

# EVOLUTION OF DISPERSED FUEL USED FOR RESEARCH AND TEST REACTORS

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**ABSTRACT:** Currently, 222 research reactors are operating in 53 countries in Brazil, four units. Nuclear fuel type used in research reactors depends on the thermal power level, classified by power classes, such as critical and subcritical units operating below 1Kw, working from 1 kW to 1 MW, and over 1 Mw, being 28 facilities operating on a medium power spectrum from 1 MW up to 10Mw. Alternatively, they divided the research reactor into low-enriched uranium below 20% of U-235 and highly-enriched uranium over this limit. In the 1950s, the superheat section of the Fifth Boiling Reactor Experiment (BORAX-V) used enriched uranium (93 wt.% U-235) as uranium dioxide in a matrix of stainless steel. In sequence, it divided a fast dispersion fuel evolution into three ages. The 1950s predominated U–Al alloy. In the 1960s, the  $U_3O_8$ –Al and  $UAl_x$ –Al followed the 1980s, focusing on high uranium density up to 4.8 gU/cm<sup>3</sup> planned to  $U_3Si_2$ –Al and U10Mo/Al.

Today, global research focuses on U-10Mo dispersion and monolithic forms. Many thermal physical features of dispersed fuels can help define the application field, mainly focused on neutron activation analysis and radiopharmaceutical production.

**KEYWORDS:** Research reactors, monolithic fuel,  $U_3O_8$ /Al,  $U_3Si_2$ /Al, U-10Mo/Al.

## INTRODUCTION

Nuclear fuels are materials capable of producing energy through a self-sustaining process of nuclear fission. They contain fertile materials such as U-235 or Pu-239, shown in many shapes as solid and liquid forms. The Research and material test reactors have employed dispersion fuel plates manufactured using an aluminum matrix. The fuel plates in which it dispersed the fissile material in powder form within an aluminum matrix to form a fuel core clad between two aluminum alloy sheets [1].

Early experiments used aluminum in the function of the superior features, such as low cost, accessible manufacturing, a low abortion cross-section, higher corrosion resistance, and excellent mechanical response. Fuel technology used for the research reactors has adopted dispersion using Al or Zr as a metallic matrix and fissile compounds based on ceramic fuels (e.g., uranium dioxide, carbides, and nitrides) [2].

Summarily, the timeline of dispersion fuel shows at least three phases. In 1950, Starting from the U-Al fuel system, increasing the volume of UAl<sub>x</sub> in the fuel plates had its limits, UAl-Al<sub>x</sub> dispersion fuel, with uranium densities increase of 30% to 50% compared with U-Al alloys in 1960 [3]. In 1980, the U<sub>3</sub>Si<sub>2</sub>-Al (77 wt.% U<sub>3</sub>Si<sub>2</sub>) for low-temperature applications showed a higher uranium density of 4.8 g.cm<sup>-3</sup>, and U<sub>3</sub>Si for high-temperature applications. In 1992, occur an increase in U<sub>3</sub>Si<sub>2</sub> density of 12.2 g. cm<sup>-3</sup> [4-5]. The current phase has developed a U-Mo monolithic form with densities of 17 g.cm<sup>-3</sup>. At the turn of the millennium began European plans, such as FUTURE and Surface Engineering of Low ENRICHED Uranium–Molybdenum (SELENIUM) to irradiations of disperse low enrichment uranium (LEU) (19.7% <sup>235</sup>U) based U-7w%Mo, with 8gU.cm<sup>-3</sup>, finished in 2015 [6]. In this period, American partners concentrated on developing monolithic UMo while Europe focused on dispersed solutions.

Highly enriched uranium (HEU) fuels with (over 20% <sup>235</sup>U) were used worldwide in research reactors for isotope production. However, in 1978, a new horizon to avoid the proliferation risk occurred with the Reduced Enrichment for Research and Test Reactor (RERTR), supported by the US Department of Energy (DOE). In the 1980 decade, the RERTR program reduced the usage of HEU fuels.

Aluminide fuels (UAl<sub>x</sub>-Al) with a density range of 0.4-0.8 gU.cm<sup>-3</sup> and peak densities of 1.7 gU.cm<sup>-3</sup>. Since 1990, metallic UMo alloy fuels have been one of the adequate fuels for converting HEU to the more proliferation-resistant LEU.

