of a sodium bond to improve gap conductivity during early life, this leads to significant differences in the radial and axial temperature profiles within the metal and oxide fuel pins. The radial temperature profile is significantly flatter in metallic fuels, and the temperature drop across the fuel-cladding gap is negligible. Although the melting point of the metal alloys is relatively low, the good heat transfer keeps the temperatures well below melting during normal operation, by about the same margin for oxide fuel.

A final characteristic important for the safety performance of metal fuels is the axial thermal profile. Because of the high thermal conductivity and gap conductance, the location of the maximum fuel temperature for metal fuels in steady operation and in most transients is well above the axial midplane. This biases the failure locations toward the top of the fuel column, where the cladding is generally the weakest and the reactivity effects due to molten fuel relocation are consistently negative. In many transient scenarios, propagation of fuel melting through the top of the fuel slug leads its extrusion into the pin plenum (above the active core) prior to cladding breach. This in-pin fuel motion introduces a substantial negative reactivity, producing a strong shutdown effect and allowing a clean recovery from severe accident initiators. If the cladding fails, the internal fission gas pressure forces the molten fuel into the coolant channel, and the upward ex-pin molten fuel motion also reduces the core reactivity, allowing to avoid recriticalities.

Acknowledgments

This work was partially supported by the Korea Atomic Energy Research Institute under interagency agreement, the U.S. Department of Energy contract DE-AC02-06CH11357.

REFERENCES

[1] The Generation IV International Forum (GIF) [Internet]. OECD Nuclear Energy Agency, Available from: https://www.gen-4.org/gif/jcms/c_9260/public.

[2] The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) [Internet]. International Atomic Energy Agency. Available from: http://www.iaea.org/INPRO/.

[3] L.C. Walters, B.R. Seidel, J.H. Kittel, Performance of metallic fuels and blankets in liquid–metal fast breeder reactors, Nucl. Technol. 65 (1984) 179–231.

[4] D. Crawford, D. Porter, S. Hayes, Fuels for sodium-cooled fast reactors: US perspective, J. Nucl. Mater. 371 (2007) 202–231.

[5] Y. Chang, Technical rationale for metal fuel in fast reactors, Nucl. Eng. Technol. 39 (2007) 161–170.

[6] H.P. Planchon, et al., Implications of the EBR-II inherent safety demonstration test, Nucl. Eng. Des. 101 (1987) 75–90.

[7] C.M. Walter, L.R. Kelman, The interaction of iron with molten uranium, J. Nucl. Mater. 20 (1966) 314.

[8] P.R. Betten, J.H. Bottcher, B.R. Seidel, Eutectic penetration times in irradiated EBR-II driver fuel elements, Trans. Am. Nucl. Soc. 45 (1983) 300.

[9] B.R. Seidel, Metallic fuel cladding eutectic formation during postirradiation heating, Trans. Am. Nucl. Soc. 34 (1980) 210.

[10] C.E. Lahm, J.F. Koenig, P.R. Betten, J.H. Bottcher, W.K. Lehto, B.R. Seidel, EBR-II driver fuel qualification for loss of flow and loss of heat sink tests without scram, Nucl. Eng. Des. 101 (1987) 25.

[11] C.E. Dickerman, et al., TREAT sodium loop experiments on performance of unbonded, unirradiated EBR-II Mark I fuel elements, Nucl. Eng. Des. 12 (1970) 381–390.

[12] W.R. Robinson, et al., Integral fast reactor safety tests M2 and M3 in TREAT, Trans. Am. Nucl. Soc. 50 (1985) 352.

[13] A.E. Wright, et al., Recent Metal Fuel Safety Tests in TREAT, Proceedings of the ANS/ENS International Conference of the Science and Technol. of Fast Reactor Safety, CONF-86050, Guernsey, England, May 1986.

[14] T.H. Bauer, et al., Behavior of uranium–fissium fuel in TREAT transient overpower tests, Trans. Am. Nucl. Soc. 53 (1986) 306.

[15] W.R. Robinson, et al., First TREAT transient overpower tests on U–Pu–Zr Fuel: M5 and M6, Trans. Am. Nucl. Soc. 55 (1987) 418.

[16] T.H. Bauer, G.R. Fenske, J.M. Kramer, Cladding Failure Margins for Metallic Fuel in the Integral Fast Reactor, Transactions of SMiRT-9, Lausanne, Switzerland, August 1987.

[17] B.W. Spencer, J.F. Marchaterre, Scoping Studies of Vapor Behavior during a Severe Accident in a Metal-fueled Reactor, Proc. Intl. Topical Mtg. on Fast Reactor Safety, Knoxville, TN, CONF-850410 vol. 1, April 1985, p. 151.

[18] T. Ogata, Y.S. Kim, A.M. Yacout, Metal fuel modeling and simulation, in: Comprehensive Nuclear Materials, Elsevier, 2011 (Chapter 75).