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Fuels for sodium-cooled fast reactors: US perspective

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Abstract

The US experience with mixed oxide, metal, and mixed carbide fuels is substantial, comprised of irradiation of over 50 000 MOX rods, over 130 000 metal rods, and 600 mixed carbide rods, in EBR-II and FFTF alone. All three types have been demonstrated capable of fuel utilization at or above 200 GWd/MTHM. To varying degrees, life-limiting phenomena for each type have been identified and investigated, and there are no disqualifying safety-related fuel behaviors. All three fuel types appear capable of meeting requirements of sodium-cooled fast reactor fuels, with reliability of mixed oxide and metal fuel well established. Improvements in irradiation performance of cladding and duct alloys have been a key development in moving these fuel designs toward higher-burnup potential. Selection of one fuel system over another will depend on circumstances particular to the application and on issues other than fuel performance, such as fabrication cost or overall system safety performance.

1. Introduction

The evolution of US Department of Energy Programs for the Generation IV Initiative [1,2], the Advanced Fuel Cycle Initiative [3], and the recently proposed Global Nuclear Energy Partnership [4] have renewed and invigorated US interest in sodium-cooled fast reactors (SFRs) for long-term actinide management and energy production. Previous programmatic efforts in the US included the Liquid Metal Fast Breeder Reactor (LMFBR) program of the 1960s through early 1980s (e.g., [5]) and the Integral Fast Reactor (IFR) [6,7] and Advanced Liquid Metal Reactor (ALMR) programs (e.g., [8]) of the 1980s and early 1990s. These efforts sought to develop SFR technology for meeting energy-related objectives of interest to the times [5], which influenced the conditions under which technology choices were made. Fuel technology is a key aspect of an SFR system, with implications for reactor safety, reactor operations, fuel reprocessing technology, and overall system economics.

This paper reviews and discusses the status of SFR fuel technology, emphasizing answers to questions that arise as US technologists and Government decision makers reconsider application of SFRs to contemporary needs. The focus herein is on US experience although, after the termination of the US IFR and ALMR programs, nations such as France and Japan have continued to develop the technology over the last 15 years. This focus is temporal and simply a reflection of our need in the US to relearn domestic experience as context for understanding international progress. Because metal fuel

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and oxide fuel (specifically, $(U,Pu)O_2$, termed mixed oxide fuel, or MOX) were developed to the point of reliable operation, performance of these fuel systems is the primary thrust of this paper, although mixed carbide and mixed nitride fuels are also considered.This paper addresses relevant performance and behavior phenomena for each fuel type, emphasizing those that impact fuel reliability and safety-related behavior. A similar review emphasizing fuel fabrication experience is being prepared for separate publication [9]. A historical review of US fast reactor fuels and materials development is also in preparation for separate publication [10]. For the fuel types used as driver fuel for test reactors in the US, metal fuel in Experimental Breeder Reactors I and II (EBR-I, EBR-II) and MOX fuel in the Fast Flux Test Facility (FFTF), a considerable amount of work was put forth to understand driver fuel performance and reliability [11–15] and to extend the utilization of those fuels to higher burnup [16–19]. This work to improve driver fuel for EBR-II and FFTF, and the safety testing programs that supported it, is addressed here as well, because that work underlies the technical maturity of these two fuel types. Additional work was performed in the US with carbide and nitride fuels as advanced options [20,21], and the carbide fuel system in particular showed some promise; so these fuel types are addressed in less detail, with emphasis on attributes that potentially render such fuels attractive alternatives to metal and MOX fuels and on what is known about their performance phenomena. Because the varied terminology used in the literature can be confusing when referenced in a paper like this, the authors clarify the terminology used in this paper, as listed in Table 1.

2. Metal fuel performance phenomena

2.1. Evolution of metal fuel design for higher-burnup operation

Metal fuel was selected for fueling many of the first reactors in the US, including the plutonium

Table 1
Suggested terminology

Suggested term	Description
Fuel *Slug*	An unencapsulated, as-fabricated piece of metallic fuel. Has also been referred to as a *pin*. The term was originally used for somewhat massive pieces used, for example, in plutonium production reactors. Even though it is less descriptive of a fast reactor fuel casting with a low diameter-to-length ratio, use of this term avoids confusion with other previous uses of the term *pin*
Fuel *Pellet*	The smallest, as-fabricated unit of a pellet-type ceramic fuel
Fuel *Column*	The column of fuel in a fuel rod, comprised of a fuel slug, a stack of segmented slugs, or a stack of fuel pellets. For a fast reactor driver fuel, the fuel column height defines the active core height
Fuel *Rod*	Fuel capsule, variously referred to as fuel *element*, *pin*, or *rod*
Fuel *Bundle*	Configured grouping of fuel rods, usually comprising all or part of the array of fuel rods in an assembly. This term is typically applied to the grouping of fuel rods inside an assembly, exclusive of the associated assembly hardware or enclosing duct
Fuel *Assembly*	Assembled array of fuel rods, comprising a convenient fuel handling unit. Typically enclosed in ducts, for fast reactor application, to ensure minimum coolant flow through a fuel bundle. Usually referred to as a *subassembly* in the EBR-II lexicon
Fuel *Smeared Density*	The areal density of as-fabricated fuel inside of the as-fabricated inner-wall surface of the cladding. Fuel porosity is reflected in this value. Usually expressed as a percentage, as determined from the following: 100 × (% of theoretical density) × (square of fuel outer diameter)/(square of cladding inner diameter). Has also been referred to as *smear density, and smeared fuel density*
Fuel *Burnup*	The amount of heavy metal (i.e., uranium and higher actinides) in the fuel that has been fissioned. Can be expressed as percent of heavy metal atoms that have fissioned (at.%) or in units of fission energy produced per unit mass of heavy metal (GWd/MTHM or MWd/kgHM). 1 at.% of burnup corresponds to roughly 9.4 GWd/MTHM
Exposure	When applied to in-core components and materials of construction this term refers to a duration and intensity of in-service use. Can be expressed in units of neutron fluence (n/cm^2), usually for neutrons of energy greater than 0.1 MeV ($E > 0.1$ MeV); units of proportional incident neutron or fission fragment collision (displacements per atom, or dpa), which incorporates neutron spectrum to reflect degree of radiation damage; or units of simple time of exposure, such as effective full-power days (EFPD)
Fuel or Cladding *Breach*	Breach of the cladding, which is the boundary between the coolant and the fuel or fission products, considered the first safety barrier against release of fission products. The term *failure* is commonly used, but the present authors suggest that a cladding breach is not necessarily a failure of the fuel design to perform its function

production reactors operated at Hanford, the Experimental Breeder Reactor (EBR, later renamed the Experimental Breeder Reactor I, or EBR-I) in Idaho, and the Dounreay Fast Reactor (DFR) in the UK [19,21,22]. The reasons for its selection included ease of fabrication, high-thermal conductivity and high-fissile and -fertile densities (which allowed higher breeding ratios and smaller core sizes for specific reactor power). Alloying elements were added to the uranium and uranium–plutonium metal primarily to enhance dimensional stability under irradiation, but also to improve corrosion resistance (e.g., in water-cooled plutonium reactors) and to tailor alloy solidus and liquidus temperatures toward desired values for fabrication and operational performance. The unusual irradiation growth and swelling exhibited by some uranium- and plutonium-bearing alloys was attributed to textured microstructural phases with anisotropic crystal structure resulting from fabrication, leading researchers to heat treat the as-fabricated fuel to remove crystallographic texture and/or alloy the metals with intent to stabilize predominantly isotropic crystal structures at irradiation temperatures. Fuels used with EBR-I included unalloyed uranium and the alloys U–Zr and Pu–Al. The Enrico Fermi Reactor (Fermi I – an SFR located near Detroit, Michigan) was fueled with U–Mo, while U–Cr and U–Mo were used for the DFR. Experimental Breeder Reactor II (EBR-II) was started up and operated with a fuel alloy of U-5Fs (Mark-I, -IA, and -II designs), where Fs is designated as the symbol for fissium, a simulated mixture of noble-metal fission products produced in equilibrium with the original EBR-II melt-refining pyrometallurgical recycle process.[1] In the 1980s, the EBR-II was converted to a driver core of U–Zr (Mark-III/IIIA design) [11,13,19].

In the late 1960s, emerging economic requirements for fast reactor fuel cycles led researchers toward fuels capable of higher burnup than was possible with the first EBR-II fuel design [19,20]. A uranium–plutonium oxide variant of the uranium oxide fuel being deployed in light water reactors was attractive because of the high-temperature capability of the oxide fuel and the technology synergy with the developing industrial base for fabrication and reprocessing of oxide light water reactor fuel [19,20]. EBR-II's mission was then changed to support irradiation testing of mixed oxide fuel for the Liquid Metal Fast Breeder Reactor Program (LMFBR) [23–25].

Over the course of EBR-II operation as an irradiation facility, the drive to reduce operating costs of the reactor motivated further development of metal fuel design to address the behavior characteristics that limited EBR-II Mark-I and Mark-IA burnup to 1.2 at.%[2] and 2.6 at.%, respectively [19,21,23]. If fabricated in a manner to avoid formation of crystallographic texture, U–5Fs had sufficient phase stability, irradiation growth resistance, and operating temperature range to support EBR-II operation. However, fuel rod design was insufficient to avoid cladding breach due to stresses induced by fuel swelling, by fission gas release, and by stress placed on fuel liftoff restraints [19]. These were addressed with design changes embodied in the Mark-II fuel design, which included addition of impurity-level amounts of Si to the fuel alloy to reduce the rate at which the fuel swelled toward the cladding, a larger fuel-to-cladding gap to accommodate swelling to the point where fuel porosity interconnected to allow fission gas release into the fuel plenum (thus reducing greatly the driving force for continued fuel swelling), a larger fuel rod plenum to accommodate released fission gas with less stress on the cladding, and a thicker cladding wall [16,21,24]. The cladding alloy was later changed from Type 304L stainless to Type 316 stainless steel, due to the improved swelling resistance of Type 316 stainless steel [16], and because fuel/cladding interdiffusion (which effectively reduces the load-bearing thickness of the cladding wall) was significantly reduced with Type 316 stainless steel [11,19]. The Mark-II fuel design proved to be capable of reliable operation to 10 at.% burnup [11,22].

2.2. Development of U–Zr and U–Pu–Zr fuel

Prior to the change in US fast reactor development toward mixed oxide fuel, U–Pu–Zr alloys were being investigated for conversion of EBR-II to a plutonium core [20,23]. Of the several alloying elements being considered for addition to the

[1] Fs: 2.4 wt% Mo, 1.9 wt% Ru, 0.3 wt% Rh, 0.2 wt% Pd, 0.1 wt% Zr, 0.01 wt% Nb.

[2] Burnup is reported here using either at.% or GWd/MTHM, consistent with the units used in the literature source. 1 at.% of burnup corresponds to roughly 9.4 GWd/MTHM. See Table 1 for explanation of terminology.

uranium–plutonium binary alloy to raise the solidus temperature, Zr seemed promising due to its reduction of fuel/cladding interdiffusion and favorable early irradiation testing results [20,22].

With the termination of the Clinch River Breeder Reactor (CRBR) project and the LMFBR program in the early 1980s, the Integral Fast Reactor program was proposed to continue development of long-term, sustainable fuel cycles with technology developed to avoid some of the challenges that contributed to CRBR termination [7,24]. The reactor and fuel cycle concept embodied in the IFR was intended to provide improved, inherent reactor safety characteristics through use of metal fuel, which provided improved feedback characteristics, good fuel/coolant compatibility in the case of cladding breach, and beneficial transient overpower behavior [19,24,26]. Such advantages in safety-related behavior offered the prospect of avoiding deterministically the accident scenarios that were proving to add complexity and cost to the CRBR design. U–Pu–Zr was also compatible with the pyrometallurgical recycle technology proposed for the IFR concept, and with the evolving metal fuel design that allowed much higher burnup than was originally achieved [19–21,23,24].

An experimental program was undertaken to adapt principles learned in evolution of EBR-II fuel design to U–Pu–Zr, for varying amounts of Pu [20,23,27–29]. These experiments, many of which are summarized in Table 2, demonstrated burnup potential of U–19Pu–10Zr and U–10Zr clad in advanced alloys such as the 20% cold-worked austentic D9 or ferritic–martensitic HT9 (advanced alloys developed in the LMFBR program [23]) near 20 at.%, the burnup level that had become a goal of the US fast reactor program as part of an effort to make fast reactor fuel cycles competitive with light water reactor fuel cycles [20,30,31]. Although the program included fuel testing in the D9 alloy, deployment of HT9 as the cladding and duct material was the long-term intention.