The U<sub>3</sub>Si<sub>2</sub>-Al is recognized as one of the best options, with high uranium loading and enrichment below 20% [7]. After the Fukushima accident in 2011, there has been a renewed interest in higher fissile density fuels, such as UN. and U<sub>3</sub>Si<sub>2</sub>, for light water reactors mainly due to their proven high thermal conductivity [8]. On the other hand, has been tested many uranium alloys, such as U<sub>6</sub>Fe, U<sub>6</sub>Mn, and many more combinations.

The code of federal regulation emitted the 10 CFR 50.64 limited the use of HEU fuels for American research reactors in 1986 [9]. However, there are still a few HEU reactors in operation. The University of Missouri Research Reactor (MURR), operating at 10 megawatts using an enrichment of 93%, is the university's most powerful research reactor. The NBSR is heavy water moderated and cooled, financed by the National Institute of Standards and Technology (NIST) work with fuel dispersion and enrichment of 93%.

This study focuses on dispersion fuel used in research and test reactors, showing the multiple options, such as ceramic dispersed and ceramic-metal dispersed in the aluminum matrix and metallic fuel. The main objective comprises permitting complementary concepts of dispersion and monolithic fuel, showing the art state of the fuel plate technology.

## MATERIALS AND METHODS

Over the years, research reactors have used fuel plates manufactured of uranium fissile particles dissolved in an inert aluminum matrix. The plates in these reactors consist of a fuel core, or “meat,” in an aluminum alloy cladding. Formerly, cast and wrought alloys of uranium and aluminum, consisting of  $UAl_3$  and  $UAl_4$  precipitates in an aluminum matrix, were used for the fuel meat. Thus, it is essential to understand uranium behavior.

### Properties of dispersion fuels

Uranium has three allotropic forms: solid from 25 °C to 1132 °C and liquid phase. In the solid state, uranium shows three phases orthorhombic, tetragonal, and gamma. The orthorhombic  $\alpha$ -uranium phase had four atoms per unit cell, stable from room temperature to < 667 °C. Phase transformation  $\alpha$ - $\beta$  occurs at 667 °C with a volume increase ( $\Delta V/V$ ) of 1.06.

The uranium structure shows a tetragonal-beta phase, stable from 667 °C to 775 °C, revealing 30 atoms per unit cell. In the second phase transition,  $\beta$ - $\gamma$  at 775 °C appoint to volume decreasing ( $\Delta V/V$ ) of 0.736 because of Body-centered cubic (BCC) present in the gamma-phase is stable between 775 °C to 1132 °C or melting point. Ideally, the  $UAl_x$  comprises a combination of  $UAl_2$ ,  $UAl_3$ ,  $UAl_4$ , and other Aluminum phases. Table I shows a simple comparison of some physical properties of intermetallic and ceramic fuels.

Physical properties	$UAl_x$	$UAl_2$	$UAl_3$	$UAl_4$	$U_3Si_2$	$U_3O_8$	U-7Mo	U-10Mo
Melting point (°C)	NA	1590	1350	730	1665	1150	1145	1150
Density (g.cm <sup>-3</sup> )	6.4	8.1	6.8	6.0	12.2	8.3	18.4	18.2
U densities (gU.cm <sup>-3</sup> )	4.5	6.6	5.1	4.1	11.3	7.0	17.1	16.4

Table I. Comparison of intermetallic fuels  $UAl_x$ ,  $U-xMO$  with the ceramics  $U_3O_8$  and  $U_3Si_2$

Tracing an evolutionary timeline of dispersion technology found four periods. During 1950, predominate U-Al alloy materials, followed by  $UAl/Al_x$  dispersion fuel, with uranium densities increase of 30% to 50% compared with U-Al alloys in 1960.. Figure 1 shows the thermal conductivity of uranium, U7Mo, and U10Mo.

