

5. Metallic Fuels

5.1. Introduction

This chapter describes fabrication technologies and experiences, irradiation behavior in steady-state and transient/accidental conditions, and advanced concepts of metallic fuels, largely depending on the reviews by Kittel et al [1], Walters et al [2], Hofman and Walters [3], Hofman et al [4], Ogata [5], OECD/NEA [6], and Aitkaliyeva [7].

A metallic fueled fast reactor core produces higher averaged neutron energy than more averaged neutron generation per fission (η -value) than an oxide or nitride fueled fast reactor core, because metallic fuels do not include oxygen nor nitrogen, which effectively moderate neutron speed. Furthermore, metallic fuels have higher density of fuel materials than any other fuel forms, enabling core design with high-performance, such as high breeding ratio and/or low fissile inventory. Owing to these characteristics of metallic fuels, uranium alloys were used as driver fuels in early fast reactors: EBR-I, Enrico-Fermi, DFR and EBR-II [2]. The burnups of these early metallic fuels were limited to lower than a few atom percent (at.%) because the significant fuel-cladding mechanical interaction (FCMI) occurred at early stage of irradiation due to rapid gas swelling of these uranium alloys. Thereafter, reduction of fuel smear density (cross-sectional areal percentage of fuel to cladding interior) down to 75% was found to be effective to enhance fission gas release, decrease further gas swelling, then suppress FCMI. This finding enabled metallic fuels to attain higher burnup. The design burnup limit of EBR-II Mk-II driver fuel was 8 at.% [2].

Extensive development of U-Zr and U-Pu-Zr metallic fuels were promoted in the Integral Fast Reactor (IFR) program (1984-1994), including irradiation tests in EBR-II and FFTF, overpower tests in TREAT, and post-irradiation tests [4]. **Figure 5.1** outlines a typical metallic fuel pin developed in the IFR program. Rods of the U-Zr or U-Pu-Zr fuel alloy are usually formed by injection casting, and both ends of the rods are sheared to 30-40 cm in length. The rod of U-Zr or U-Pu-Zr fuel alloy is called “fuel slug”. When a reactor core height is higher than about 40 cm, a few fuel slugs are stacked. When an axial blanket is placed, U-Zr slug(s) are allocated above and below U-Pu-Zr fuel slug(s). The gap between the fuel slug and cladding is filled with sodium, which is called “bond sodium”. Relatively large volume space, called “gas plenum”, is provided above the fuel stack to accommodate increase in the pressure of fission gas accumulated in the course of irradiation. Zr content in the fuel alloy is basically 10 wt.%. This is added to improve the irradiation instability of the orthorhombic uranium phase, increase the fuel alloy solidus temperature, and improve the compatibility with the cladding material. As a result of the IFR program, high performance of metallic fuel such as the capability to attain >19 at.% peak burnup has been demonstrated [4].

Spent metallic fuels can be recycled by using the dry process. The application of the dry process to metallic fuels brings about another advantage of utilizing metallic fuels [5]. The

dry process has many merits, such as 1) suitability for remote automatic operation due to its simplicity, 2) cost effectiveness even in a small-scale facility due to its batch process, 3) no need to be concerned with radiolysis of organic agents, 4) resistance against proliferation due to low decontamination nature of the process, 5) easiness of minor actinides (MAs) recycle. In the 1960s, the EBR-II driver fuel pins at that time had been recycled in the hot-cell facility, called Fuel Cycle Facility (FCF), by using a dry process consisting of melt-refining and injection casting. This fuel cycle campaign demonstrated the feasibility of metallic fuel cycle [36].

Concepts of fuel assembly and fast reactor plant system for metallic fuels are not different from those for oxide fuels. However, higher averaged neutron energy in a metallic fueled core results in 30-40% higher neutron fluence of the fuel cladding than that in an oxide fueled core, so that the irradiation swelling-resistant cladding materials, such as ferritic/martensitic steel, are favorable for metallic fuels [5]. While the melting (solidus) temperature of U-Pu-Zr fuel alloy is lower than that of oxide fuel form, metallic fuels have a larger margin to fuel-centerline melting than oxide fuels due to higher thermal conductivity of the metallic fuel alloy [5].

After termination of the IFR program, research and development of metallic fuels have been continued in the United States, Japan, Republic of Korea and India. Some of these activities are listed below.

- The US: post irradiation examination of the MA-bearing metallic fuel pins irradiated in ATR and Phenix [8-12], measurements and compilation of MA-bearing U-Pu-Zr alloy properties [13,14], development of irradiation behavior analysis code, BISON [15,16], improvement in metallic fuels such as addition of indium and/or palladium to stabilize lanthanide fission products which are the cause of the fuel-cladding chemical interaction (FCCI) [17-19], etc.
- Japan: development of irradiation behavior analysis code, ALFUS [20], engineering-scale injection casting of U-Zr alloys [21], fabrication of U-Pu-Zr fuel pins to be irradiated in JOYO [22,23], post irradiation examination of the MA-bearing metallic fuel pins irradiated in Phenix [24-29].
- Republic of Korea: development of irradiation behavior analysis code, MACSIS [30], fabrication of U-Zr particles [31], development and design of metallic driver fuel for PGSFR [32], irradiation tests of U-Zr-Ce fuel pins in HANARO.
- India: measurements of thermochemical properties of U alloys [33-35].

5.2. Fuel Fabrication Technologies and Experiences

The uranium alloy fuels used for the early fast reactors were fabricated through extrusion, centrifugal casting and injection casting processes [1]. The most experienced process among them is injection casting, which was used for the fuel cycle demonstration at the EBR-II Fuel Cycle Facility [36] in the 1960s and about 34,500 fuel pins were remotely fabricated. In the IFR program, more than 13000 fuel pins (~2000 test pins plus >11,000 driver fuel pins)

of the U-10 wt.% Zr binary alloy fuel and about 600 test pins of the U-Pu-10 wt.% Zr ternary fuel were fabricated.

The injection casting process is outlined in [Figure 5.2](#) [5,21]. The starting materials are U and/or U-Pu ingots, Zr metal, heel and scraps from preceding casting batch(s). Here, “heel” is residual fuel alloy in the crucible after casting, and “scraps” are shards from both ends of cast alloy rods and rejected products. The weight-adjusted materials are charged in the graphite crucible, inner surface of which is coated with yttria paint. The chamber is closed, and its interior is purged with highly purified Ar gas. The materials are induction-heated up to sufficiently higher temperature (approximately 1830 K for the case of U-10wt.%Zr alloy) than the liquidus temperature of the alloy. After the materials are completely melted and homogenized, the chamber is evacuated, then a bundle of quartz molds is immersed into the melt. The upper ends of the molds are closed, and their inside surfaces are coated with zirconia. After immersion of the molds in the melt, by quickly refilling the chamber with Ar gas, the melt is injected into the molds and solidified. The mold bundle is raised before the melt in the crucible is solidified. After being cooled down, the bundle is taken out from the chamber, then cast alloy rods are retrieved by breaking the molds. Both ends of the rods are sheared to a predetermined length, and the fuel slugs are obtained. The heel and scraps are recovered and recycled for subsequent batches.

In the engineering-scale U-Zr injection casting tests conducted by the Central Research Institute of Electric Power Industry (CRIEPI) [21], more than 500 U-10 at.% Zr fuel slugs were fabricated, and statistic data of diameter and weight of the products were obtained. Through the tests, efforts were made to increase injection fraction (weight ratio of injected melt to total metal charge) for improvement in the production yield of the injection casting process. Rough sketch of material flow in the injection casting process is shown in [Figure 5.3](#) based on the result of the U-Zr injection casting tests. About 70% of the starting materials (100%) charged in the crucible is injected into the molds, and the shards sheared from both ends of the cast rod amount to 14.4%. Assuming 10% rejection at the inspection, net production yield is 50% of the initial charge.