Extension of metal fuel design to the U–Zr/U–Pu–Zr fuel system was straightforward. Phenomena known to control metal fuel lifetime, such as fuel swelling, fission gas release, and fuel/cladding interdiffusion proved to be no more severe or better than observed in the U–5Fs fuel that had formed the bulk of the metal fuel database. In particular, the application of cladding alloys with improved swelling and irradiation creep behavior compared to Type 316 stainless steel, which were also being applied to mixed oxide fuel [32], proved to be key to enabling higher-burnup performance.

The use of U–Pu–Zr alloy and the targeted higher-burnup capability required further evaluation of key fuel performance phenomena, some of which are unique to the Zr alloys. Although these phenomena did not prevent attaining 20 at.% burnup, they were of interest for determining fuel operational limits based on performance under design basis accidents (DBAs). In addition to fuel swelling and fission gas release, such phenomena included fuel constituent redistribution (interdiffusion of U, Pu, and Zr within the fuel to create Zr-depleted radial zones with attendant lower, local solidus temperatures), and fuel/cladding interdiffusion enhanced by lanthanide fission products present in increasing amounts with higher burnup, which could lead to formation of lower-melting-temperature composition regions in the fuel, effectively thinning of the cladding. These phenomena were studied in some detail [33–39].

Observations of Zr-depleted zone formation indicated the phenomenon is more pronounced in U–Pu–Zr than in U–Zr and is temperature dependent. The significant result was the understanding that formation of such zones is due to phase equilibria effects established in the temperature gradient across the radius of the fuel – meaning that the low-solidus-temperature Zr-depleted would form in a region of the fuel where, under normal conditions, the temperature would not exceed the local solidus temperature [34,36–38]. Studies of U–Pu–Zr and cladding interdiffusion found that interdiffusion zones would melt under simulated transient conditions at temperatures as low as 675 °C, leading to cladding penetration and breach [33,39]. The beneficial effects of Zr at the fuel/cladding interface were exhibited in various ways (e.g., [40]). The practical effects of these phenomena were manifest in limits derived through conservative application of analytical methods to determine Limiting Conditions of Operation (LCOs) of EBR-II fueled with a U–Pu–Zr driver core [41].

3. Metal fuel irradiation performance database

3.1. Steady-state performance: reliability and burnup potential

The metal fuel development path was necessarily entwined with operation of metal driver fuel for EBR-II. This development path included the

Table 2
Summary of selected metal fuel irradiation experiments in EBR-II and FFTF

Experiment number	Fuel composition	Cladding material	No. of rods in assembly	Smeared density (%)	Cladding OD (cm)	Wall thickness (cm)	Plenum/ fuel vol. ratio	Peak power (kW/m) (beginning of life)	Peak cladding temp (°C) (beginning of life)	Peak burnup (at.%)	Fast fluence $\times 10 * 22$ n/cm^2 ($E > 0.1$ MeV)	Key breached rod information
X419 Prototype and fuel behavior	U–10Zr, U–8Pu–10Zr, U–19Pu–10Zr	D9	61	75	0.584	0.038	1	39.4	560	11.9	12	
X420 Prototype, fuel behavior, failure mode, RBCB	U–10Zr, U–8Pu–10Zr, U–19Pu–10Zr	D9	61	75	0.584	0.038	1	36.1	590	18.4	18.5	1 breach @ 16.4 at.% burnup; 530 °C at breach
X421 Prototype, fuel behavior, failure mode	U–10Zr, U–8Pu–10Zr, U–19Pu–10Zr	D9	61	75	0.584	0.038	1	39.4	560	17.1	19.6	
X423 Fuel swelling and restructuring	U–10Zr, U–3Pu–10Zr, U–8Pu–10Zr, U–19Pu–10Zr, U–22Pu–10Zr, U–26Pu–10Zr	316	37	75	0.737		1	42.7	522	4.9	8.07	
X425 (X425A/B/C) Lead IFR	U–10Zr, U–8Pu–10Zr, U–19Pu–10Zr	HT9	61	75	0.584	0.038	1	48.2	590	3,11,16.2, 19.3	20.6	
X429 (X429A/B) Fabrication variables and strain prediction	U–10Zr, U–8Pu–10Zr, U–19Pu–10Zr	HT9, 316SS	61	75	0.584	0.038	1	42.7	600	7.7, 10.6, 14.4	13.8	1 breach @ 6.5 and 1 breach @ 10 at.% burnup
X430 (X430A/B) HT9, peak cladding temp, large diameter, compatibility	U–10Zr, U–19Pu–10Zr, U–22Pu–10Zr, U–26Pu–10Zr	HT9	37	75	0.737	0.041	1.4	49.2	540	11.5	20.6	
X431 (X431A) Blanket safety	U–2Zr, U–6Zr, U–10Zr	HT9	19	85	0.940	0.038–0.051	1.8	39.4	507	3.9	15.4	
X432 (X432A) Blanket safety	U–2Zr, U–6Zr, U–10Zr	HT9	19	85	0.940	0.038–0.051	1.8	39.4	507	4.5	16.6	
X435 (X435A) Mk-III qual.	U–10Zr	D9	61	75	0.584	0.038	1.4	49.2	591	19.8	22.8	
X436 Mk-III qual.	U–10Zr	D9	61	75	0.584	0.038	1.4	34.4	596		8.45	
X437 Mk-III qual.	U–10Zr	D9	61	75	0.584	0.038	1.4	37.7	597		10	
X438 Mk-III qual.	U–10Zr	D9	61	75	0.584	0.038	1.4	32.8	623		9.45	

X441 (X441A) FCMI test and LIFE-METAL benchmark	U–19Pu–6Zr, U–19Pu–10Zr, U–19Pu–12Zr	HT9&D9	61	70–85	0.584	0.038	1.1–2.1	45.9	600	12.7	10.1	
X447 (X447A) U–Zr high temp.	U–10Zr	HT9	49	75	0.584	0.046	1.4	36.1	660	10	9.17	2 breaches @ 9.5 at.% burnup; 630 °C at breach
X448 (X448A) Mk-IV qual.	U–10Zr	HT9	61	75	0.584	0.046	1.4	45.9	552	14.6	14.9	
X449 Mk-IV qual.	U–10Zr	HT9	61	75	0.584	0.046	1.4	29.5	578	11.3	17.7	
X450 Mk-IV qual.		HT9	61	75	0.584	0.046	1.4	36.1	576	10.2	13.1	
X451 (X451A) Mk-IV qual.	U–10Zr	HT9	61	75	0.584	0.046	1.4	32.8	623	13.7	13.7	
X452 Fuel impurities	U–10Zr	D9	61	75	0.584	0.038		34.4	596	6.1	5.38	
X453 Fuel impurities	U–10Zr	D9	61	75	0.584	0.038		34.4	596	8.5	8.45	
X454 Fuel impurities	U–10Zr	D9	61	75	0.584	0.038		49.2	547	8.3	9.12	
X455 Fuel impurities	U–10Zr	D9	61	75	0.584	0.038		49.2	547	10.3	9.16	
X481 Mk-III design with Pu	U–19Pu–10Zr	D9	61	75	0.584	0.038	1.4	49.2	579	10	11.3	
X483 (X483A) Mk-IIIA, reference 316SS qual.	U–10Zr	316	61	75	0.584	0.038	1.4	49.9	552	14.8	15.7	
X484 Mk-IIIA, reference 316SS qual.	U–10Zr	316	61	75	0.584	0.038	1.4	36.1	576	11.7	11.9	
X485 Mk-IIIA, reference 316SS qual.	U–10Zr	316	61	75	0.584	0.038	1.4	39.7	576	10.5	10.7	
X486 Mk-IIIA, reference 316SS qual.	U–10Zr	316	61	75	0.584	0.038	1.4	37.1	623	13.9	13.9	
X489 High-Pu for PRISM design	U–19Pu–10Zr, U–28Pu–10Zr	HT9, HT9M	61	75	0.584	0.046	1.4	36.1	606	5.4	4.83	
X492 (X492A/B) Zr-sheathed fuel	U–3Zr, U–20.5Pu–3Zr	HT9, HT9M	61	75	0.584	0.038	1.4	41.0	551	10.5	11.1	

(*continued on next page*)

Table 2 (*continued*)

Experiment number	Fuel composition	Cladding material	No. of rods in assembly	Smeared density (%)	Cladding OD (cm)	Wall thickness (cm)	Plenum/ fuel vol. ratio	Peak power (kW/m) (beginning of life)	Peak cladding temp (°C) (beginning of life)	Peak burnup (at.%)	Fast fluence $\times 10 * 22$ n/ cm^2 ($E > 0.1$ MeV)	Key breached rod information
X496 Long lifetime	U–10Zr	HT9	37	59	0.686	0.056	3	63.3	536	8.3	6.9	
X501 Minor-actinide-bearing fuel	U–20.2Pu–10Zr–1.3Np–1.2Am, U–10Zr	HT9	2 + 59	75	0.584	0.046	1.4	44.9	⩽540	7.6	6.4	
IFR-1 Fuel column length effects	U–10Zr, U–8Pu–10Zr, U–19Pu–10Zr	D9	169	75	0.686	0.056	1.2	49.2	615(604)	94 GWd/MTHM	15.4	
MFF1A FFTF lead Metal fuel test	U–10Zr	HT9	8	75	0.686	0.056	1.2	42.7	577	38 GWd/MTHM	5.6	
MFF-1 FFTF lead metal fuel test	U–10Zr	HT9	5	75	0.686	0.056	1.2	43.0	577	95 GWd/MTHM	17.3	
MFF-2 FFTF Metal prototype	U–10Zr	HT9	169	75	0.686	0.056	1.3	54.1	618	143 GWd/ MTHM	19.9	
MFF-3 FFTF Metal prototype	U–10Zr	HT9	169	75	0.686	0.056	1.3	59.1	643	138 GWd/ MTHM	19.2	
MFF-4 FFTF Series III.b qualification	U–10Zr	HT9	169	75	0.686	0.056	1.5	56.8	618	135 GWd/ MTHM	19	
MFF-5 FFTF Series III.b qualification	U–10Zr	HT9	169	75	0.686	0.056	1.5	55.8	651	101 GWd/ MTHM	14	
MFF-6 FFTF Series III.b qualification	U–10Zr	HT9	169	75	0.686	0.056	1.5	55.8	588	141 GWd/ MTHM	12.8	

fabrication and irradiation in EBR-II of about 90 000 Mark-I/IA driver fuel rods [22] (35 000 from 1964 to 1969 alone, fabricated remotely in the EBR-II Fuel Cycle Facility [21]), over 30 000 Mark-II driver fuel rods [11,24], 13 000 Mark-III/IIIA/IV (U–10Zr alloy) driver fuel rods, and over 600 U–Pu–Zr fuel rods [13,24]. Mark-II driver fuel was qualified for 8 at.% burnup, while Mark-IIIA driver fuel was qualified for 10 at.% burnup. U–Zr rods (in experiments and qualification assemblies) and U–Pu–Zr rods (in experiments) clad in Type 316, D9 or HT9 cladding reached terminal burnup values of 15–20 at.% burnup, with some 2-sigma high-temperature assemblies reaching 11–12 at.% burnup [29,42]. In addition to EBR-II irradiations, over 1050 U–10Zr fuel rods and 37 U–Pu–Zr fuel rods were irradiated in the FFTF to burnup values above 14 at.% and 9 at.%, respectively [29]. The significance of these irradiation tests were to (1) effectively qualify U–Zr as the Series III.b driver fuel for FFTF, and (2) demonstrate that there were no length or aspect ratio effects in metal fuel that were obscured by the relatively short core height of EBR-II [42]. (Specifically, the fuel did not slump under the weight of the longer fuel column, and irradiation in EBR-II was shown to be a conservative indicator of fuel performance because the shorter fuel column combined with the same assembly outlet temperature as found with larger core heights resulted in a more challenging combination of local fuel burnup and fuel and cladding temperatures.)