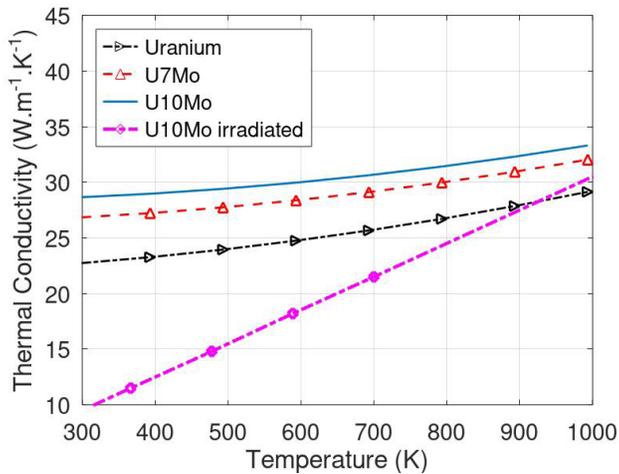


Figure 1: Thermal conductivity of uranium and versus temperature

During 1980, the silicide dispersion  $U_3Si_2/Al$  was typically 7.5 wt% Si, with densities superior to U-Al. In 1990,  $UMo/Al$  started, followed by monolithic fuels. Early experiments used aluminum for its superior features, such as low cost, accessible manufacturing, a low absorption cross-section around 0.22 barn, higher corrosion resistance, and excellent mechanical response. Over the years, the research reactors adopted uranium intermetallic fuels, such as U-Al, U-Si, and U-Mo, which provided an easy solution for neutron sources and material tests. Uranium metallic has a higher density of  $18.1 \text{ g.cm}^{-3}$  and thermal conductivity of  $35 \text{ W.m}^{-1}.\text{K}^{-1}$ . The fuel element is formed by assembling a series of spaced fuel plates, allowing the light water flow that serves as a coolant and moderator. Figure 2 displays some thermal conductivity of commons materials used in dispersions.

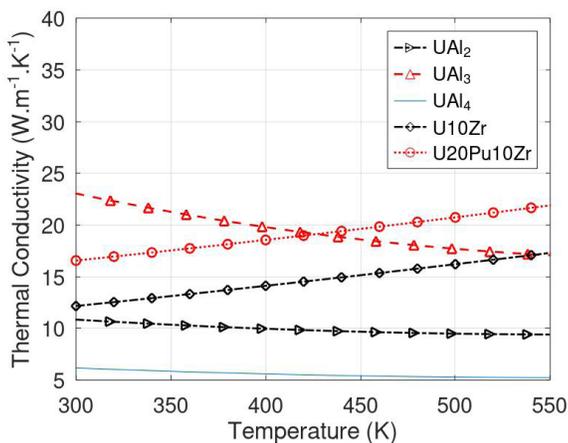


Figure 2: Thermal conductivity of metallic fuels

Physical properties	U	U <sub>3</sub> Si <sub>2</sub>	U-10Mo	U-6Zr-2Nb	U-7Nb
Density (Ug cm <sup>-3</sup> )	19	18.1	16.06	12.92	15.0
Melting point (°C)	1132	1665	1160	1160	1499
Thermal conductivity (W.m <sup>-1</sup> .°C <sup>-1</sup> )	27	8.5	11.3	11.4	21.1

Table II. Fissile materials to produce aluminum dispersion fuel for thermal and fast reactors.

Over time, many applications of aluminum dispersion at high temperatures reveal a higher potential for self-propagation of aluminothermic reactions under irradiation. Besides, U-Al dispersion showed good thermal conductivity comparable to metal alloys and excellent corrosion resistance. Also, it shows suitable mechanical strength at high temperatures and easy fabrication. An essential drawback of the U-Al<sub>x</sub> dispersion is a low uranium content, of 1.7 gU.cm<sup>-3</sup>, in the fuel meat found (37 vol% of UAl). Besides, the swelling is an unwanted result of the chemical reaction between uranium and aluminum intermetallic compounds produced, so-called aluminides. Table III shows the properties of uranium-aluminide compounds contained in the fuel plates.

Compound	UAl <sub>2</sub>	UAl <sub>3</sub>	UAl <sub>4</sub>
Melting Point (°C)	1619.85	1349.85	730.85
Enthalpy of formation (kJ.mol <sup>-1</sup> )	-92.27	-104.60	-129.99
Theoretical density (g.cm <sup>-3</sup> )	8.14	6.8	6.06 - 5.7
Uranium density (gU.cm <sup>-3</sup> )	6.64	5.08	4.16
Uranium content (wt.%)	81.3	74.4	68.5 - 64
Formation energy ΔE (kJ.mol <sup>-1</sup> )	6.8	-9.3	-3.8
Lattice volume (Å <sup>3</sup> )	115.630	77.285	187.285
Crystal system	FCC	Cubic	Orthorhombic

Table III. Physical properties of uranium-aluminide compounds contained in the fuel plates.