[Figure 5.4](#) illustrates the fabrication process of a metallic fuel pin. This process was applied to the fabrication of the metallic fuel pins for the irradiation test in JOYO. Extruded sodium rod then fuel slug(s) are inserted in a cladding tube with a lower-end plug welded. An upper-end plug is welded after the interior of the cladding tube is purged with He gas. The cladding tube containing sodium and fuel slug(s) is put into a furnace, keeping it at upright position. The cladding tube is heated and kept at approximately 800 K, then vibrated vertically. During this “sodium bonding” step, the sodium rod is melted, the fuel slug(s) is settled down, and the gap between the fuel slug(s) and the cladding is filled with sodium. After the sodium bonding and cooling, the fuel slug(s) position and the bond sodium level are checked by X-ray radiography. Fabricated metallic fuel pins are transferred to the fuel assembly process, which is the same as that for oxide fuels.

Since the 1990s, growing interests in MA transmutation in fast reactors or accelerator-driven systems promoted the development of MA-bearing metallic fuel. In the case of injection

casting of MA-bearing U-Pu-Zr alloy, vaporization of Am, which has a relatively higher vapor pressure, was concerned during the evacuation step just before injection of the melt. Another concern was the broken molds that would be treated as radioactive waste. These two concerns led to development of new metallic fuel casting methods. Idaho National Laboratory (INL) has developed a gravity casting method [37,38], where molten fuel alloy is poured into reusable, coated graphite molds by gravity without an evacuation step. Korea Atomic Energy Research Institute (KAERI) has successfully fabricated U-10wt.%Zr alloy rods by continuous casting [39] and uranium alloy particles by centrifugal atomizing [31,39]. These two methods need no evacuation step and no mold.

5.3. Fuel Performance

5.3.1. Steady-state irradiation behavior [5]

After neutron irradiation, the cross sections of metallic fuels show two or three ring structure and exhibit characteristic behavior as illustrated in [Figure 5.5](#). Steady-state irradiation behavior of U-Pu-Zr metallic fuel pins with approximately 75% fuel smear density is described below.

When the center temperature of a cross section of a U-Pu-Zr fuel slug is higher than about 923 K, Zr radially migrates to the central and peripheral regions from the intermediate ring region, the temperature of which ranges from about 873 K to 923 K, depending on the fuel alloy composition. U migrates in the opposite directions of Zr migration. Pu does not migrate remarkably. The Zr in the intermediate region is depleted down to a several atom percent. This redistribution of fuel alloy constituents is considered to occur by thermo-diffusion, which is caused due to the chemical potential gradients of the constituents along the radial temperature gradient in the fuel slug. The central region ($>$ about 923 K) includes relatively larger, spherical fission gas bubbles. The peripheral region ($<$ about 873 K) contains highly distorted cavities, which are generated presumably due to tearing at the grain boundaries caused by anisotropic irradiation growth of the grains and/or phases of lower temperature uranium alloys in addition to agglomeration of fission gas atoms. In the intermediate region, which appears to be dense, many micron-scale fine pores are found. The fuel alloy cross section exhibits three-ring structure as a result of the characteristic features of gas bubbles and pores in respective regions described above. When the fuel center temperature is lower than about 923 K, the Zr-depleted region forms at the fuel central part, accordingly the fuel cross section shows two-ring structure, as shown in [Figure 5.5](#).

In the early stage of irradiation, a fuel slug swells out due to gas swelling and cracking. At a burnup level of approximately 1 at.% or less, the swollen fuel slug starts contact with the cladding, and the swollen fuel slug volume amounts to about 140 % of initial fuel slug volume in the case of 75% fuel smear density. At this stage, gas bubble population in the swollen slug becomes so high that the coalescence among gas bubbles and cracks occurs, leading to open pore formation. Because fission gas escapes through open pore, gas swelling rate is suppressed since then. Fraction of released fission gas to generated fission gas

increases with burnup, and approaches to an asymptotic value of 70-80% at approximately 10 at.% or higher burnup.

Onset of fission gas release before or at the time of the fuel slug-cladding contact leads to suppression of further fuel slug gas swelling, then to moderation of the fuel-cladding mechanical interaction (FCMI). Collapsing of open pore and crack by compressive stress also contribute to moderation of FCMI as long as they exist. The cladding diametral strain measured after irradiation indicate that significant FCMI does not occur until about 10 at.% burnup. After this burnup level, FCMI gradually grows due to inexorable solid fission product accumulation in the fuel slug (solid FP swelling). The upper bound of solid FP swelling rate is approximately 1.5 vol.%/at.% when all the fission products except gaseous element are assumed to contribute to solid FP swelling.

Although the gas swelling and solid FP swelling of the fuel slug is isotropic, the fuel slug deformation as a whole is anisotropic. The radial deformation of the fuel slug is limited at about 16%, as further fuel slug deformation is restrained by the cladding. The axial deformation, however, is about 8% for the case of U-10wt.%Zr fuel pins with 75% fuel smear density and less than 8% for the case of U-Pu-10wt.%Zr fuel pins with 75% fuel smear density depending on the Pu content. This anisotropy of the fuel slug deformation can be attributed to the stress state of the whole fuel slug, radial cracking, anisotropic irradiation growth of lower temperature grains or phases.

The fuel slug become porous in the course of irradiation because of gas bubble formation and cracking, then its effective thermal conductivity decreases. When the fuel slug swells up to 140% of the initial fuel slug volume (~28% porosity), the effective thermal conductivity decreases down to 40-50% of that of unirradiated fuel slug. After the open pore formation, the bond sodium infiltrates into the open pore and the effective thermal conductivity recovers.

Part of lanthanide fission products migrate to the peripheral region of the fuel slug and react with the cladding inner surface. The interaction of lanthanide fission products and fuel constituents with the cladding is called Fuel-Cladding Chemical Interaction (FCCI). FCCI during steady-state irradiation is dominated by the reaction of lanthanide fission products with cladding. Possible mechanisms of the lanthanide migration to the fuel slug peripheral region may be thermo-diffusion and/or transportation through open pore, but quantitative explanation of the lanthanide migration has not yet been made.

5.3.2. Transient behavior [5]

When the cladding temperature exceeds a certain threshold value in the abnormal transient or accidental conditions, another mode of FCCI emerges. In this mode, fuel constituents cause eutectic reaction with Fe-base cladding, and the inner surface of the cladding is eroded at a significantly higher rate. Fe, a major constituent of the cladding material, diffuses into the fuel slug and forms a liquid phase in the peripheral region. The threshold value of onset of liquid phase formation at the fuel-cladding interface is independent of Zr content in the fuel slug. The liquid phase does not appear at lower

temperatures than 923 K in the case of Pu to (U+Pu) ratio lower than 0.25, as indicated in [Figure 5.6](#).

When the central part of the fuel slug melts in the case of an accidental overpower condition, the molten part of the fuel slug expands due to decrease in the fuel surface tension and gas bubble coarsening through gas bubble coalescence. Since this fuel centerline melting starts near the top of the slug unlike oxide fuel because of higher thermal conductivity of metallic fuel, the molten part is extruded into the gas plenum region without causing cladding breach, as observed in the transient overpower tests in TREAT. This extrusion phenomenon implies that metallic fuel centerline melting does not cause excessive FCMI.

5.3.3. Fuel pin failure

During a steady-state irradiation condition, metallic fuel pin failure is mainly caused by superposition of plenum pressure rise by fission gas accumulation and FCCI-induced cladding thinning with increasing burnup. In the transient or accidental conditions, metallic fuel pin failure is mainly caused by superposition of plenum pressure rise with increasing coolant temperature, decrease in the cladding strength with increasing cladding temperature and the liquid phase attack to the cladding (another mode of FCCI). In both cases, the cladding failure is considered to occur at the position of the top of the fuel stack, where the cladding temperature is highest, its strength is lowest, and FCCI is significant. When the transient overpower condition terminates within the order of 10 seconds and the peak cladding temperature is lower than 1273 K, the rate of cladding thinning due to liquid phase attack is not significant, as observed in the transient overpower tests in TREAT. On the other hand, the liquid phase attack is not negligible when the cladding temperature is maintained at about 1100 K or higher for 1 hour in a loss-of-coolant accident. In both cases, when the fuel pin burnup is high ($>$ about 10 at.%), the cladding will burst by plenum pressure rise before liquid phase attacks the cladding significantly.