Although detailed discussion of cladding breaches in fast reactor fuel is beyond the scope of this paper, some discussion is provided. Of the greater-than 13 600 U–Zr and U–Pu–Zr rods that have were irradiated in EBR-II (see Table 2 for a summary of selected irradiation experiments), roughly 22 breached under irradiation. Of those 22 breached rods, 16 breached in defective welds (a welding problem early in the program that was eradicated), three breached in the plenum region for due to unknown causes, and three breached in the fuel column region due to creep failure of the cladding [10]. The first of the fuel-column breaches occurred in a D9-clad, U–Pu–Zr rod in assembly X420B, the second reconstitution of IFR lead assembly X420, at 16.4 at.% burnup. The breach occurred about 2/3 up the height of the fuel column at an area where rod–rod interaction was expected to be the highest, suggesting that cladding interaction with the fuel and/or with an adjacent rod played a role in the cladding breach. Two breaches occurred in the X429A and X429Bassemblies, reconstitutions of the fabrication variables experiment, at 6.5 and 10 at.% burnup, respectively. The X420 and X429 assemblies contained fuel rods with a plenum-to-fuel-volume ratio of 1, while the X429 assembly also contained rods with varying as-fabricated characteristics for the purpose of investigating sensitivity to deviation from fuel specifications. Metal fuel designs were subsequently modified to incorporate larger fission gas plena, with a plenum-to-fuel-volume ratio of 1.4 becoming the reference for EBR-II experiments and driver fuel. Four breaches occurred in fuel rods fabricated with the 1.4 plenum-to-fuel-volume ratio. Two occurred in an HT9-clad, U–Zr rods at around 9.5 at.% burnup in assembly X447A, which was orificed to operate at a higher-than nominal cladding temperature [27]. The experience and data collected from these experiments was used to improve the fuel design for higher-burnup operation and to support efforts to model fuel behavior.

Behavior of fuel rods after a breach during operation can be important; release of considerable amounts of fuel into the coolant could increase the radionuclide content of the primary system to unacceptable levels or could even perturb flow in coolant channels, thereby affecting other fuel rods in what is termed propagation of failure. In addition to the 'natural' breaches described above, seven fuel rods were intentionally breached during irradiation [42–44] in Run-Beyond-Cladding-Breach (RBCB) tests (Table 3). The experiments were conducted in EBR-II using pre-irradiated fuel rods with defects machined into the cladding, with intention of inducing breach early in the reactor cycle. The reactor then operated with the breached fuel rod for the purpose of assessing post-breach behavior. Four of those breached fuel rods contained U–Fs, U–Zr, or U–Pu–Zr fuel clad in Type 316 stainless steel in experimental assemblies XY-21A, XY-24 and XY-27. Three occurred in U–Zr and U–Pu–Zr rods clad in D9 and HT9 stainless steels within assemblies X482, X482A, and X482B, which were tests intended to operate with breached fuel. Another naturally breached fuel rod in assembly X420B was also maintained under irradiation for some time after breach for the same purpose.

Evaluation of these tests confirmed the benign post-breach behavior of metal fuel [19,24,44], because the fuel is compatible with the sodium coolant and does not form fuel-coolant reaction products that can further stress the breach crack

Table 3
Summary of run-beyond-cladding-breach (RBCB) Experiments in EBR-II

Test designation	Test type	No. of rods	Fuel composition	Cladding type	Cladding OD (cm)	Pitch-to-diameter ratio	Linear power[e] (kW/m)	Cladding temp (°C)	Burnup[f] (at.%)	No. of rods breached	Days Irr'd after breach
Metal Fuel RBCB Tests (taken from Seidel, et al. [43], Hofman, et al. [23], and Batte and Hofman [44])											
XY-21	BFTF[a]	1, 60[b]	U–5Fs	316SS	0.44	1.38	24	573	7.9	0	N/A
XY-21A	BFTF	1, 60[b]	U–5Fs	316SS	0.44	1.38	25	593	9.3	1	54
XY-24	FPTF[c]	2, 59[b]	U–19Pu–10Zr	316SS	0.44	1.38	21	541	7.6	1	233
XY-27	BFTF	2, 59[b]	U–8Pu–10Zr	316SS	0.44	1.38	23	520	~6.0	2	131
X482	Open core	1, 60[b]	U–19Pu–10Zr	D9	0.58	1.24	39	600	14.4	1	168
X482A	Open core	1, 60[b]	U–10Zr	D9	0.58	1.24	36	600	13.5	1	100
X482B	Open core	1, 60[b]	U–19Pu–10Zr	HT9	0.58	1.24	36	600	~14	1	150
X420B	Natural breach	61	U–19Pu–10Zr	D9	0.58	1.24	–	–	~17	1	34
MOX fuel RBCB tests (taken from Lambert, et al. [66])											
RBCB-1	Open core	37	$(U–0.25Pu)O_2$	316SS	0.584	1.26	34	580	10	1	5
RBCB-2	Open core	37	$(U–0.25Pu)O_2$	316SS	0.584	1.26	28	645	14	2	14
RBCB-3	Open core	37	$(U–0.25Pu)O_2$	316SS	0.584	1.26	27	660	13	1	22
XY-2	BFTF[a]	1	$(U–0.25Pu)O_2$	321SS	0.635	–	42	730	8	1	6
K1	BFTF	19	$(U–0.25Pu)O_2$	D9	0.584	1.26	32	680	20	3	97
K2A	BFTF	19	$(U–0.25Pu)O_2$	D9	0.584	1.13	36	655	6	5	5
TOPI-2[d]	BFTF	19	$(U–0.25Pu)O_2$	D9	0.584	1.13	37	650	6	1[d]	3
K2B	BFTF	19	$(U–0.25Pu)O_2$	D9	0.584	1.13	33	645	8	2	110
K2C	BFTF	19	$(U–0.25Pu)O_2$	D9	0.584	1.13	35	800	5	4	22
V6	BFTF	37	$(U–0.25Pu)O_2$	D9	0.584	1.26	35	675	10	3	142
D1	BFTF	1	$(U–0.25Pu)O_2$	D9	0.584	–	31	550	6	1	45
D2[d]	BFTF	1	$(U–0.25Pu)O_2$	D9	0.584	–	31	550	9	1[d]	10
V2	Open core	4	$(U–0.25Pu)O_2$	D9	0.584	–	30	580	3–9	3	152
V7A	Open core	1	$(U–0.25Pu)O_2$	D9	0.584	–	26	540	7	1	5
V7B	Open core	1	$(U–0.25Pu)O_2$	D9	0.584	–	26	540	7	1	5
V7C	Open core	1	$(U–0.25Pu)O_2$	D9	0.584	–	–	370	7	–	–
V4	Open core	19	$(U–0.15Pu)O_2$	316SS	0.737	1.19	33	670	12	3	3
V5	Open core	7	$(U–0.04Pu)O_2$	316SS	1.23	1.07	32	640	2	3	150

[a] BFTF: Breached Fuel Test Facility in EBR-II which provided separate delayed neutron signal monitoring for the experiment and an above-core sampler for collection of released fuel and contamination.

[b] First number indicates the number of pre-defected (thinned) rods, and the second number indicates the remaining number of rods in the assembly. Note that the XY-series tests used instrumented assemblies that contained 61 Mark-II-size EBR-II rods, which would typically fill a 91-pin EBR-II Mark-II driver assembly.

[c] FPTF: Fission Product Test Facility in EBR-II with provision for monitoring fission products released from a breached fuel rod.

[d] With 15% overpower transient.

[e] Linear power values for metal fuel tests are pre-test predictions.

[f] Burnup values for metal fuel tests are burnup at end of test. It is unclear whether values for oxide fuel burnup are for pre-test or post-test; but the distinction does not impact the results summary, given the durations of the tests.

or wash out of the cladding into the coolant. Post-test examination indicated that mass loss from the rods was due to expulsion of bond sodium, fission gas, and cesium. The release of fission gas reduced the source of stress for post-breach widening of the crack, although it was found that with lower-swelling HT9 cladding the closer cladding proximity to the fuel could allow cracks to be subsequently widened by FCMI. The amount of fuel washed out of the breached fuel rods was small, and any observed widening of the breach due to fuel swelling was inconsequential, as there was no fuel extruded through the crack. These results indicate that an SFR can operate benignly with breached metal fuel, allowing shutdown for removal of the breached fuel to be deferred to convenient time or perhaps even to the planned end of an operating cycle.

3.2. Transient and off-normal behavior

With the start-up of the FFTF, which was to be used for the steady-state irradiation testing previously addressed by EBR-II, the mission for EBR-II evolved to support the Operational Reliability Test (ORT) program [20,21]. This included operational transient testing of oxide fuel to determine the effects of slow transients on fuel performance and lifetime [25,45]. Conduct of this test program required assessing the effect that the transients would have on EBR-II driver fuel (Mark-II design) lifetime, and to qualify the driver fuel for transient operation. A program of out-of-reactor and in-reactor testing was completed, including a regimen of 56 low-ramp-rate transients (1.6% power increase per second) and 13 high-ramp-rate tests (4 MW_t/s) [46]. The low-ramp-rate tests were evaluated with simple surveillance of the core, finding no indication of cladding breaches of the driver fuel and no performance degradation. The high-ramp-rate tests were evaluated with detailed examination of five test assemblies containing driver fuel which had experienced the low-ramp-rate tests prior the high-ramp-rate tests and with burnup ranging from 0 to 10 at.%. Some of these assemblies were designed to operate at higher temperatures to address effects at the boundaries of expected operating temperature (with or without uncertainties incorporated) uncertainties. The test program demonstrated that the metal driver fuel, including assemblies operating at the upper boundaries of expected temperature, endured the transient regimen with no discernable damage. Although these tests were completed with U–Fs Mark-II fuel, the results were extended to U–Zr (Mark-IIIA) driver fuel, which experienced EBR-II transient-overpower tests during and after the core conversion from Mark-II to Mark-IIIA driver fuel.

As the IFR safety program associated with development of the IFR concept progressed, it was determined to conduct a set of passive safety tests in EBR-II, designated as the Shutdown Heat Removal Test (SHRT) program [47]. The program successfully demonstrated the ability of a metal-fueled fast reactor to withstand loss-of-flow-without-scram (LOFWS) events and loss-of-heat-sink-without-scram events (LOHSWS) with no core damage. As with the ORT program, conduct of this aggressive test program required assessment of the damage the transients would cause to the EBR-II driver fuel and to qualify the fuel for transient operation. These issues were addressed with a high-temperature irradiation test assembly irradiated in EBR-II for 42 min with cladding temperatures reaching as high as 800 °C [46,48,49]. Post-test examination of the fuel rods revealed molten-phase attack of the cladding. Attack was more extensive in lower-burnup fuel than in higher-burnup fuel, but the attack was not sufficient to induce breach. Subsequent irradiation testing of the fuel rods that endured the transient demonstrated fuel breach occurred near 10 at.% burnup, beyond the 8 at.% design burnup limit of the fuel, and the breach was a stress rupture-induced breach typical of end-of-life breaches in metal fuel. Examination of other fuel rods showed evidence of molten-phase attack, but without significant cladding degradation. Although the examined test rods were U–Fs Mark-II fuel, the results were extended to U–Zr Mark-IIIA fuel rods. Assessment of damage to the driver fuel was performed through analysis of cumulative cladding damage, which indicated the incremental damage to fuel with each transient to be quite low, and the probability of the driver fuel reaching its burnup limit after the tests to be quite high [50,51].

Assessment of safety of an operating fast reactor requires understanding how fuel rods and bundles behave under off-normal conditions. The six M-series tests performed in the Transient Reactor Test Facility (TREAT) evaluated transient-overpower margin to failure, pre-failure axial fuel expansion, and post-failure fuel and coolant behavior for 15 rods with various combinations of U–5Fs, U–Zr, and U–Pu–Zr fuel clad in Type 316, D9, and HT9 stainless steel [23,52,53], summarized in Table 4.