In the 1960s, the Oak Ridge National Laboratory (ORNL) and Argonne National Laboratory (ANL) developed the manufacturing route of the U3O8 powder by oxidation of uranium metal in air oxygen. The U3O8-Al dispersion fabrication process combines powder metallurgy hot roll bonding procedure, which may generate swells or blisters if heated above 600 °C. Early, U3O8-Al dispersion fuel made it possible to increase between 30% to 50% the volume fraction of fissile material. Before 1978, the U3O8-Al dispersion fuel produced contained a ceramic compound with a density of 1.3 gU.cm<sup>-3</sup> in fuel meat 18 vol% of U<sub>3</sub>O<sub>8</sub>. The Experimental programs qualified some lower-density fuel dispersion, including UAl<sub>x</sub>-Al with a uranium density of 2.3 g.cm<sup>-3</sup> and U<sub>3</sub>O<sub>8</sub>-Al for densities up to 3.2 gU.cm<sup>-3</sup>. The UZrHx shows uranium loading for up to 3.7 gU.cm<sup>-3</sup> and U<sub>3</sub>Si<sub>2</sub>-Al for up to 4.8 g.cm<sup>-3</sup>.

Though aluminum dispersion has many advantages, many remaining subjects need a deep study, such as the interaction layer formed between uranium and aluminum for metallic fuels U-Mo and ceramic U<sub>3</sub>Si<sub>2</sub>/Al generating (UAlSi)<sub>3</sub> phase. Moreover, the current

phase has developed a U-Mo monolithic form with densities of  $17 \text{ g.cm}^{-3}$ . During the evolution of intermetallic fuels, many tests were performed using  $\text{U}_6\text{Fe}$ ,  $\text{U}_6\text{Mn}$ , U-MO, and much more combinations.

Developing low-enriched fuels to replace high-enriched fuels used for high-flux reactors requires increased uranium density to offset the enrichment decrease. U-Mo/Al dispersion fuel with uranium loadings of up to  $8 \text{ g.cm}^{-3}$  under irradiation increased coarse porosity between the matrix and the interaction phase when the fuel-Al interaction consumed a significant part of the Al-matrix. Besides, the meat porosity depends on a metallurgic powder route, which can reduce with atomized powder from 12% to 1.2%.

The oxidation of the cladding is a barrier to the thermal flux transport because of the low thermal conductivity of the boehmite of  $2.25 \text{ W.m}^{-1} \text{ C}^{-1}$  than Al-6061 alloy of  $160 \text{ W.m}^{-1} \text{ C}^{-1}$ . This way, the  $\text{U}_3\text{Si}_2/\text{Al}$  submitted at high heat flux over  $550 \text{ W.cm}^{-2}$  produces a dense oxide layer developed, followed by a layer with a granular aspect, most probably caused by pitting corrosion.

## Brazilian research reactors

In this context, In Brazil, there are four research reactors in operation: the IEA-R1 and the MB-01 reactors, both at the Nuclear Energy Research Institute (IPEN, São Paulo); the Argonauta, at the Nuclear Institute of Nuclear Engineering (IEN, Rio de Janeiro) and the IPR-R1 reactor, at the Nuclear Technology Development Center (CDTN, Belo Horizonte). The IEA-R1 is in operation supported by IPEN, with its first criticality in 1957. It has a thermal neutron flux of  $4.6 \times 10^{13} \text{ n.cm}^{-2} \text{ s}^{-1}$  and a fast flux of  $1.3 \times 10^{14} \text{ n.cm}^{-2} \text{ s}^{-1}$  under a steady state.

IEA-R1 operates ceramic fuel-dispersed  $\text{U}_3\text{O}_8\text{-Al}$  with demineralized water serving as coolant and moderator at the velocity of  $0.8 \text{ m.s}^{-1}$  and uses beryllium and graphite as reflectors. In 1997, the IEA-R1 got an operating license for 5 MW. In 1998, the IPEN started producing and qualifying its own  $\text{U}_3\text{O}_8/\text{Al}$ , with a density of  $2.3 \text{ g.cm}^{-3}$ . Since 2007, the IPEN has expanded into manufacturing fuel plates using uranium silicide  $\text{U}_3\text{Si}_2/\text{Al}$ , with a density of  $3.0 \text{ g.cm}^{-3}$ . Training, Research, Isotopes, General Atomics (TRIGA) IPR-R1 reactor, acquired in 1960 by current CDTN, was the second research reactor installed in Brazil, with the first criticality notified in 1960.