The behavior of a breached metallic fuel pin will be moderate, because U-Zr or U-Pu-Zr fuel alloy is compatible with sodium coolant.

5.4. Advanced Metallic Fuels

5.4.1. MA-bearing metallic fuel

Toward the development of MA transmutation systems, MA-bearing metallic fuel irradiation tests have been performed, such as the X501 test fuel assembly irradiated at EBR-II [40-42], the AFC-series test fuel pins irradiated at ATR [43-47], the FUTURIX-FTA test fuel pins irradiated at Phenix [11,48,49] and the METAPHIX test fuel pins irradiated at Phenix [24-29,50-54]. These MA-bearing metallic fuel irradiation tests are listed in [Table 5.1](#) [5]. The AFC-series and FUTURIX-FTA fuel pins with higher than 25 wt.% TRU (Pu+MA) contain higher than 10 at.% Zr. Post-irradiation examination of X501 and METAPHIX test fuel pins suggested that the addition of less than 5 wt.% MA to U-Pu-10wt.%Zr does not significantly influence the irradiation behavior of U-Pu-10wt.%Zr metallic fuels [6]. The

results of the AFC-1B irradiation test indicated that the dependencies of fission gas release and swelling volume on cumulative “fissions per volume” for the non-fertile fuel pins are similar to those for U-Pu-10wt.%Zr fuel pins [43].

5.4.2. Metallic fuels with additives

The addition of In and/or Pd to U-Pu-Zr metallic fuels are considered to suppress FCCI by stabilizing lanthanide fission products, which are expected to form intermetallic compounds with In or Pd [17-19]. Irradiation tests and out-of-pile diffusion couple tests have been continued in order to confirm the effectiveness of this idea.

5.4.3. Helium-bonded particulate metallic fuel

He-bonded particulate metallic fuel [55] has been considered as an advanced metallic fuel concept. Bimodal mixture of fuel alloy particles is contained in the cladding tube with He gas, as illustrated in [Figure 5.7](#). Typically, larger fuel particles with ~1 mm diameter is mixed with smaller fuel particles with ~0.1 mm in the ratio of approximately 7:3. Small modular fast reactors, ARC-100 and MCR-200, assume to use this type of fuel [56,57]. The fuel particles can be fabricated by, for example, centrifugal atomizing, which have been experienced successfully by KAERI as mentioned above. The He-bonded particulate metallic fuel has following advantages. 1) Because of no use of bond sodium, the gas plenum can be located at the bottom of the fuel pin, where the temperature is relatively low, so that the gas plenum length then total fuel pin length can be shortened. 2) The atomizing process, which can be used for fuel alloy particle fabrication, needs no mold, has higher production yield than the injection casting process, and may suppress vaporization of Am. 3) The sodium bonding step can be eliminated from fuel fabrication process. On the other hand, disadvantage of He-bonded particulate metallic fuel is the necessity of preconditioning to improve the thermal conductivity of the fuel alloy particle bed by sintering among the particles, because the thermal conductivity of the particle bed before irradiation is as low as about 18% of the thermal conductivity of the solid fuel alloy. One of the possible preconditioning methods is to increase the reactor power linearly from zero to a nominal power at the reactor start-up. An evaluation result indicated that this method would complete preconditioning within one or a few days [56].

6. Thermophysical Properties

6.3. Metallic Fuel

Some measured densities were reported for U-Zr alloys [58] and U-Pu-Zr alloys [59-61], as shown in [Figure 5.8](#) [5]. The U-Zr and U-Pu-Zr densities estimated from the molar volumes of U, Pu, and Zr are also shown in the figure. The estimated values appear to be an upper bound of measured values. Difference between estimated and measured values may be attributed to the impurities, such as C, N and O.

Measured data of solidus and liquidus temperatures were reported for U-Zr alloys [35,63,64] and U-Pu-Zr alloys [61-63]. The following correlations [65] were proposed for the solidus, T_{sol} , and liquidus, T_{liq} , temperatures based on the thermochemical assessment of U-Pu-Zr system [66].

$$\begin{aligned}
 T_{sol} &= A_0 + A_1 C_{Zr} + A_2 C_{Zr}^2 + A_3 C_{Zr}^3 \\
 A_0 &= 1408 - 1187 C_{Pu} + 967 C_{Pu}^2 \\
 A_1 &= 572 - 732 C_{Pu} + 4960 C_{Pu}^2 \\
 A_2 &= 740 + 3305 C_{Pu} - 29182 C_{Pu}^2 \\
 A_3 &= -624 - 3139 C_{Pu} + 36120 C_{Pu}^2
 \end{aligned} \tag{5.1}$$

$$\begin{aligned}
 T_{liq} &= B_0 + B_1 C_{Zr} + B_2 C_{Zr}^2 + B_3 C_{Zr}^3 \\
 B_0 &= 1408 - 749 C_{Pu} + 93 C_{Pu}^2 \\
 B_1 &= 1313 + 3869 C_{Pu} + 5072 C_{Pu}^2 \\
 B_2 &= -1052 - 6637 C_{Pu} - 44769 C_{Pu}^2 \\
 B_3 &= 521 + 1683 C_{Pu} + 66380 C_{Pu}^2
 \end{aligned} \tag{5.2}$$

Here C_{Zr} , C_{Pu} and C_U are atomic fractions of Zr, Pu and U, respectively. The correlations (5.1) and (5.2) are applicable for $C_{Pu}/C_U < 1$ and $C_{Zr} < 0.8$. In the case of U-Zr binary alloy, $C_{Pu} = 0$. The above correlations are plotted in [Figure 5.9](#). Since there are large deviations from the reported data: <50 K for the solidus, <120 K for the liquidus [5], accurate measurements of the solidus and liquidus temperatures and thermochemical reassessment of U-Pu-Zr system are desirable in the future.

Experimental values of thermal conductivities of U-Zr and U-Pu-Zr alloys were reported in Refs. [34,67-70] and Ref. [71], respectively. An experimental correlation for thermal conductivities of the U-17.2 at.% Pu-22.5 at.% Zr-1.7at.% Am alloy was proposed in Ref. [72]. The correlations for U-Zr and U-Pu-Zr thermal conductivities were proposed by Billone et al [73], Ogata [5, 65] and Kim et al [74]. The correlation by Billone et al [73] does not reflect experimental data reported after 1988 [34,69,70]. The other correlations were expressed as follows.

Ogata [5, 65]:

$$k_0 = 16.309 + 0.02713 \cdot T + 46.279 \cdot C_{Zr} + 22.985 \cdot C_{Zr}^2 - 53.545 \cdot C_{Pu} \tag{5.3}$$

where C_{Zr} and C_{Pu} are atomic fractions of Zr and Pu, respectively, and $T < 1173$, $C_{Zr} < 0.72$, $C_{Pu} < 0.16$.