Table 4
Summary of selected TREAT experiments

Test designation	Fuel/cladding	Burnup (at.%)	Transient rate	Overpower attained in test (P/P_0)[a]	Calculated breach threshold	Comments
Metal fuel TREAT tests (taken from Bauer et al. [52])						
M2	U–5Fs/316SS; Mark-II design	0.3	8-s period	4.1	4.7	16% max. axial expansion; fuel damaged but intact
	U–5Fs/316SS; Mark-II design	4.4	8-s period	4.2	4.5	Cladding breached
	U–5Fs/316SS; Mark-II design	7.9	8-s period	4.1	3.6–4.0	3% max. axial expansion; cladding breached
M3	U–5Fs/316SS; Mark-II design	0.3	8-s period	4.1	4.8	18% max. axial expansion; fuel damaged but intact
	U–5Fs/316SS; Mark-II design	4.4	8-s period	4.0	4.4	4% max. axial expansion; fuel damaged but intact
	U–5Fs/316SS; Mark-II design	7.9	8-s period	3.4	3.6–4.0	4% max. axial expansion; fuel damaged but intact
M4	U–5Fs/316SS; Mark-II design	0.0	8-s period	3.8	4.3	4% max. axial expansion; fuel damaged but intact
	U–5Fs/316SS; Mark-II design	2.4	8-s period	4.1	4.4	7% max. axial expansion; cladding breached
	U–5Fs/316SS; Mark-II design	4.4	8-s period	3.8	4.3	4% max. axial expansion; fuel damaged but intact
M5	U–19Pu–10Zr/D9; X419,420,421 design	0.8	8-s period	4.3	5.1	1% max. axial expansion; fuel damaged but intact
	U–19Pu–10Zr/D9; X419,420,421 design	1.9	8-s period	4.3	5.1	2% max. axial expansion; fuel damaged but intact
M6	U–19Pu–10Zr/D9; X419,420,421 design	1.9	8-s period	4.4	4.6	2–3% max. axial expansion; fuel damaged but intact
	U–19Pu–10Zr/D9; X419,420,421 design	5.3	8-s period	4.4	4.5	3% max. axial expansion; cladding breached
M7	U–19Pu–10Zr/D9; X419,420,421 design	9.8	8-s period	4.0	4.4	3% max. axial expansion; cladding breached
	U–10Zr/D9; X425 design	2.9	8-s period	4.8	4.4	2–4% max. axial expansion; fuel damaged but intact
MOX fuel TREAT tests (taken from Wright et al. [53] and Alderman and Pitner [68])						
RFT-CAL-L	$(U,Pu)O_2$/316SS[b]	0	\$1/s	N/A	N/A	Calibration test; fuel damaged but intact
	$(U,Pu)O_2$/316SS[b] simulated fuel pin cracking	0	\$1/s	N/A	N/A	Calibration test; fuel damaged but intact
	$(U,Pu)O_2$/316SS[b] solid and annular pellets	0	\$1/s	N/A	N/A	Calibration test; fuel damaged but intact
RFT-L1	$(U,Pu)O_2$/316SS[b]	0.2	\$0.5/s	5.9	N/A	Fuel damaged but intact
	$(U,Pu)O_2$/316SS[b]	0.2	\$0.5/s	5.7	N/A	Fuel damaged but intact
	$(U,Pu)O_2$/316SS[b]	0.2	\$0.5/s	9.3	N/A	Cladding breached
RFT-L2	$(U,Pu)O_2$/316SS[b]	0.2	\$0.05/s	3.7	N/A	Fuel damaged but intact
	$(U,Pu)O_2$/316SS[b]	0.2	\$0.05/s	1.8	N/A	Fuel damaged but intact
	$(U,Pu)O_2$/316SS[b]	0.2	\$0.05/s	3.4	N/A	Fuel damaged but intact

RFT-L3	$(U,Pu)O_2$/316SS[b]	2.7	\$0.05/s	2.3	N/A	Fuel damaged but intact
	$(U,Pu)O_2$/316SS[b]	5.3	\$0.05/s	2.0	N/A	Fuel damaged but intact
	$(U,Pu)O_2$/316SS[b]	5.3	\$0.05/s	1.9	N/A	Fuel damaged but intact
RFT-L4	$(U,Pu)O_2$/316SS[b]	2.7	\$1/s	8.5	N/A	Fuel damaged but intact
	$(U,Pu)O_2$/316SS[b]	5.3	\$1/s	7.5	N/A	Fuel damaged but intact
	$(U,Pu)O_2$/316SS[b]	5.3	\$1/s	7.3	N/A	Fuel damaged but intact
TS-1	$(U,Pu)O_2$/316SS[b]	0.2	\$0.05/s	3.0	N/A	Margin-to-failure test; cladding breached in upper 1/3 of fuel column; internal gas pressure-induced failure at elevated cladding temperature
TS-2	$(U,Pu)O_2$/316SS[b]	5.8	\$0.05/s	3.4	N/A	Margin-to-failure test; cladding breached in upper 1/3 of fuel column; internal gas pressure-induced failure at elevated cladding temperature
CDT-1	$(U,Pu)O_2$/HT9[c]; rod ACO-1-094 solid pellets	12.5	\$0.05/s	4.5	N/A	Fuel damaged but intact
CDT-2	$(U,Pu)O_2$/HT9[c]; rod ACO-1-096 solid pellets	11.5	\$1/s	16.5	N/A	Fuel damaged but intact
	$(U,Pu)O_2$/HT9[c]; rod FO-2-J01 solid pellets	6.2	\$1/s	16.5	N/A	Cladding breached due to melt-through just above midplane
	$(U,Pu)O_2$/HT9[c]; rod FO-2-LO2 annular pellets	6.4	\$1/s	16.5	N/A	Cladding breached due to melt-through just above midplane
CDT-3	$(U,Pu)O_2$/HT9[c]; rod FO-2-LO4 annular pellets	6.3	\$0.05/s	4.5	N/A	Fuel damaged but intact

[a] For the metal fuel tests, the overpower value is reported as peak linear power attained in TREAT test relative to nominal power typical of fast reactor application (~40 kW/m). For the MOX tests, the reported value is ambiguous, apparently reported as peak linear power attained in TREAT test relative to nominal power relative to the nominal power for the test rod in prior FFTF irradiation, presumably at beginning of life; this value can be taken to be indicative of a value nominal to a fast reactor. There are some discrepancies in the MOX overpower values reported in Ref. [48,62], but these are small and do not change the interpretation of the tests.

[b] FFTF standard driver design.

[c] FFTF Core Demonstration Experiment designs, see Bridges et al. [17]. Previously irradiated in the FFTF assemblies ACO-1 and FO2, as indicated.

The results consistently showed that metal fuel rods of modern design exhibited failure thresholds around 4 times nominal power (under the relatively fast transient-overpower conditions used in the tests). Fuel rod breaches that occurred were located at the top of the fuel column and in all cases were attributed to cladding rupture induced by at-temperature pin-plenum pressure and cladding thinning due to eutectic-like formation of a molten fuel/cladding phase that penetrated the cladding wall. Pre-failure axial fuel expansion (which has the beneficial effect of removing reactivity from the core during an overpower transient) for the U–Pu–Zr and U–Zr was similar to, though less than, that observed with higher-burnup U–5Fs fuel, and in amounts significantly greater than would be caused by thermal expansion alone. Post-failure behavior observed in all tests was characterized by rapid fuel dispersal, with about half of the fuel inventory being ejected from the fuel rod – again, with the beneficial effect of removing reactivity from the core during postulated sever accidents. The data from these tests and from a large number of previous metal fuel transient test in TREAT were used to develop and validate models of fuel behavior under transient overpower conditions [26,54].

Other safety-related testing focused on fuel behavior during unlikely loss-of-flow events, using hot-cell furnace heating tests of irradiated U–Pu–Zr clad in HT9 [55,56]. The results demonstrated significant safety margin for the particular transient conditions studied (a bounding unlikely loss-of-flow event for EBR-II). The observed cladding breaches were induced by pin-plenum gas pressure at temperature, with cladding thinning due to eutectic-like formation of a molten phase at the fuel/cladding interface. In addition, fission gas expansion in the fuel induced axial fuel expansion, enabled by reduction of constraint from the cladding with the formation of the molten phase at the fuel/cladding interface. The data from these tests, and other similar tests, were used to develop and validate models of fuel behavior under loss-of-flow conditions [54,55].

3.3. Perspective on the significance of the metal fuel database

Recent programmatic discussion in the US has considered whether metal fuel technology is sufficiently mature to allow its use in a newly constructed test reactor. So, the present authors offer the following perspective.

The development of U–Pu–Zr fuel was evolutionary in nature, with fuel design based on that established for U–5Fs driver fuel in EBR-II. The design concepts were validated with the successful conversion of the EBR-II driver core to U–Zr fuel, and the subsequent utilization of over 13 000 driver fuel rods with a 10 at.% burnup limit. The addition of Pu to form the ternary U–Pu–Zr alloy did not change the mechanisms that control fuel element lifetime [57], although the Pu addition and other characteristics of higher-burnup fuel exacerbate the effects of some of those phenomena (e.g., fuel/cladding interdiffusion and fuel constituent migration).

So, although the number of U–Pu–Zr suggests a limited number of rods have been evaluated (relative to the number of rods supporting the MOX fuel database, discussed in the next section), the work performed was sufficient to determine the behavior of life-limiting and safety-related phenomena, which are the same as those for the U–Fs and U–Zr fuel and with which there is substantial reliability and fabrication experience. The metal fuel database in 1994 was sufficient to support the preparation and initial review of the safety case to convert the EBR-II driver core to U–Pu–Zr (Mark-V/VA) fuel [41]. Any ambiguities in fuel behavior mechanisms were accommodated with conservative application of uncertainties in the analytical derivation of Limiting Conditions of Operation for EBR-II. A fuel qualification program was prepared for the purpose of ensuring that U–Pu–Zr fuel performance would be adequately enveloped by the safety case, to establish fuel failure thresholds, and to gather data necessary to support the anticipated extension of the burnup limit beyond the 10 at.% interim value [58].

4. Mixed oxide fuel performance phenomena

The burnup limitations on early metal fuel designs initially led developers toward other fuel types. Solid solution uranium–plutonium mixed oxide (MOX) fuels, though unproven in the 1960s, appeared attractive for SFR application due to its high-temperature potential, the irradiation stability, and technological similarity with uranium oxide, which was being implemented industrially for light water reactors [21,25]. By the late 1970s, when sufficiently low uranium prices and non-proliferation policy in the US prevented implementing fuel

recycle in fast reactors, economic competitiveness with light water reactors became an objective for SFRs [30], and this further motivated achieving the full burnup potential of MOX fuel.

The low-thermal conductivity of MOX fuel leads to relatively high-temperature operation (i.e., the fuel centerline operates at relatively high-homologous temperature, or $T/T_{melting}$), and this leads to fuel restructuring with increasing burnup [21,32]. The enhanced diffusion of gaseous and volatile fission products at high temperature also allows fairly high fractions of fission gas release to the fuel rod plenum, increasing the pressure-induced stress of the cladding. At higher burnup, accumulation of solid fission products in the fuel will lead to fuel swelling, which can also induce stress in the cladding as discussed below.

Early MOX testing and development showed that fuel/cladding chemical interaction (FCCI) and fuel/cladding mechanical interaction (FCMI) were the phenomena that limited the lifetime of MOX fuel designs [21,32,59,60]. FCCI in MOX fuel, as with metal fuel, acts to reduce the load-bearing thickness of the cladding wall, which provides locations for stress-induced cladding breach. Initial FCCI effects were attributed to segregation of volatile fission products (such as Cs, Te, I) to fuel/cladding gaps and pellet–pellet interfaces, which would cause corrosion of the cladding [21]. Other FCCI effects thought to induce breach included attack in the upper UO_2 insulator-pellet region; the high oxygen-to-metal (O/M) ratio of the UO_2 insulator pellets was thought to have led to local attack of the cladding, which was exacerbated by swelling-driven growth of the fuel column which placed higher-temperature fuel pellets near the attacked region, leading to failure. These effects were brought under control primarily through adjustment of the fuel O/M ratio. FCCI remains a phenomenon of interest in modern MOX fuel designs, but is not fuel-life limiting of itself. FCCI in modern design is attributed to oxidation of the cladding constituents due to the increasing oxygen potential in the fuel with increasing burnup, which arises because not all fission products (for example, noble metal fissions products – about 25% of the plutonium fission yield) form compounds with oxygen freed up from the fissioning U or Pu, leaving excess oxygen in the fuel [61].

FCMI in fast reactor MOX fuel results when fuel pellets swell faster than the cladding swells and/or creeps away from the pellets, allowing the pellets to locally stress the cladding. Because MOX fuel operates at sufficiently high temperature (due to its relatively low-thermal conductivity) to allow plastic deformation of the fuel, FCMI can be reduced if the swelling fuel has sufficient space to accommodate plastic flow within the fuel volume (e.g., into available as-fabricated porosity). Therefore, experimental irradiation programs in EBR-II (e.g., [59,60]) were used to assess the effects of various fuel design parameters, such as smeared fuel density (the fraction of cross-sectional area taken up by the fuel pellet cross-section, accounting for percentage of theoretical density and fuel/cladding gap) and amount of porosity in the fuel. For a given fuel design, combinations of these parameters, including occasional incorporation of a centerline hole in the fuel pellets, were shown to reduce FCMI. Another mechanical phenomenon of consequence in the EBR-II tests was the rod–bundle interactions with the duct when cladding diametral strains were higher, such as with higher-swelling cladding and at higher burnup, to which a number of fuel rod breaches were attributed. The use of lower swelling cladding alloys, such as the austentic D9 or the ferritic–martensitic HT9 was expected to make FCMI a life-limiting phenomenon because the cladding would swell or dilate less than the fuel at higher burnups [62].