TRIGA IPR-R1 reactor, acquired in 1960 by current CDTN, was the second research reactor installed in Brazil, with the first criticality notified in 1960. The timeline of IPR-R1 had various purposes, from iodine irradiation for health applications to the production of radioisotopes for industrial applications and tracers for environmental studies. In worldwide are 66 units installed in 24 countries, operating with metallic fuel, the uranium-zirconium hydride ( $\text{UZrH}_x$ ).

## Ceramic fuels and metallic fuels

Nuclear fuel technology widely uses ceramics as a superior option for a vast gamma of applications, mainly in light water reactors, providing about 10% of the global electricity demand with nearly 443 power reactors in operation. Ceramic oxides and metallic cladding comprise the most used fuel systems, completing over sixty years of experience. Ceramic oxides, such as  $\text{UO}_2$ ,  $\text{U}_3\text{Si}_2$ , and mixed oxide fuel (MOX)  $(\text{U-Pu})\text{O}_2$  and  $(\text{Th-Pu})\text{O}_2$ , using zirconium alloys as a coating, are preferable in thermal reactors. Besides, fast reactors do not use neutron moderators like water or graphite. The best nuclear option is plutonium or composed ceramics, such as nitrides  $(\text{U-Pu})\text{N}$  or carbide  $(\text{U-Pu})\text{C}$ .

The Ceramic-metallic (CERMET) fuels were research objects of the nuclear space propulsion program, which started in 1953 and ended in 1973. Cermet fuel comprises ceramic fuel particles uniformly dispersed in a metal matrix. The aerospace program GE700 developed cermet fuel, such as  $\text{W-UO}_2$  and  $\text{Mo-UO}_2$ , from 1962 to 1968 in ANL, besides for a naval propulsion plan to qualify uranium nitride (UN) particles dispersed in a tungsten-molybdenum alloy matrix (W/Mo-UN). Nautilus was the first nuclear submarine, with a sea trial in 1955, operating with light water as a coolant and moderator. Table IV shows the thermal properties of ceramics dispersed into Al-matrix at 25 °C.

Physical Properties	$\text{UO}_2$	$\text{U}_3\text{O}_8$	$\text{U}_3\text{Si}_2$	$(\text{U-20Pu})\text{O}_2$
Melting Point (°C)	2875	1150	1665	2750
Density ( $\text{g}\cdot\text{cm}^{-3}$ )	10.06	8.4	12.2	11.07
Uranium density ( $\text{gU}\cdot\text{cm}^{-3}$ )	9.7	7.1	11.3	9.77
Thermal conductivity ( $\text{W}\cdot\text{m}^{-1}\cdot\text{C}^{-1}$ )	8.68	6.0	10.2	7.82

Table IV. The thermal properties of ceramic fuels used in light water reactors.

This way, the Nautilus had a compact core work with highly enriched uranium, using plates of  $\text{UO}_2$  dispersed in a zirconium matrix. However, the old option fuel planned for the naval reactor showed  $\text{UO}_2$  dispersed in a metal matrix of zirconium or stainless steel. Figure 3 displays the linear thermal expansion coefficients of materials used in dispersions.

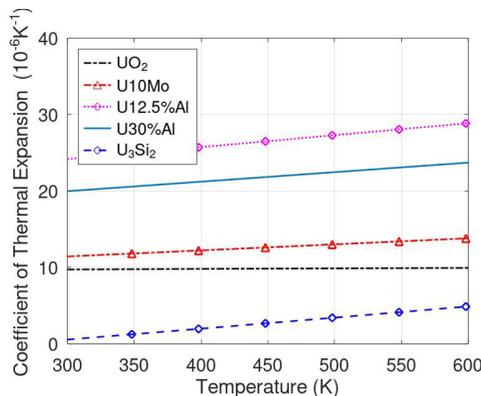


Figure 3: Linear thermal expansion of fuels used in dispersions.