Kim et al [74]:

$$\begin{aligned}
k_{U-Pu-Zr} &= (1 - \sqrt{1 - x_{Pu}})k_{Pu} + \sqrt{1 - x_{Pu}}\{(1 - x_{Pu})k_{U-Zr} + x_{Pu}k_{c,Pu}\} \\
k_{U-Zr} &= (1 - \sqrt{1 - x_{Zr}})k_{Zr} + \sqrt{1 - x_{Zr}}\{(1 - x_{Zr})k_U + x_{Zr}k_{c,Zr}\} \\
k_{c,Pu} &= -61.915 - 53.456x_{Pu} + 362.453x_{Pu}^2 + 0.1319T - 7.851 \times 10^{-5}T^2 \\
&\quad - 0.0115x_{Pu}T \\
k_{c,Zr} &= -102.0 + 200.1x_{Zr} - 109.2x_{Zr}^2 + 9.435 \times 10^{-3}T + 3.459 \times 10^{-5}T^2 \\
&\quad - 0.02093x_{Zr}T \\
x_{Pu} &= w_{Pu} \\
x_{Zr} &= w_{Zr}/(1 - w_{Pu}) \\
k_U &= 21.73 + 1.591 \times 10^{-2}T + 5.907 \times 10^{-6}T^2 \\
k_{Zr} &= 8.853 + 7.082 \times 10^{-3}T + 2.533 \times 10^{-6}T^2 + 2.992 \times 10^3T^{-1} \\
k_{Pu} &= -4.987 + 4.408 \times 10^{-2}T - 1.566 \times 10^{-5}T^2
\end{aligned} \tag{5.4}$$

where $k_{U-Pu-Zr}$ is the thermal conductivity of U-Pu-Zr alloy (W/m/K), T is temperature (K), w_{Zr} and w_{Pu} are weight fractions of Zr and Pu in U-Pu-Zr alloy respectively; k_U , k_{Zr} and k_{Pu} are thermal conductivities of U, Zr, and Pu, respectively; k_{U-Zr} is an empirical expression for the thermal conductivity of U-Zr; and $k_{c,Pu}$ and $k_{c,Zr}$ are empirical factors to account for the influence of alloying behavior of Pu in U-Pu-Zr alloy and Zr Pu in U-Zr alloy, respectively. For the U-Zr binary alloy, $w_{Pu} = 0$. The correlation by Kim et al gives a relatively better agreement with the experimental data than that by Ogata, while the difference between two correlations is small.

The measured data of heat capacities of U-Zr and U-Pu-Zr alloys are reported in Refs. [34,70,75] and [71], respectively. Ref. [72] reports the measured data of the U-17.2 at.% Pu-22.5 at.% Zr-1.7at.% Am alloy. These data are not consistent with each other, as can be seen in **Figure 5.10**, so that the correlation for U-Zr and U-Pu-Zr heat capacities with the composition cannot be developed. Until more accurate data are obtained and relevant thermodynamic assessment is performed in the future, the heat capacities of U-Zr and U-Pu-Zr alloys are recommended to be approximated from the heat capacities of respective the

elements by the Neumann-Kopp law. While the measured heat capacity data for U-Pu-Zr and U-Pu-Zr-Am-Np alloys have recently been compiled in Ref. [77], these data are within the data band in **Figure 5.10**.

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Table 5.1 Irradiation tests of MA-bearing metal fuel [5]

	X501	METAPHIX	AFC-1B	AFC-1F, -1H	AFC-2A, -2B	FUTURIX-FTA(Metal)
Reactor for irradiation	EBR-II	Phenix	ATR	ATR	ATR	Phenix
Fuel alloy composition (wt.%)	U-20.2Pu-9.10Zr	U-19Pu-10Zr	Pu-12Am-40Zr	U-29Pu-4Am-2Np-30Zr	U-20Pu-3Am-2Np-15Zr	U-29Pu-4Am-2Np-30Zr
	-1.2Am-1.3Np	U-19Pu-10Zr-2MA-2RE	Pu-10Am-10Np-40Zr	U-34Pu-4Am-2Np-20Zr	U-20Pu-3Am-2Np-1.0RE-15Zr	Pu-12Am-40Zr
		U-19Pu-10Zr-5MA-5RE	Pu-60Zr	U-25Pu-3Am-2Np-40Zr	U-20Pu-3Am-2Np-1.5RE-15Zr	
		U-19Pu-10Zr-5MA	Pu-40Zr	U-28Pu-7Am-30Zr	U-30Pu-5Am-3Np-1.5RE-20Zr	
		(MA: Np, Am, Cm) (RE: Y, Ce, Nd, Gd)			U-30Pu-5Am-3Np-1.0RE-20Zr U-30Pu-5Am-3Np-20Zr (RE: La, Pr, Ce, Nd)	
Fuel slug O.D. (mm)	4.27	4.9	4.01	4.01	4.27	
Fuel slug length (mm)	343	485 (100 of MA section)	38.1 (25.4 for Pu-60Zr)	38.1	38.1	
Cladding Material	HT9	15-15Ti	HT9	HT9	HT9	
Cladding O.D. (mm)	5.84	6.55	5.84	5.84	5.84	
Peak liner power rate (W/cm)	450	350	330 (design limit)	330 (design limit)	350	272 (low-fertile) 320 (non-fertile)
Peak cladding temp. (K)	813	845	823 (design limit)	823 (design limit)	823 (design limit)	
Peak burnup (at.%)	7.6	2.5, 7, 11	4~8	4~8 (1F), 35~40 (1H)	>10 (2A), >25 (2B)	7.0 (low-fertile) 11.4 (non-fertile)