5. Irradiation performance database for MOX fuel

5.1. Steady-state performance: reliability and burnup potential

The bulk of US experience with irradiation of SFR MOX fuel was obtained initially with a test program in EBR-II [59] to establish the reference MOX fuel design for FTFF and the Clinch River Breeder Reactor and with an advanced oxide fuel program [60] to evaluate designs that would accommodate higher-burnup operation and improved fuel cycle economics. More extensive experience followed with operation of FFTF using MOX driver fuel and with a large array of FFTF tests [62,63]. These important irradiation test programs are summarized in Tables 5–7. A range of MOX fuel parameters were evaluated in the EBR-II tests, such as the Advanced Oxide fuel tests [60], over a range of conditions expected in subsequent operation of the FFTF. The breaches occurring in the EBR-II tests were attributed to FCMI for rod designs with higher fuel smeared

Table 5
Summary of mixed oxide fuel irradiation experiments in EBR-II to evaluate reference design (taken from Leggett et al. [59])

Test ID	Purpose	Fuel composition and oxygen-to-metal ratio	Cladding material	Cladding OD (cm)	Wall thickness (cm)	Peak power (kW/m) (beginning of life)	Peak cladding temp (°C) (beginning of life)	Peak burnup (at.%)	Fast fluence ($\times 10 * 22$ n/cm^2) ($E > 0.1$ MeV)
Reference fuel design tests									
PNL-9	Statistical test of reference design; RTCB[a]	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.584	0.038	18.5	510	9.6	11
PNL-10						30.4	634	7.1	5.1
PNL-11						40.0	557	11.9	9.9
HEDL H/E	Statistical test of reference design w/comm. fab.	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.584	0.038	41.8	647	4.7	3.1
HEDL H/F						38.8	633	5.1	3.3
PNL-13	Effects of wire pitch and clearances on vibration and wear	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.584	0.038	43.9	649	16.6	11.3
PNL-14						45.3	664	12.5	10.3
PNL-14A						29.9	624	9.9	11.8
P-12A	Effects of cladding cold work and internal pressure	$(U–0.25Pu)O_2 \pm 0.05Pu$	10,20,30%CW 316SS	0.584	0.038	39.8	738	8	4.6
P-12AA						39.9	730	4.1	2.2
P-12AB						39.8	738	8.3	4.8
P-15	Comparison of vendor FFTF fuel with development test fuel	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.584	0.038	16.1	695	5.7	9.7
P-23A	Reference design at extended operation at high temperature	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.584	0.038	40.1	717	15.7	12.6
P-23B						40.7	769	13	10.2
P-23C						38.3	705	14.6	12
P-23B/C	Confirm FFTF pin performance w/in FFTF constraints	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.584	0.038	40.2	787	11.3	8.0
Grid spaced tests									
WSA-3	Rod parameters and performance in grid spacers	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.584	0.038	32.8	680	12.6	7.8
WSA-4	Grid-spaced rods at high temp and fluence	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.584	0.038	21.3	670	12.8	19.4
WSA-8	Various cladding alloys at high temp	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS, 20%CW 316SS + Ti, 20%CW 321SS,	0.584	0.038	38.7	690	10.7	9.2
F9E	Grid-spaced rods to moderate burnup	$(U–0.25Pu)O_2 \pm 0.05Pu$	316A	0.68	0.038	45.9	604	11.8	7.5

F9F	Grid-spaced rods with reference cladding	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.584	0.038	42.6	649	11.9	7.7
Thermal performance tests									
P-19	Effects of fuel/cladding gap on fuel melting and power-to-melt	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.584, 0.635	0.038, 0.041	65.6	521	0.06	0.1
P-20	Effects of gap size at low burnup on fuel melting and power-to-melt	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.584	0.038	72.2	721	1.2	1
F20	Effects of rod parameters and burnup on power-to-melt	$(U–0.25Pu)O_2 \pm 0.05Pu$	316A, 316C, 304A, 321A	0.64	0.038	68.9	682	0.005–9.0	0.003–5.8
P-17	Instrumented for power and flow	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.584	0.038	28.9	641	3.7	2.7
P-17A	Instrumented; effect of gap size on fuel and coolant temps	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.584	0.038	38.6	695	0.4	0.2
General development tests									
PNL-1	Encapsulated rods of various parameters	$(U–0.25Pu)O_2 \pm 0.05Pu$	304SS	0.635	0.025 and 0.041	37.7	537	1.1	0.8
PNL-2						37.4	537	5.5	3.5
PNL-3	Statistical behavior of fuel swelling and gas release	$(U–0.25Pu)O_2 \pm 0.05Pu$	304SS	0.635	0.041	16.9	462	7	13
PNL-4						30.2	487	10.4	9.3
PNL-5						47.0	576	17.1 (?)	9.9
PNL-6	Statistical performance data for varying dimensional tolerances and processes	$(U–0.25Pu)O_2 \pm 0.05Pu$	316SS	0.635	0.041	20.3	547	9.1	7.9
PNL-7						30.0	545	11.1	10
PNL-8						41.9	586	9.8	5.5
F9A	Behavior for ranges of rod parameters and operating conditions	$(U–0.25Pu)O_2 \pm 0.05Pu$	316A, 304A, 321A	0.64	0.038	42.6	566	19	14
F9B	Behavior for ranges of	$(U–0.25Pu)O_2 \pm 0.05Pu$	316A	0.64	0.038	42.6	566	13.5	9
F9C	Rod parameters and operating conditions					42.6	566	19.5	14
F9D						29.5	560	14	9

(*continued on next page*)

Table 5 (*continued*)

Test ID	Purpose	Fuel composition and oxygen-to-metal ratio	Cladding material	Cladding OD (cm)	Wall thickness (cm)	Peak power (kW/m) (beginning of life)	Peak cladding temp (°C) (beginning of life)	Peak burnup (at.%)	Fast fluence ($\times 10 * 22$ n/cm^2) ($E > 0.1$ MeV)
F10A	Encapsulated rods; effects of high cladding temp and rod variables	$(U–0.25Pu)O_2 \pm 0.05Pu$	316CW, 316A, 316CA, 304CW, 304A, 304CA, 347A	0.64	0.025 and 0.038	45.9	721	11.3	7.5
F10B									
F11A	High-temp performance of various cladding alloys	$(U–0.25Pu)O_2 \pm 0.05Pu$	316CW, 316A, 316CA, 304CW, 304A, 304CA, 321A	0.64	0.038	52.5	732	7	5.7
WSA-1	Effects of various rod parameters	$(U–0.25Pu)O_2 \pm 0.05Pu$	SA 316SS and 20%CW 316SS	0.559 0.643 0.630	0.025 0.032 0.038	27.5	620	11.3	11.4
WSA-2	Effects of various rod parameters	$(U–0.25Pu)O_2 \pm 0.05Pu$	SA 316SS and 20%CW 316SS	0.559 0.630	0.025 0.038	34.8	620	16.9	10.8
WSA-5	Delineate cladding stain mechanisms	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.617	0.025	36.0	648	10.5	6.6
ANL-04	Effects of Rod parameters	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.635	0.041	36.0	620	8.8	4.9
ANL-08	Study FCCI[b] worth two types of O_2 absorbers and PuO_2 contents	$(U–0.25Pu)O_2 \pm 0.05Pu$	20%CW 316SS	0.762	0.047	39.0	704	5.1	5.7

[a] RTCB: Run to cladding breach; i.e., up to the point of breach detection.
[b] FCCI: Fuel-cladding chemical interaction.

Table 6
Summary of advanced mixed oxide fuel irradiation experiments in EBR-II (taken from Lawrence et al. [60])

Test ID	Fuel composition and oxygen-to-metal ratio	Cladding material	Smeared density (%)	Cladding OD (cm)	Wall thickness (cm)	Peak power (kW/m) (beginning of life)	Peak cladding temp (°C) (beginning of life)	Peak burnup (at.%)	Breached rod information[a]
P-40	(U–0.25Pu)O_2 O/M: 1.951–1.972	20%CW 316SS	85.9–88.8	0.584	0.025–0.038	52.0	650	13.4	12.2 at.%, FCCI
P-41R	(U–0.25Pu)O_2 O/M: 1.929–1.943	20%CW 316SS	90.3	0.686	0.028	41.0	590	11.7	11.6 at.%, FCCI 11.7 at%, embrittlement or loss of ductility
P-42R	(U–0.25Pu)O_2 O/M: 1.916–1.945	20%CW 316SS	90.5–92.3	0.686	0.028	46.0	610	6.2	3.9 at.%, FCMI 1.8 at.%, FCMI 2.5 at.%, FCMI
P-43	(U–0.25Pu)O_2 O/M: 1.933–1.955	20%CW 316SS, D9, IN-706	85.4–90.7	0.584	0.025–0.038	36.5	665	17.0	12.7 at.%, FCMI 11.6 at.%, FCMI 5.5 at.%, FCMI
P-50	(U–0.15Pu)O_2 O/M: 1.965–1.968	20%CW 316SS	8.2	0.737	0.041	43.0	625	10.3	9.8 at.%, UO_2-fission product interaction
P-52	(U–0.15Pu)O_2 O/M: 1.973–1.978	20%CW 316SS	88.2	0.737	0.030	43.0	605	12.7	12.5 at.%, cladding hot spot 12.3 at.%, cladding hot spot 7.5 at.%, FCMI 9.3 at.%, cladding hot spot
P-53	(U–0.15Pu)O_2 O/M: 1.968–1.972	20%CW 316SS	88.2	0.737	0.030	24.0	620	8.5	
P-60	(U–0.15Pu)O_2 O/M: 1.966–1.972	IN-706, PE-16	85.6–90.6	0.737	0.061	46.0	720	7.6	

[a] Breach rod information includes peak burnup attained for each breached rod and attributed cause of breach. FCMI: Fuel/cladding mechanical interaction; FCCI: Fuel/cladding chemical interaction.

Table 7
Summary of mixed oxide fuel irradiation experiments in FFTF (taken from Baker et al. [63])

Test category	Primary cladding and duct materials	Fuel or absorber[a]	Rod size and wall[b] (mm)	No. of rods irradiated	No. of assemblies irradiated	Maximum test pin				No. of assemblies with rod breach	No. of pins tested in TREAT
						Burnup (GWd/MTHM)	Fast fluence ($\times 10^{22}$ n/cm^2)	Power (kW/m)	Cladding temp (°C)		
Reference fuel	316SS[c]	UO_2–PuO_2	5.84/0.38	6076	28	122	17	36.0 [53.1]	670	3	14
Extended Lifetime	D9[c], 316SS[c]	UO_2–PuO_2	5.84/0.38, 6.99/0.38, 6.86/0.56	D9: 2377 316SS: 3413	29	188	27	43.9	675	8	0
Long lifetime	HT9	UO_2–PuO_2	6.86/0.56	2996	17	238	39	44.2	661	0	5
Miscellaneous fuel	316SS[c]	(U,Pu)O_2,	5.84/0.38	1357	8	113	15	46.9	615	0 1	0
	D9[c]	UO_2, (U,Pu)C	9.40/0.51	128	2	79	13	86.0	607		
Blankets	316SS[c]	UO_2	12.85/0.38	122	2	14	13	22.0	556	0 0	0
	HT9	UO_2	9.91/0.51	546	6	42	23	39.0	612		

[a] Fuel and absorber columns are 91.4 cm long, blankets are 124.5 cm long, pins are 244 cm long.
[b] 5.84-mm pins used in 217-pin assembly, 6.99- and 6.86-mm in 169-pin assembly, 9.9-mm in 91-pin assembly and 12.85-mm in 61-pin assembly.
[c] All 316 and D9 stainless steel is 20% cold worked.

density ($\geqslant$90%), or to rod–bundle mechanical interactions with the duct [62]. The EBR-II test program suggested that FFTF MOX driver fuel might be limited to 6 at.% burnup, but subsequent FFTF operation indicated that prototype-length fuel rods had enhanced burnup potential over shorter test rods used in EBR-II [30]. The observation seems to be similar to that made regarding metal fuel in FFTF vs. EBR-II: the shorter rods in EBR-II allow a more aggressive combination of fuel and cladding temperatures with local burnup than occurs in prototype-length fuel rods. (In this context, prototypic fuel-rod length is considered to be that with roughly 90-cm fuel column height, where as the EBR-II fueled core height, and thus the fuel column height for test rods, was roughly 35 cm.) Because the FFTF irradiation database is extensive and definitive (from a US perspective), due to the more-prototypic dimensions, that experience will be the focus of this section.