Metal fuels are ideal for fast reactors, showing higher densities of fissile and fertile materials and providing higher core performance, such as a higher breeding ratio and less fissile inventory. The U-Zr hydride fuel with densities of  $3.7 \text{ g/cm}^3$ , using aluminum or stainless steel as cladding, is the most used in training reactors (TRIGA). Thus, has arisen other alloys, such as U-Zr, U-Zr-Nb, and U-Pu-Zr alloys.

## RESULTS AND DISCUSSION

The U-Al<sub>x</sub>-Al dispersion's thermal conductivity depends on the uranium content weight fraction and porosity. Assuming an average porosity of the order of 7%, with 20 wt% of uranium, found a thermal conductivity is  $160 \text{ W}\cdot\text{m}^{-1}\cdot\text{K}^{-1}$  and decreases with uranium increase. When uranium content reaches 70 wt%, the thermal conductivity drop to  $20 \text{ W}\cdot\text{m}^{-1}\cdot\text{K}^{-1}$ . In part because the thermal conductivity of metallic uranium is reduced compared with Al-6061. The thermal conductivity of U<sub>3</sub>Si-Al and U<sub>3</sub>Si<sub>2</sub>-Al dispersions also rely on fuel content and porosity. With 20 wt% of U<sub>3</sub>Si<sub>2</sub>, the thermal conductivity is  $90 \text{ W}\cdot\text{m}^{-1}\cdot\text{K}^{-1}$ . When it reaches 50 wt%, the thermal conductivity drop to  $22 \text{ W}\cdot\text{m}^{-1}\cdot\text{K}^{-1}$ . For U<sub>3</sub>Si, the thermal conductivity at 20 wt% is  $120 \text{ W}\cdot\text{m}^{-1}\cdot\text{K}^{-1}$  and drops to small values at 50 wt%. For U<sub>3</sub>Si and U<sub>3</sub>Si<sub>2</sub>, the porosity will increase with volume content that achieves a maximum of 50%. Table V displays some aluminum matrix properties.

Physical properties	Al-6062	Al-606-T4	Al-606-T6	Al-7075-T6
Temperature, solidus to liquidus (°C)	582 - 651	582 - 651	582 - 651	477 - 635
Thermal conductivity ( $\text{W}\cdot\text{m}^{-1}\cdot\text{K}^{-1}$ )	189	154	167	173
Specific heat capacity ( $\text{J}\cdot\text{kg}^{-1}\cdot\text{°C}^{-1}$ )	896	896	896	960
CTE ( $\mu\text{m}\cdot\text{m}^{-1}\cdot\text{°C}^{-1}$ )	23.6	23.6	23.6	21.6
Ultimate tensile strength (MPa)	117	241	310	524
Yield tensile strength (MPa)	48.3	145	276	462
Elongation at break (%)	25%	22%	17	3%
Modulus of elasticity (GPa)	68.9	68.9	68.9	71.7
Shear modulus (MPa)	26	27	26	26.9
Poisson	0.33	0.33	0.33	0.33

Table V. Aluminum-matrix materials normally used for dispersions U<sub>3</sub>Si<sub>2</sub>/Al and U<sub>10</sub>Mo/Al

U<sub>3</sub>Si<sub>2</sub> is a qualified fuel with high uranium loading and performance for research and test reactors. Also, U<sub>3</sub>Si<sub>2</sub> fuel shows high resistance to driven gas swelling. Under the manufacturing route of U<sub>3</sub>Si<sub>2</sub>, it exhibited chemical reactions with Al-matrix to form some U(SiAl)<sub>3</sub> phases, U<sub>3</sub>Si, and U<sub>3</sub>Si<sub>2</sub>. The composition and homogeneity of the alloy and heat treatment determine the quantity of each aluminide phase present. The two phases, U<sub>3</sub>Si<sub>2</sub> and USi, can exist at equilibrium. This reaction between silicide intermetallic and aluminum matrix produces an interface layer that transforms crystalline into amorphous compounds

with lower thermal conductivity. Still, the critical temperature to occur amorphization is 290 °C.  $U_3Si_2$  has advantages compared with  $U_3Si$  related to fission gas swelling: comparatively  $U_3Si$  forms giant bubbles, and  $U_3Si_2$  tiny bubbles. Both  $U_3Si_2$  and  $U_3Si$  have a thermal conductivity of  $\sim 15 \text{ W}\cdot\text{m}^{-1}\cdot\text{C}^{-1}$ . The average thermal expansion coefficients of  $U_3Si_2$  and  $U_3Si$  over the range of 20°C to 600°C are  $15.2 \times 10^{-6}$  and  $15.8 \times 10^{-6}$  per °C, respectively.