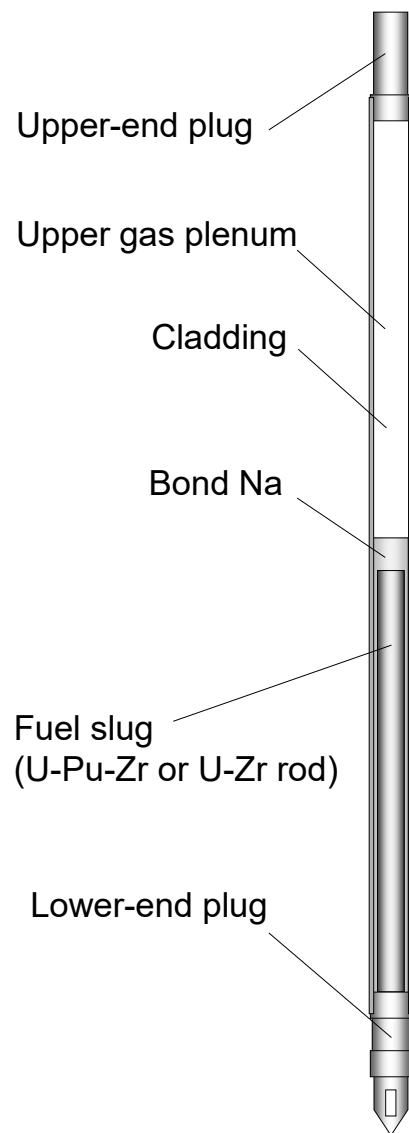


Figure 5.1 Metallic fuel pin concept

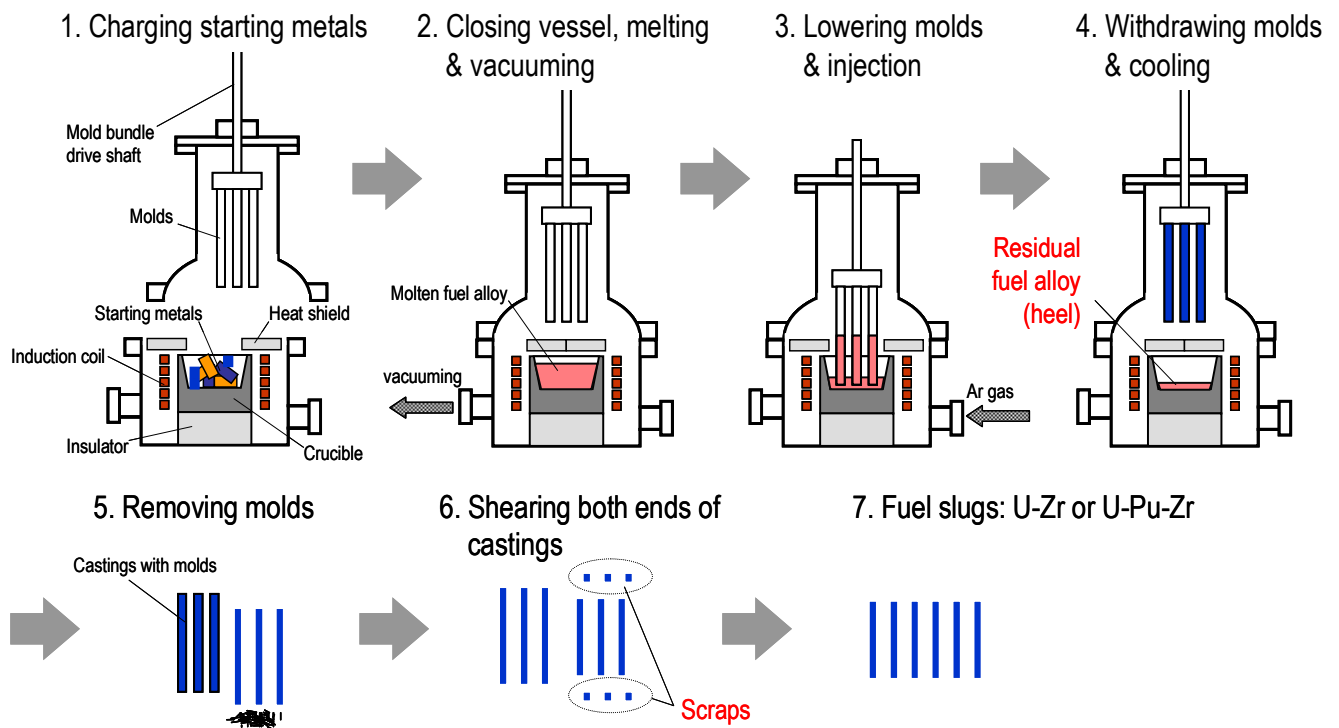


Figure 5.2 Outline of injection casting process [5,21]