The FFTF irradiated over 63 500 Series-I and Series-II driver fuel rods, which were fabricated by industrial suppliers, with a burnup limit of 80 GWd/MTHM (or around 8 at.%) [63]. No breaches occurred in any driver fuel irradiated to the burnup limit, and in driver fuel irradiated beyond design exposure only one breach was observed, at 103 GWd/MTHM exposure [14]. In total, over 4300 rods of Series I and II driver fuel and similarly designed test fuel were irradiated to burnups greater than 100 GWd/MTHM (or 10 at.%), and a Wiebull analysis (see [12] for a discussion of the technique) of breach/non-breach occurrence as a function of burnup indicated this fuel design had a 99.99% reliability at a peak rod burnup of 100 GWd/MTHM [14]. This suggests that the burnup limit on Series I and Series II driver fuel could have been raised from 80 GWd/MTHM to 100 GWd/MTHM, had FFTF operation required it.

Baker et al. [14,63] summarized the larger FFTF fuel irradiation program, including prototype fuel testing and testing of extended-lifetime fuel designs with D9 austenitic stainless steel for cladding and duct material and long-lifetime fuel designs with HT9 ferritic–martensitic stainless steel for cladding and duct material. D9-clad MOX fuel rods were irradiated to burnups as high as 189 GWd/MTHM, with neutron fluence exposure of 28×10^{22} n/cm^2, in FFTF. But D9-clad fuel rods were ultimately limited to 21×10^{22} n/cm^2, equivalent to around

140 GWd/MTHM burnup, due to embrittlement concerns, as described in the section on cladding and duct materials. D9-clad fuel rods irradiated at higher temperature exhibited cladding strain near the top of the fuel column, coincident with a relatively-high local Cs content [14]. It appeared that Cs had migrated to the top of the fuel column and interacted with the fuel to cause local FCMI, at an axial location where cladding temperatures are high.

The long-lifetime fuel experiments investigated and demonstrated the potential for HT9-clad and HT9-ducted fuel assemblies to achieve 200 GWd/MTHM burnup. A key part of this test series was the FFTF Core Demonstration Experiment [17], which consisted of 10 fuel and six blanket assemblies. Previous to irradiation of the CDE assemblies, five lead-test assemblies were irradiated, and it was one of those assemblies (ACO-4) that attained a peak pellet burnup of 238 GWd/MTHM with peak neutron fluence exposure of 38×10^{22} n/cm^2 with irradiation through FFTF cycle 12-B2. Overall, the CDE assemblies were irradiated to burnup values ranging from around 130 to 164 GWd/MTHM through the end of FFTF cycle 11B. Many of the CDE assemblies apparently received additional exposure through the end of FFTF cycle 12-B2. As was the case with the D9-clad rods irradiated at higher temperatures, as described in the preceding paragraph, the higher-temperature rods in assembly ACO-1 showed some cladding strain at the top of the fuel column, again coincident with migrated Cs content. Again, the strain was attributed to Cs interaction with the fuel, causing local FCMI, at an axial location where cladding temperatures are high [17].

Additional tests in FFTF addressed specific issues, including fuel performance sensitivity to various design parameters and irradiation with beginning-of-life centerline melting [14,63,64]. One assembly (test DE-9) contained 60 MOX rods with enriched uranium (of the 217 total reference driver rods), providing steady-state power generation sufficiently high to melt the fuel along the fuel column centerline. This assembly reached the reference driver goal burnup of about 100 GWd/MTHM before cladding breach. The one breached element was determined to have breached in a manner unrelated to the centerline melting. It was concluded that beginning-of-life centerline melting in MOX driver fuel did not noticeably impact fuel lifetime, thus indicating that certain design constraints applied to FFTF to avoid such fuel melting could be relaxed [14,64]. Another test (designated C-1) investigated the sensitivity of fuel performance to variations in as-fabricated characteristics, such as O/M ratio, pellet contamination, etc., by irradiating such pellets in a low-swelling cladding, austentic stainless steel D9 at relatively high temperatures. The results indicated that the imperfections in the pellets had no adverse effects on performance, suggesting that fuel specifications could be relaxed, potentially reducing cost of fabrication.

As with metal fuel, MOX fuel irradiation at high temperature and the effects of fission products induce phenomena that impact the reliable lifetime of the fuel and safety-related fuel behavior under off-normal conditions. Unlike metal fuel, however, MOX fuel reacts with the sodium coolant if a fuel rod breach occurs (e.g., [65]), and the nature of that reaction brings additional phenomena of interest. The reaction of sodium with uranium–plutonium oxide, assumed to form a sodium urano–plutonate, posed an early concern for use of MOX fuel with respect to operation with breached fuel [25,30,64,66]. Postulated consequences of operating with breached fuel included fuel loss into the coolant due to 'washout', exacerbation of cladding breaches due to cladding stresses induced by the reaction product's higher specific volume, propagation of breaches to multiple rods due to localized hot spots in a fuel rod bundle, and formation of local coolant blockages due to aggregation of washed out fuel and reaction product. To investigate those issues, the Run-Beyond-Cladding-Breach (RBCB) test program, jointly sponsored by Japan and the US, conducted a series of experiments in EBR-II for the purpose of understanding the nature and consequences of fuel/coolant interactions after a MOX fuel rod breach [66]. This test series, summarized in Table 3, included 18 test irradiations with 1, 7, 19, or 37 pre-irradiated rods each. Some rods were irradiated to their natural breach occurrence, while others were pre-defected to induce an earlier breach. In total, 30 rods breached during reactor operation, and post-breach operating durations ranged from 2 to 152 days.

Observations of post-breach fuel behavior and the reaction product were consistent with envisioned effects for sodium urano-plutonate, and under certain circumstances (e.g., low-smeared fuel density and/or low-swelling cladding) the expanded volume of the reaction product further opens fuel breaches, raises fuel temperatures, and can cause

secondary breaches, all leading to fuel release into the coolant [65,66]. However, the general conclusion of the program was to determine that fuel breaches in MOX fuel rods were predominantly benign events, with fuel loss being typically small, release being suppressed by the formation of a coherent reaction product layer, and with no propagation of breach to adjacent fuel rods. The latter result seems the most significant, as it indicates a low-probability breach event will have limited consequences. Yet, the investigators noted that irradiation embrittlement of cladding or higher-operating temperature could limit benign operation. The program also determined that post-breach reactor shutdown need not be immediate, but a reactor could typically continue to operate at power until a convenient shutdown time, perhaps for days, weeks, or even months [20,30,66].

5.2. Transient and off-normal behavior

The Operational Reliability Test (ORT) program in EBR-II, described in the previous section on transient performance of metal fuel, was conducted to evaluate the performance and reliability of MOX fuel under slow-ramp, operational transients [20,21,45]. This program conducted four extended overpower transient tests on assemblies of pre-irradiated MOX fuel from the US MOX fuel development program, incorporating rods of conservative, moderate, and aggressive design. The most recent of these tests used fuel rods with burnup values ranging from 2.5 to 9.7 at.% and subjected them to a 65–99% transient overpower [45] excursion. Extensive melting occurred in all the tested rods, but none of the rods breached. It was also observed that the fission gases exerted stress in the radial direction that strained the cladding and in the axial direction to move the fuel upward, to varying degrees. An important conclusion of the test series is that the MOX fuel rods are able to survive extended overpower and duty-cycle transients [20].

FFTF driver fuel designs and other experimental oxide fuel designs were also tested in the TREAT reactor under transient overpower conditions, [53,67,68]. As is true for metal fuel, the more recent and more-definitive test series were built upon experience with previous test series that are not addressed here. Those test series included capsule tests and single-pin flowing-sodium loop tests before the testing required the sophistication of multi-pin flowing-sodium loop tests [53]. The most relevant of these tests are summarized in Table 4.

The first of the more relevant test series is the PFR/TREAT series, which was a collaboration between the United Kingdom Atomic Energy Authority (UKAEA) and the US Department of Energy (USDOE). The design features of the UKAEA PFR fuel were somewhat different than those used in FFTF (e.g., the PFR rods had the fission gas plenum below the fuel column rather than above), but were considered relevant for full-length designs. A total of 55 rods, with burnups of 0, 4 and 9 at.%, were tested under transient-overpower (TOP) and transient-undercooling-overpower conditions. (The latter conditions included coolant voiding in addition to an overpower burst.) The next tests were the Reference Fuel Transient (RFT) series and the Transient Safety (TS) series performed on FFTF Series I/II-driver-design rods (i.e., full-length fuel rods in 316SS cladding) irradiated in the FFTF. A total of 17 rods, with burnups of 0, 0.2, 2.7, and 5.3 at.%, were tested in six separate tests. The results of these test series indicated that rod breach thresholds, in 316SS-clad rods, were 1.8–7.3 times nominal, steady-state power – comfortably above the 1.25 times overpower limit at which the secondary plant protection system would terminate a transient, preventing damage to the core. Other results included observing that cladding breach typically occurred near the top of the fuel column or otherwise above core midplane. In the PFR/TREAT tests, some flow blockage was observed and attributed, possibly, to the relatively small fuel channels in those tests, suggesting that coolability could be inhibited in a full assembly [53,67]. Similar behavior was not reported for the RFT, TS, or CDT (discussed below) test series.

It was also observed that fuel rods tested at faster ramp rates exhibited higher-breach margins, with breach just above the core midplane where fuel temperatures would be higher, while the slower-ramp tests induced breach nearer the top of the fuel column, where cladding temperatures would be higher. The faster-ramp tests are essentially adiabatic, and fuel melting leads to cladding penetration before the cladding temperature increases sufficiently to reduce cladding strength, whereas the slower-ramp transients allow cladding temperatures to increase such that the weakened cladding is more like to breach by stress rupture.

Fuel rods irradiated in FFTF as part of the Core Demonstration Experiment [17] were tested under

transient-overpower conditions in TREAT [14,53,68]. Five rods were used in three tests, after irradiation to burnups of 62, 110, or 115 GWd/MTHM. The rods included fuel with solid and annular fuel pellets. The results in these tests were consistent with results of the RFT and TS series, in that fuel breaches occurred (only in the CD-2 test) due to fuel melt-through of cladding in the fast-ramp tests, in a region just above the core midplane. Significant fuel motion was also reported [68], with molten fuel being moved within the cladding upward from the fuel column into the plenum, a region of lower reactivity worth, suggesting that fuel behavior during a transient-overpower event would help to reduce the magnitude of a positive reactivity insertion. The overpower margins to breach deduced from the slower-ramp tests and faster-ramp test were 4.5 and 16.8 times nominal, steady-state power, respectively. These tests confirmed the robust transient overpower margin, under the tested conditions, of mixed oxide fuel in HT9 cladding. It was also observed that the annular fuel had a 20% greater induced fuel enthalpy than did the solid fuel tested at the slower ramp rate, but exhibited less cladding strain, attributable to the annular fuel placing less stress on the cladding because there was more space available to accommodate plastic fuel deformation [53].

Post-breach phenomena are complex and beyond the scope of this paper; however, certain phenomena should be noted [69]. Location of the breach is important, as it determines the location at which fuel leaves the fuel rod and enters the coolant. The phenomena associated with the faster-ramp transients suggests that fuel leaving the fuel rod from near core midplane allows other fuel within the rod to be driven toward the expulsion site, putting more fuel into the higher-worth region of the core center. Local coolant flow can be briefly stopped by pressure transients induced by sodium boiling that occurs when hot oxide fuel contacts the coolant, leading to further fuel damage in extreme cases. Post-breach motion of fuel into the coolant channel, and mixing with molten steel in the more severe events, seems to form blockages to further complicate removal of heat from the debris [67]. The literature on the MOX fuel TREAT tests does not address post-breach phenomena in detail, perhaps because the overpower margins to breach are comfortably high for the accidents for which the plant protection systems function as designed. The conclusions state the tests show the overpower-transient behavior to be acceptable and the models developed at that point to be conservative [53,68].[3]

6. Irradiation performance of cladding and duct materials

As can be seen with review of the irradiation experience described in previous sections, cladding behavior is a key aspect of fuel performance. And it was really the incorporation of low-swelling alloys into cladding that allowed the big gains in burnup capability. These developments came as the result of a pointed effort to understand the mechanisms responsible for swelling, embrittlement, and irradiation creep in metal alloys [31]. The application of low-swelling alloys to assembly ducts was also an important development, as duct exposure effects were the life-limiting phenomena in driver fuel assemblies in both EBR-II [31,70] and FFTF [15,20,71].