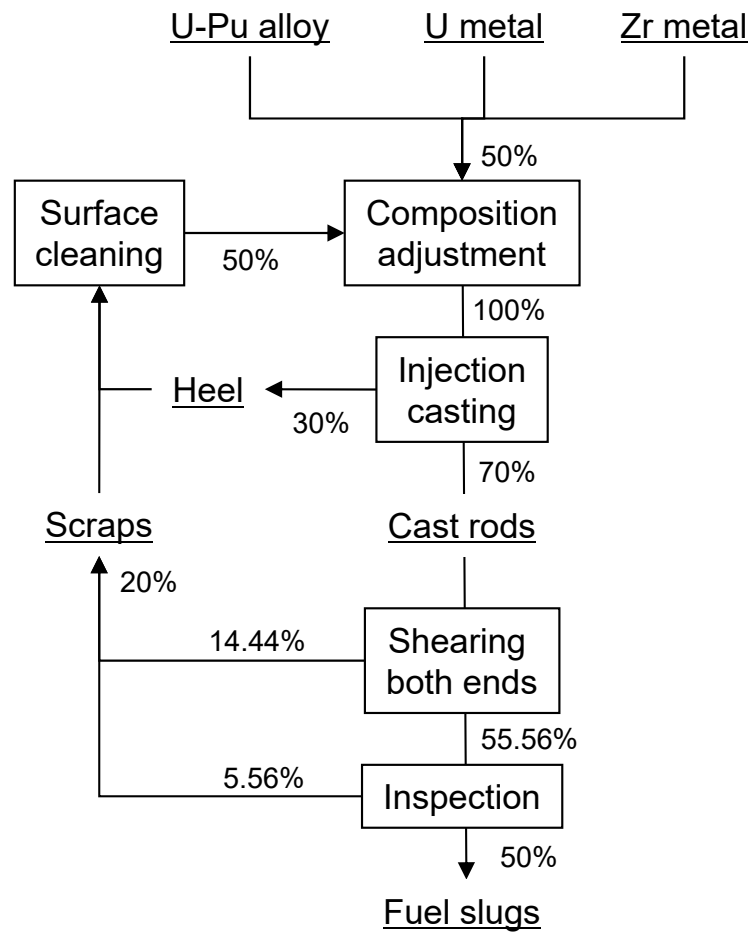


Figure 5.3 Rough sketch of material flow in an injection casting process

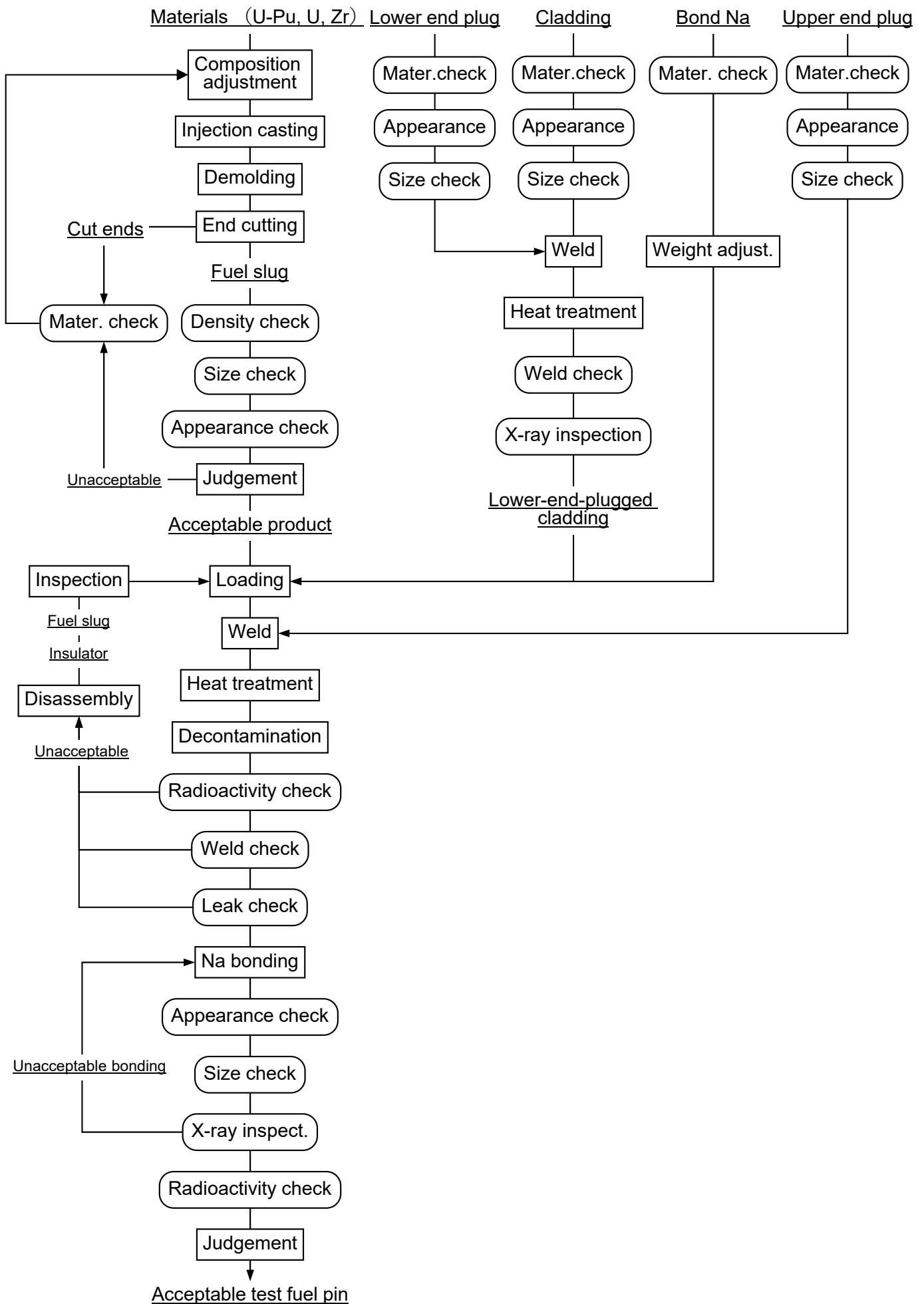


Figure 5.4 Metal fuel pin assembling process

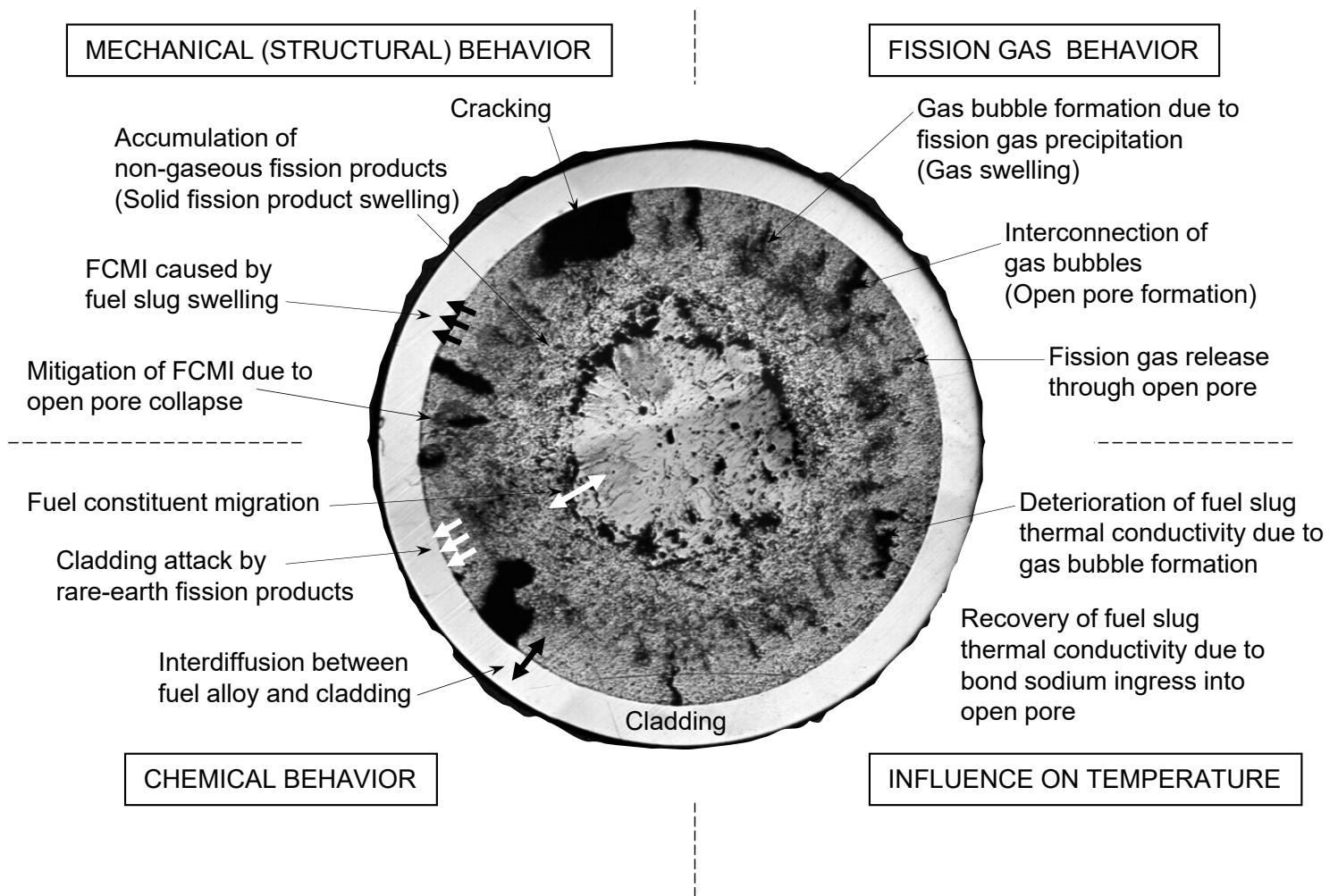


Figure 5.5 Outline of irradiation behavior of metal fuel [5]

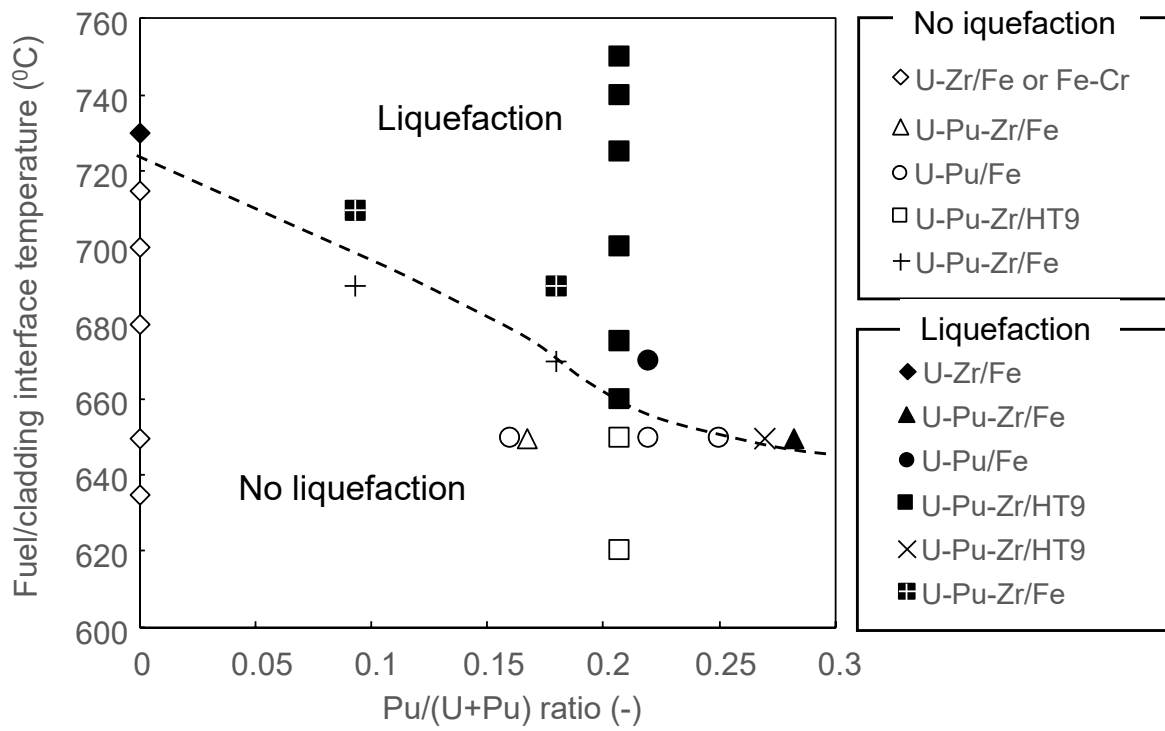


Figure 5.6 Liquefaction map with respect to fuel/cladding interface temperature and Pu/(U+Pu) ratio [5]