Cladding and duct materials for long-life fuel assemblies are selected in large part for swelling and creep (thermal and irradiation-induced) resistance. Resistance to swelling in austenitic stainless steels can also indicate resistance to embrittlement at high-neutron doses as it has been observed that swelling at ~10% by volume results in extreme loss of ductility [72,73]. Ferritic–martensitic (FM) alloys are very resistant to swelling and irradiation creep and ducts fabricated from HT9 have been irradiated in FFTF to neutron fluences of 3.8×10^{23} n/cm^2, ($E > 0.1$ MeV) [17,20]. However, some of these materials can develop 'notch' sensitivity – embrittlement and an increase in ductile-to-brittle transition temperature (DBTT) upon irradiation to high exposures at low temperatures [74]. The high-operating temperatures (in areas of high dose) and relatively thin walls of fuel cladding tend to minimize these problems for fuel applications. However, the lower operating temperatures and potentially thicker walls of ducts can allow these effects to influence material choices, component design, or allowable exposures and handling temperatures.

[3] The more severe behavior alluded to here is associated with hypothetical core-disruptive accidents (HCDAs), which have very low probability of occurrence and for which the treatment in safety and licensing was a matter of debate. Background on the approaches used to address these events can be found in [94,95].

6.1. EBR-II and FFTF experience with duct materials

The life-limiting phenomenon for fuel assemblies in EBR-II was duct deformation caused by creep and swelling, because too much dilation (or bow, as described below) would prevent an assembly from fitting through the receiving rings in the EBR-II storage basket during fuel handling operations. The duct walls in EBR-II were very thin (1.0 mm), and pressure-drop-induced stresses caused the ducts to dilate from creep as well as swelling. Therefore, replacing the original 304 stainless steel with a more swelling resistant alloy, cold-worked 316, produced only a modest gain in allowed exposures/lifetimes. The reactor thimbles, which acted as guide tubes for the control and safety rods, had no such pressure drops and therefore deformed primarily due to swelling alone. For this reason, these thimbles exhibited considerably less deformation (which was fortunate, because the thimbles were more difficult to replace than standard assemblies). In those applications, HT9 was being used as a replacement for 316SS thimbles, and exposure limits were set at 3×10^{23} n/cm^2 ($E > 0.1$ MeV). The thin walls mitigated embrittlement effects. No HT9 thimbles had been replaced at the time that EBR-II was shut down.

EBR-II ducts not only deformed in cross-section but also changed shape by bowing orthogonal to their axial dimension, and severe bow also made insertion into the storage basket very difficult. Radial gradients in neutron flux and temperature often caused radial gradients in swelling or creep rates and hence bow of the assembly in the direction of these gradients [70,75]. As the assemblies encountered a neighboring assembly stresses could develop that caused bow in the other direction and the assembly formed an 'S' shape [75]. Use of low-swelling/creep materials would have minimized these effects. Rotation of reflector and blanket fuel assemblies one or more times during their lifetime in EBR-II allowed the bow effects to be mitigated.

The criteria by which fuel assembly lifetimes were limited in FFTF included fuel-handling forces, as dilation or bow would increase the pressure between assemblies, and the in-vessel fuel-handling machine had limits on the force that could be accommodated in fuel handling. FFTF duct deformation could be caused by bundle and duct interaction, particularly in higher-temperature assemblies in which irradiation creep might be more active; however, due in part to the relatively high thickness of the FFTF duct wall (2.54 mm) this creep phenomenon appears to have been secondary to the deformation caused by swelling. In FFTF duct bow was minimal due to the heterogenous nature of the core, sufficient core restraint from neighboring assemblies, and rotation of assemblies during fuel handling. The other phenomenon of interest was FFTF duct elongation, as it was necessary to maintain clearance above the top of the core for rotation of the instrument trees.

Exposure of 20% cold-worked 316SS ducts in FFTF was limited to 12×10^{22} n/cm^2 ($E > 0.1$ MeV) to ensure that fuel handling loads remained comfortably below limits, and this allowed fuel utilization up to the 80 GWd/MTHM burnup limit. Subsequent testing indicated that the Series I and II fuel rods were capable of higher burnup [18], and later evaluation indicated that those rods were capable of 100 GWd/MTHM and a duct exposure limit of 17.5×10^{22} n/cm^2 ($E > 0.1$ MeV) would likely allow the additional fuel burnup while maintaining fuel handling loads within the limits [15]. It appears that the higher burnup and exposure limits were not fully implemented at FFTF, although the evidence in the literature appears to provide a basis for making such a change.

6.2. EBR-II and FFTF experience with cladding materials

Fuel cladding performance is complex, as the fuel pin design and the fuel material properties and operating performance have as much influence on attainable fuel pin lifetime as do the cladding properties. However, as fuel materials/designs evolve the cladding performance may again become limiting, especially if higher fuel operating temperatures and very long exposures are desired. All austenitic stainless steel alloys eventually swell as neutron irradiation doses increase, and even if the assemblies were designed to accommodate swelling strains, when these materials swell to 10 vol.% or greater they are likely to become very brittle (near zero ductility) [73,74].

D9, a swelling-resistant austenitic stainless steel developed for cladding use showed promise as an EBR-II metal fuel experiment using D9 cladding achieved 20 at.% burnup. But D9 also eventually swells and exhibited embrittlement at higher exposures in FFTF. This raised issues for fuel handling, as indicated by a brittle fracture observed in D9-clad fuel rods during handling in a hot cell; as a

result, D9-clad fuel rods were limited to neutron fluence of 21×10^{22} n/cm^2 ($E > 0.1$ MeV), which was less than fuel breach behavior alone would have allowed [63].

These potential limitations for austenitic alloys have driven cladding material development to the ferritic–martensitic alloys [76] and oxide dispersion-strengthened (ODS) varieties where very high-cladding temperatures are required. This class of stainless steel materials has allowed fueled irradiation experiments to achieve burnups greater than 200 GWd/MTHM (or 20 at.%). The limitation for ferritic–martensitic alloys has been the stress-rupture strength, limiting steady-state operating temperatures to not much more than 600 °C. Optimizing types and quantities of alloying additions has improved these properties [76], and ODS alloys may extend useful operating temperatures even further.

7. Advanced ceramic fuels

Considered as advanced fuel concepts, carbide and nitride fuels have been investigated as alternatives to metal and oxide fuels for LMFBR applications since the 1960s. Initial interest was motivated by the high metal atom density of both the carbide and nitride fuels relative to the oxide, which makes possible a substantially higher-breeding ratio and lower doubling time. Interest in these fuels was sustained well into the 1980s due to their attractive combination of thermophysical properties relative to both metal and oxide fuels: namely, high-fissile density, high-thermal conductivity, and high-melting point [5,20,21,59]. Additionally, both carbide and nitride fuels can be fabricated with either a small, helium-bonded fuel/cladding gap or a large, sodium-bonded gap. In many respects, they have been viewed as combining the best attributes of both metal and oxide fuels [77].

With the successes of both the metal and oxide fuels demonstrated through the 1980s and early 1990s, and the considerable technology infrastructure built up in support of these two fuel types, interest in carbide and nitride fuels has waned. UN had been developed for compact, high-temperature space reactor applications in the 1980s and early 1990s [20], and interest in that fuel for such applications continues. In addition, nitride fuel with relatively high-minor actinide content is being considered for fast reactor transmutation; e.g., [78]. Therefore, the results of previous US experience with carbide and nitride fuels merit inclusion here.

7.1. Mixed carbide fuels

Although much smaller than that for either metal or oxide fuels, the irradiation performance database for (U,Pu)C mixed carbide (MC) fuels is considerable [5,79]. Over 470 MC fuel rods were irradiated in EBR-II using a range of parameters, sodium or helium bonding, and cladding made from Type 316 stainless steel, PE-16 (a nickel-based alloy used in the U.K.), D9 stainless steel, or D21 stainless steel [79]. Over 200 MC fuel rods were irradiated in FFTF in two assemblies: the ACN-1 experiments with rods fabricated using Type 316SS and D9 cladding and the FC-1 test, which was a full-size, 91-rod FFTF assembly using Type 316SS and D9 cladding and ducts [5,79–81]. The AC-3 test was composed of 91 full-size, D9-clad rods of which 25 rods contained sphere-pac fuel and 66 rods contained pellet fuel [80–82]. That assembly was irradiated to the goal 9 at.% burnup without breach.

Considerable cracking was observed in the irradiated MC fuels; this results from the fact that relatively high-thermal conductivity carbide fuel typically operates at a fairly low-homologous temperature where its brittle ceramic nature cannot withstand even the modest tensile stresses formed under a thermal gradient. However, such cracking and resulting fuel relocation has not been observed to cause premature pin failure. Of the 21 fuel breaches that occurred in the EBR-II tests prior to reaching goal burnup, 15 were in PE-16-clad rods as locations under the wire wrap and attributed to irradiation embrittlement in that cladding alloy rendering the cladding less capable of enduring the stress induced by FCMI and fission gas pressure. (The other six were observed to be in Na-bonded rods, but those results were deemed less relevant because emphasis was placed on He-bonded MC fuel as Ref. [80].) One rod breached in FFTF, but there was no post-irradiation examination of that rod. The FC-1 FFTF experiment (a full-size, 91-rod FFTF assembly) attained goal burnup with breach. A peak fuel burnup of 20 at.% in 10 MC fuel rods clad in Type 316 stainless steel was achieved in EBR-II [30,83]. Of those rods, five had experienced a 15% transient-overpower test in EBR-II after attaining 12 at.% burnup. Thirteen other He-bonded rods and three Na-bonded rods attained 16 at.% burnup in

EBR-II without breach. The FFTF AC-3 experiment results showed that, for the relatively low-temperature conditions used for the test, the pellet fuel and sphere-pac exhibited only minor observed differences in behavior, and both performed in a manner consistent with the rest of the MC fuel database [81].

Carbide fuel failures typically result from FCMI, owing to the fact that fuel swelling is greater than that of the oxide which leads to early fuel/cladding gap closure, and because it generally operates at relatively low-temperature fuel creep is not effective at relieving cladding stress [80]. For this reason, MC fuel pin design must incorporate a large fuel/cladding gap and make use of a low-density fuel in order to delay the onset of FCMI. While cladding carburization has been an historical concern for MC fuels, and was observed in the Type 316SS-clad rods irradiated in EBR-II, no fuel failures have been attributed to this phenomenon [5,80].

Ten transient-overpower tests involving MC fuels were conducted in TREAT using fuel irradiated in EBR-II to burnups ranging from 0 to 12 at.%, primarily for the purposes of establishing that cladding breach would occur at a margin above that of the FFTF plant protection system settings (at 115% and 125% overpower). The results suggested FCMI-induced breaches, but most importantly indicated comfortable margins to failure (roughly 3 times nominal linear heat generation rate in a MC fuel and up to 6 times the nominal linear heat generation rate typical of oxide fuel cores). The rods indicated only small cladding strains and small amounts of liquid-phase penetration of the cladding. The conclusion of the test series was a determination that nothing in fuel transient-overpower response would prevent or limit application of MC fuels to fast reactors [30,80].

The EBR-II tests also included rods irradiated beyond goal burnup to breach and one intentionally-defected rod irradiated for 100 days beyond cladding breach. The intentionally defected rod exhibited a reaction between the fuel and the coolant (presumed to be the oxygen within the coolant) that resulted in a higher-specific-volume reaction product that caused expansion and widening of the defect. However, it was observed that little fuel was released from the cladding into the coolant. Other rods irradiated to natural breach in EBR-II did not exhibit that phenomenon. MC fuels appear to operate benignly after cladding breach [30,80]. Another experiment in EBR-II irradiated a MC fuel rod with a purposely-induced Na-bond void intended to simulate a Na bond void resulting from Na expulsion during irradiation, and although the fuel rod exhibited microstructural changes reflecting a local high-fuel temperature, there was no loss of cladding integrity. This experiment indicated that MC fuel would withstand a Na bond expulsion of some magnitude [30,80].