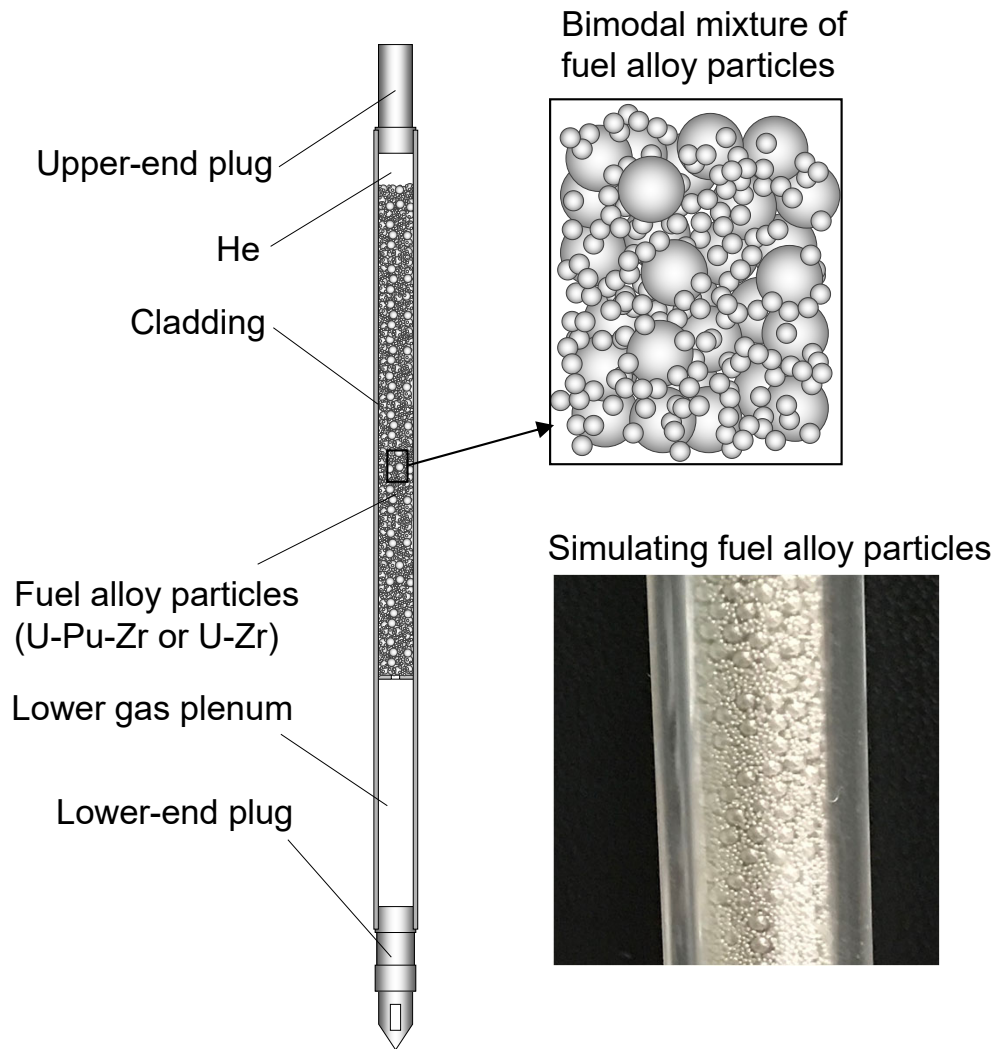


Figure 5.7 He-bonded particulate metallic fuel pin concept [56]

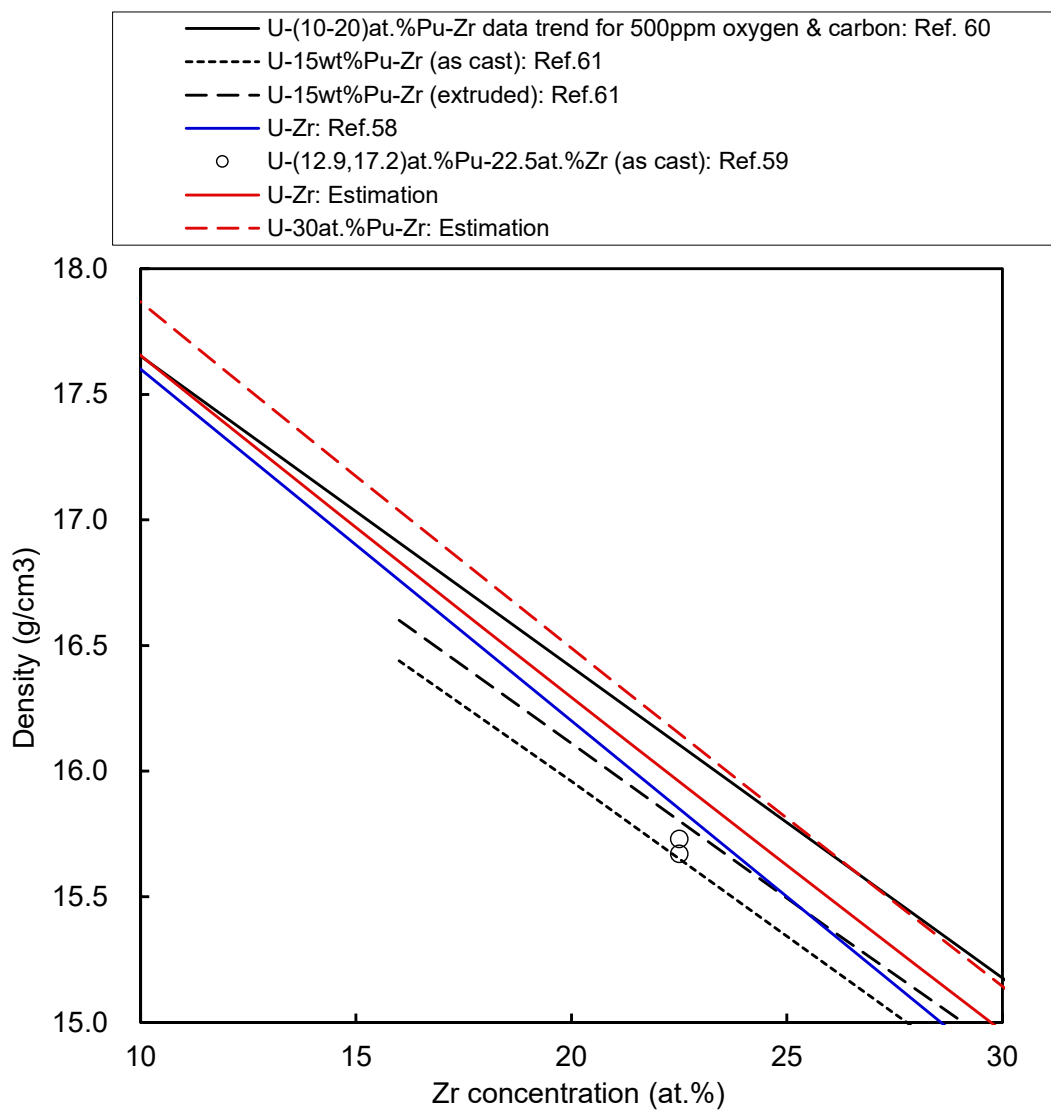
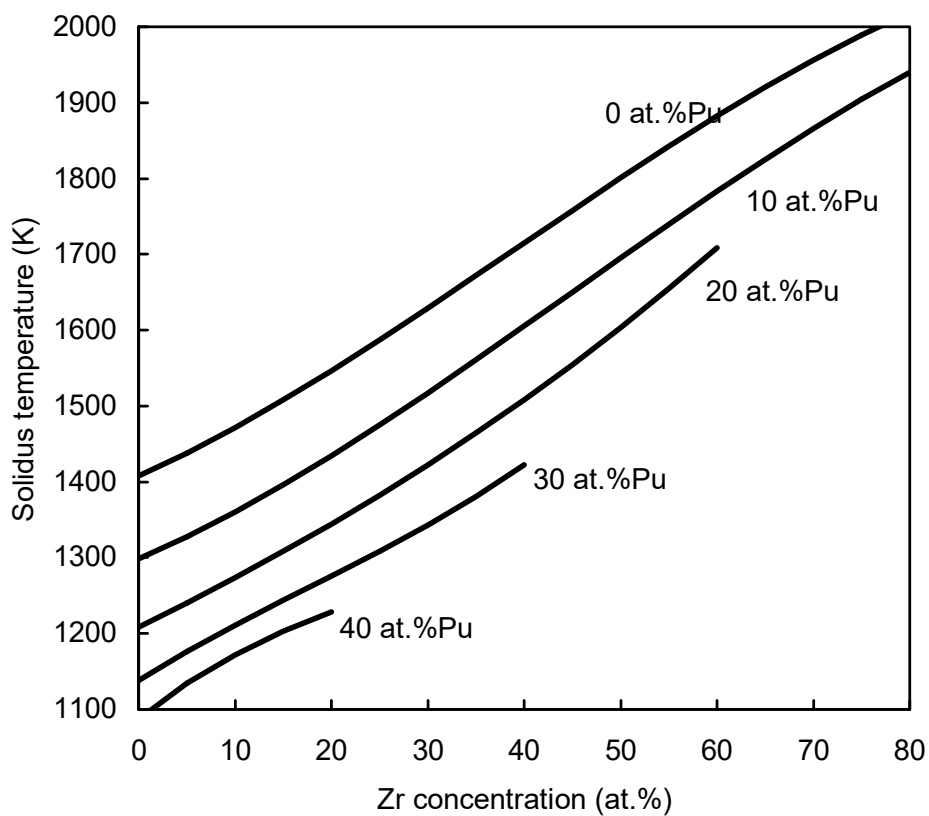
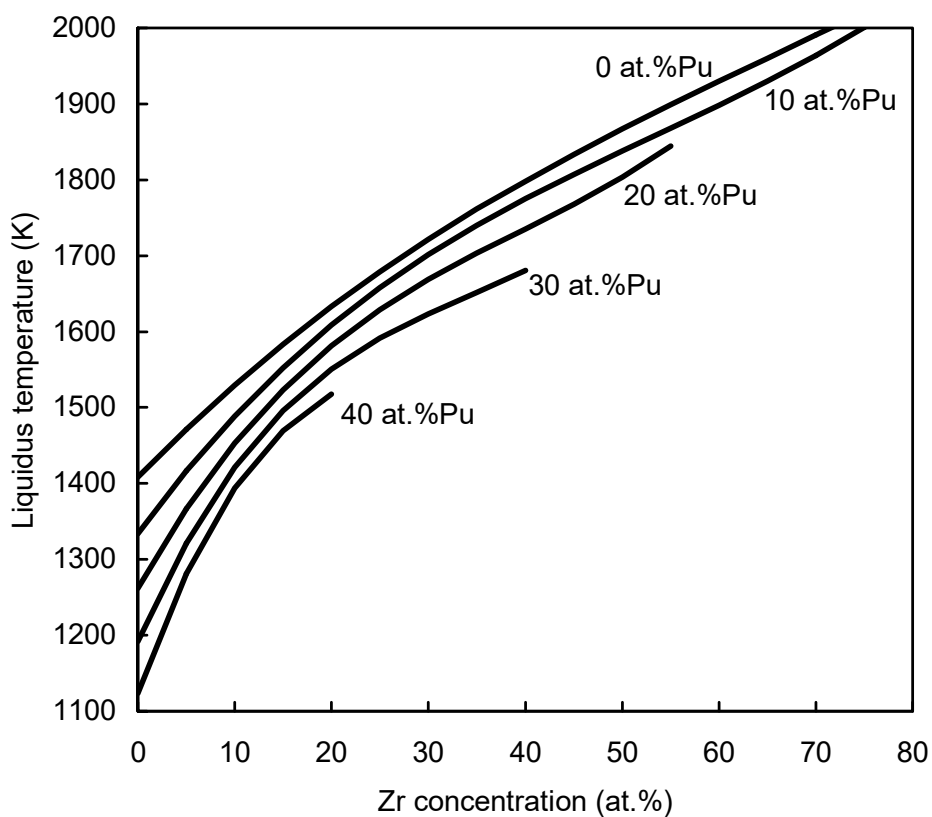


Figure 5.8 Measured and estimated densities of U-Zr and U-Pu-Zr alloys



(a) Solidus temperature



(a) Liquidus temperature

Figure 5.9 Evaluated solidus and liquidus temperatures of U-Zr and U-Pu-Zr alloys [5].

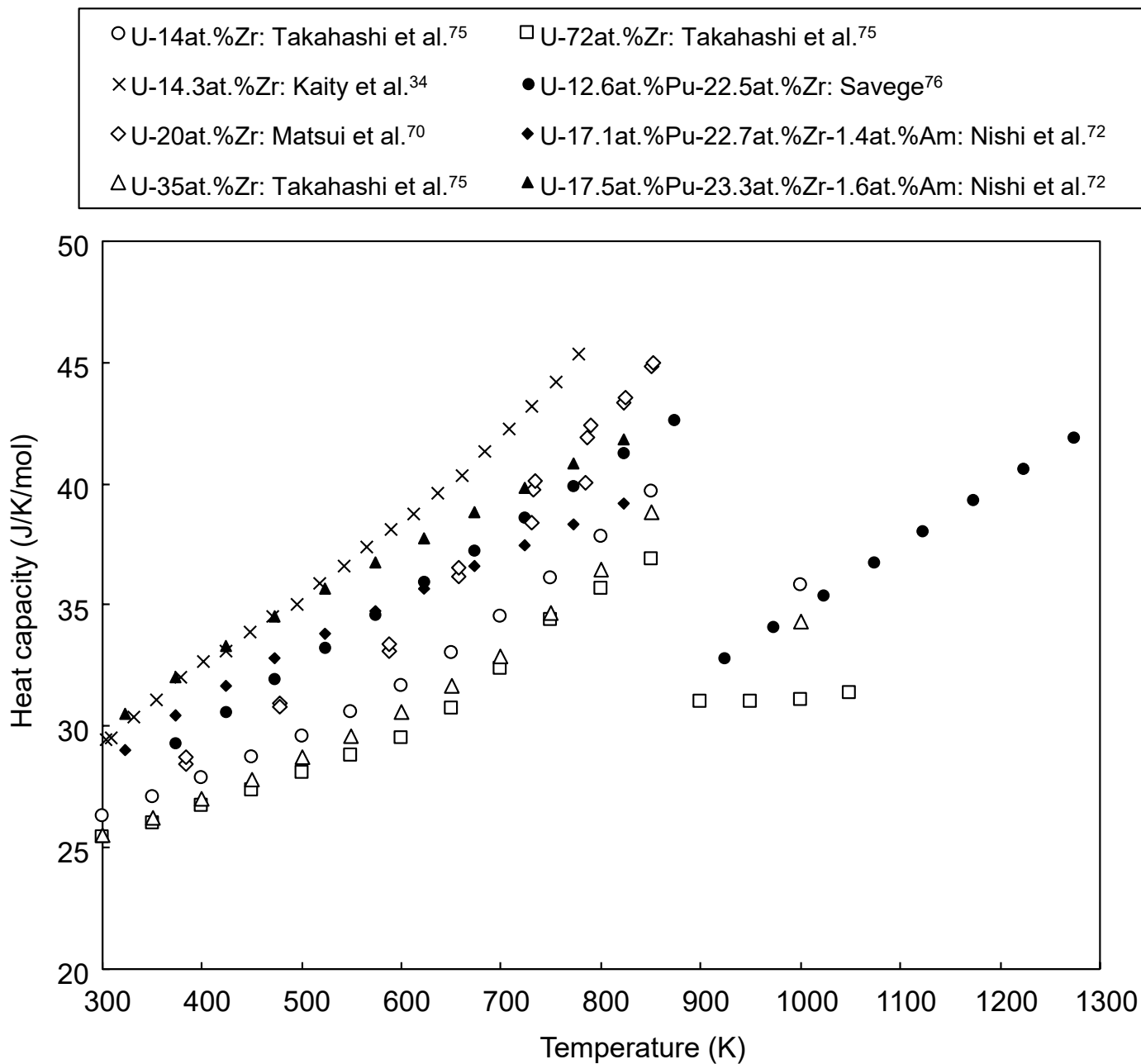


Figure 5.10 Measured heat capacity of U-Zr and U-Pu-Zr alloys [5].