Overall, the US experience base with MC fuels, while not large, was sufficient to instill confidence that such fuels have irradiation performance adequate for use in SFRs. In particular, the He-bonded rods clad in lower-swelling cladding alloys with around 80% fuel smeared density appear to show the best performance potential. Other issues associated with fuel recycle technology and reactor safety have been considered, but are not addressed here.

7.2. Mixed nitride fuels

The irradiation performance database for (U,Pu)N mixed nitride (MN) fuels is substantially smaller than that for MC fuels. Compared to MC fuels, MN fuels exhibit less fuel swelling, lower fission gas release, and are considerably easier to reprocess; however, the problem of the production of biologically hazardous ^{14}C in nitride fuels fabricated using natural nitrogen poses a considerable concern for the reprocessing of MN fuels [84]. Nevertheless, the attractive properties of nitride fuels have continued to point to a strong potential for their benefit and to motivate interest [30,85].

More so than MC fuels, MN fuels were found to exhibit extensive cracking and fragmentation during simple startup and shutdown transients if operated at high temperatures. This cracking phenomenon was considered to be the reason for the early fuel failures seen in US irradiation tests [86]. Use of a metallic shroud around the fuel column to prevent fuel relocation subsequent to fragmentation provided an engineering solution to this problem to some extent. Nevertheless, the US specifications for MN fuels recommended that their peak fuel temperatures be restricted to as low as 1200 °C to mitigate fragmentation and eliminate the need for the use of shrouds [85,86]. A further issue of concern regarding MN fuels was that they dissociate at temperatures substantially lower then their congruent melting point if a nitrogen overpressure is not maintained.

8. Future applications for fast reactor fuels

Recent program emphasis on actinide management in the US and abroad has technologists investigating SFR systems with minor actinides recycled from light water reactors and from fast reactors operated with conversion ratios considerably less than 1 [1,2,4,87]. Adaptation of the existing SFR fuel experience to the envisioned applications appears feasible, but several technical issues must be addressed (for example, [79,88,89]). Among these issues are those associated with fabrication of fuel containing minor actinides in remote (shielded) environments and in a manner that does not result in loss of volatile americium as a secondary, process waste. There will be fuel performance issues as well, driven by the effects of differing and challenging compositions and by the physicists' desire for increasingly high-fuel burnup. Some of these issues were just starting to be considered in the 1990s when programs in the US and the UK were terminated, but efforts in Japan, France, India, and Russia have continued. The US re-initiated fast reactor fuel work with the advent of the Advanced Fuel Cycle Initiative, and recent results [90,91] have confirmed that today's objectives for fuel technology are challenging, but not insurmountable.

9. Summary of US fast reactor fuel experience

The US fast reactor fuel experience reflects the efforts of a large number of organizations from government, national laboratories, industry, and academia, all of whom have worked from 1947 until now (with a brief hiatus in the 1990s) as part of the nation's broader effort to develop fast reactors with fuel recycle. A strong emphasis on preparing the technology for commercialization in the 1970s and 1980s led to extensive test programs in EBR-II, FFTF, and TREAT, which were performed alongside similar efforts in other nations and included international collaboration. Table 8 presents a comparative summary of the experience with metal, MOX, and MC fuels. These programs and the associated efforts in materials development and evaluation, fabrication process development, and

Table 8
Summary of US fast reactor fuels experience base

US experience	Fuel type, cladding, and burnup		
	Metal	Mixed oxide $(U,Pu)O_2$	Mixed carbide $(U,Pu)C/(U,Pu)_2C_3$
Driver fuel operation	~90000 U–Fs rods in 304LSS to 1–3 at.% >30000 U–Fs rods in 316SS to 8 at.% ~13000 U–Zr in 316SS 10 at.%	>48000 MOX rods in 316SS (Series I&II) to 80 GWd/MTHM	None applicable
Through qualification	U–Zr in 316SS, D9, HT9. 10 at.% in EBR-II and FFTF	MOX in HT9 to 15–20 at.% bu (CDE) MOX in 316SS (Series I&II) to 100 GWd/MTHM	None applicable
Demonstrated burnup capability and experiments	U–Pu–Zr in D9 and HT9 to 10–20 at.% in EBR-II and FFTF	4300 MOX rods in 316SS to 100 GWd/MTHM in FFTF; fab var's; CL melt >3000 MOX rods in 316SS to 5–17.5 at.% in EBR-II 2377 MOX rods in D9 to 100–129 GWd/MTHM; peak rods to 188 GWd/MTHM	18 EBR-II assemblies with 472 rods in 316SS cladding; 10 rods up to 20 at.% w/o breach, five of which experienced 15% TOP at 12 at.%; 13 He-bonded and 3 Na-bonded rods up to 16 at.% w/o breach 219 rods in FFTF, including 91 in D9 cladding, and 91 with particulate and pellet fuel
Safety and operability	6 RBCB tests U–Fs and U–Pu–Zr/U–Zr 6 TREAT tests U–Fs in 316SS (9rods) and U–Zr/U–Pu–Zr in D9/HT9 (6 rods)	18 RBCB tests; 30 breached rods four slow ramp tests 9 TREAT tests MOX in 316SS (14 rods) and HT9 (5 rods)	10 TREAT tests (10 rods); up to 3–6 times transient overpower margins to breach Loss-of-Na bond followed by irradiation to 3 at.% RBCB for 100 EFPD centerline melting test

Table 9
Suggested reference design parameters for mixed oxide (MOX) fuel, U–Pu–Zr fuel, and mixed carbide (MC) fuel based on US experience

Parameter	Mixed oxide (MOX)[a]	U–Pu–Zr[b]	Mixed carbide (MC)[c]
Nominal composition	$(U,Pu)O_2$	U–20Pu–10Zr	(U,Pu)C
Pu/(U + Pu) range	22–30%	17–28%	21–23%
Oxygen-to-metal ratio	1.95	n/a	
Fuel theoretical density	92%	100%[d]	80–82%
Fuel smeared density (% TD)	80–85%[e]	75%	78–79%
Plenum-to-fuel volume ratio	1.0	1.4	1.0
Fuel height	91 cm[f]	91 cm[f]	91 cm
Fuel outer diameter, as-fabricated	0.56 cm[g]	0.5 cm[g]	TBD[g]
Fuel inner diameter, as-fabricated	0.15 cm[g]	n/a	n/a
Fuel-cladding bond	He	Na	He
Cladding material	HT9[h] or 20% cw 316SS	HT9[h] or 20% cw 316SS	HT9[h] or 20% cw 316SS
Cladding outer diameter[f]	0.69 cm	0.69 cm	0.94 cm
Cladding inner diameter[f]	0.57 cm	0.57 cm	TBD[g]
Peak linear heat generation rate	44–46 kW/m	49–52 kW/m	66–80 kW/m
Peak inner-wall cladding temperature, nominal	620 °C	620 °C[i]	620 °C
Duct material	HT9[h] or 20% cw 316SS	HT9[h] or 20% cw 316SS	HT9[h] or 20% cw 316SS

[a] Based on core demonstration experiment fuel design [17].
[b] Based on EBR-II Mark-V driver fuel (which was the Mark-IV design adapted to U–Pu–Zr [13] and the MFF experiment series in FFTF [29,63]).
[c] Based on successful irradiation tests of MC fuel in EBR-II and the FFTF AC-3 test [79,80,82].
[d] Percent of apparent, as-measured theoretical density of 15.6–15.9 g/cm^3. Does not consider pores or hot tears in as-cast fuel.
[e] MOX fuel smeared density of 80% (of theoretical density) allowed burnup of 100 GWd/MTHM in 316SS rods and 238 GWd/MTHM in HT9 rods.
[f] Fuel height shown is that used for FFTF driver core and experiments, the experience for which was considered relevant for fuel columns as high as 135 cm [63].
[g] The fuel and cladding sizes shown are based on FFTF experience with MOX and metal fuels [17,63]. Other experience with U–Pu–Zr in EBR-II indicates that metal fuel rods with larger diameters will perform acceptably [93]. Different dimensions can therefore be used, but fuel smeared density and plenum-to-fuel volume ratio should be preserved. MC fuel has not been tested with HT9, so the desired cladding wall thickness and fuel diameter has not yet been considered.
[h] HT9 is proposed for reference design here, based on established database of materials properties and for fuel rod and assembly performance. Type 316SS (with 20% cold work) is suggested as a lower-cost alternative to HT9 if economics or schedule for a particular application favors lower burnup with less-expensive or more-available cladding. But note that dimensions shown for all but the MC fuel are those used for HT9 cladding, which would use a thicker wall than a Type 316SS cladding intended for lower burnup.
[i] Peak cladding temperature shown for metal fuel represents a value for steady-state irradiation in full-length rods. The overall limit of 650 °C [39] will likely be more limiting as transient analysis results are incorporated into core design requirements, which would reduce the cladding temperature allowed in steady-state operation.

fuel behavior modeling addressed key questions and ultimately demonstrated that two, and possibly three, fuel types will meet SFR safety and performance requirements. In particular, the metal and MOX fuel programs evolved from experiments designed to assess fuel response under varied conditions and fuel lifetime to larger programs for establishing reliability of fuel designs in specific applications and addressing safety issues raised by independent safety experts – i.e., issues associated with implementation rather than issues of feasibility. The questions that arise today are those associated with meeting the economic and mission requirements of SFR deployment, and internationally those are motivating extensions of the fuel technology R&D into new fuel compositions and designs to reduce fabrication costs, increase burnup, and enhance safety.

The MOX fuel database is well established in the US due to the experience gained with experiments in EBR-II and FFTF and with the operation of FFTF with a driver fuel design that closely resembles the design that would be implemented in a full-scale, MOX-fueled SFR. This maturity is also prevalent worldwide, as the result of efforts in Japan, France, Russia and the UK. The quality and safety discipline implemented for FFTF operation and experimentation were relevant to industrial implementation, so the US should have high confidence in the reliability and performance of MOX fuel at fuel utilization beyond 200 GWd/MTHM and fast neutron exposures up to 4×10^{23} n/cm^2. The

authors believe, leaving aside the issues associated with beyond-design basis events and hypothetical core disruption events, a licensing case for an industrial-scale demonstration reactor could be prepared with the existing MOX fuel database. A set of suggested design parameters for use of MOX fuel, similar to those used as reference for the FFTF Core Demonstration Experiment [17], is presented in Table 9.

Considered in total, the metal fuel database is substantial, with nearly 30 years of operating experience in EBR-II alone. The evolutionary development leading to U–Pu–Zr has been sufficient to identify and understand the mechanisms that control metal fuel lifetime and safety-related response, and the operating experience in EBR-II has been sufficient to establish the reliability of metal fuel such that fuel utilization of 200 GWd/MTHM can expected. U–Pu–Zr fuel technology will benefit from large-scale utilization similar to that of U–Fs and U–Zr fuel in EBR-II and MOX fuel in FFTF, and that will likely be necessary to build confidence prior to an industrial-scale implementation. However, the authors assert that a safety case for use of such fuel in a demonstration or test reactor can be prepared and defended from the existing database, with incorporation of uncertainty factors that can be reduced with further experience. A set of suggested design parameters for use of U–Pu–Zr fuel, derived from experience with fuel irradiated in EBR-II and FFTF, is presented in Table 9.

The experience base with mixed nitride fuels and mixed carbide fuels is limited, but carbide fuel technology in the US progressed sufficiently to demonstrate burnup potential of 200 GWd/MTHM and the absence of any limiting transient-overpower behavior. Mixed carbide fuel technology would benefit from additional development work to evaluate a broader range of irradiation conditions and design parameters. More importantly, future development of MC fuel should address behavior with HT9 (or other low-swelling, ferritic–martensitic alloys) and should seek to establish a sufficient database to establish reliability – such as with a large-scale use of MC fuel for a reactor core. The program and experience in India would necessarily be considered prior to any significant work in the US with carbide fuels. A set of suggested reference design parameters for mixed carbide fuel is presented in Table 9, reflecting experience with prior US irradiation testing [79,80,82].

In the competitive programs within the US through the 1970s, 1980s, and 1990s each of the R&D groups advocated the fuel system they worked on, concluding that each fuel system appears capable of meeting SFR fuel requirements. After this retrospective review, the present authors agree MOX, metal, and MC fuel types all appear capable of meeting SFR fuel requirements, with the reliability of MOX and metal fuel already well established. Selection of one fuel system over another will depend on circumstances particular to the application and on issues other than fuel performance, such as fabrication, recycle cost or, overall system safety performance, as has been suggested previously [92].

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