## Fuels swelling

The swelling level is a function of the temperature, irradiation of fuel density, and concentration of the fissile uranium. Because of the fission of  $U^{235}$ , the swelling has two reasons: the solid fission products can induce crystal damage coupled with fission gas forming bubbles. Thus, it reduces the thermal conductivity of the fuel plate, producing a higher temperature gradient. Figure 4 shows the consequence of temperatures on total fuel swelling in fuel plates.

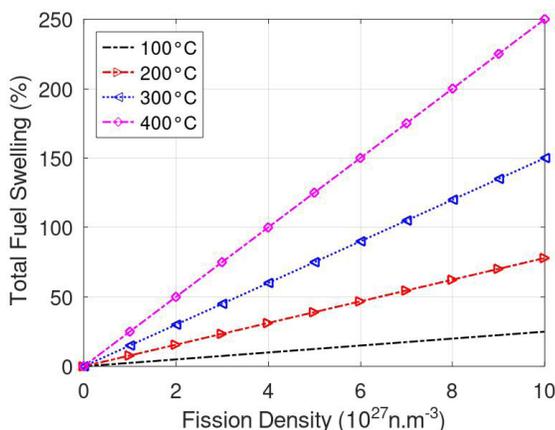


Figure 4: Fuel plate swelling versus irradiation temperature.

Then, the fission gas released speeds up the loss of the mechanical properties of the plate. The swelling curve exhibits a sharp knee that shifts to a higher fission density. The knee of the curve happens because of the rapid multiplication of the gas bubble, and the bubble size increases linearly. UMo surfaces. Equations (A) and (B) representing swelling rate given as function of fission and molybdenum content in weight for  $U_xMo$ .

$$\frac{\Delta V}{V} = 5.8336 \times 10^{-23} \cdot (1.25 - 2.5w_{Mo}) \cdot f, \text{ for } f \leq 2.0 \times 10^{21} \quad (\text{A})$$

$$\frac{\Delta V}{V} = (1.25 - 2.5w_{Mo}) \cdot [0.1667 + 1.1667 \times 10^{-22} \cdot (f - 2.0 \times 10^{21})], \text{ for } f \geq 2.0 \times 10^{21} \quad (\text{B})$$

where  $\Delta V/V$  represents the fuel swelling for the U-Mo alloy,  $w_{Mo}$  is the weight fraction of Mo in the fuel alloy, and  $f$  represents the fission density ( $\text{fissions}\cdot\text{cm}^{-3}$ ).

In UMo/Al, swelling produces geometric deformations, which could fracture. The swelling is a function of fission densities;  $3 \times 10^{21}$  f/cm<sup>3</sup> has about 20% of swelling, while the fission rate of  $7 \times 10^{21}$  f.cm<sup>-3</sup> achieves 44%. Swelling deformation increases the stress on meat fuel, and it calculated the cladding matrix as a function of fission density. Figure 5: Uranium loading versus fission phase percents.

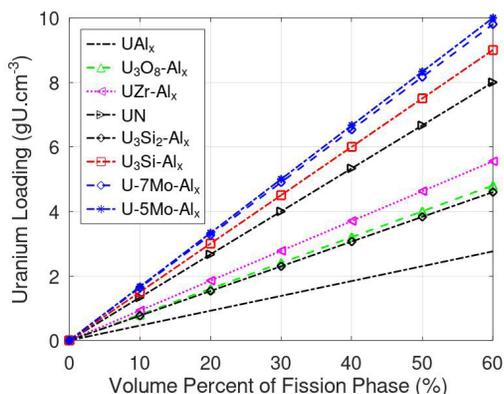


Figure 5: Uranium loading versus fission phase percents

## Interdiffusion layer

Interaction layer (IL) growth between fuel particles and the matrix partially represents the plate fuel performance. The interaction layer growth data for U<sub>3</sub>Si, U<sub>3</sub>Si<sub>2</sub>, and USi dispersions in the Al matrix show a few empirical correlations for silicides dispersion. The IL growth rates did not decrease with the decreasing U/Si ratio by fission, while the growth is a typical interdiffusion process between U, Si, and Al. Early irradiation studies showed that adding small amounts of silicon to the aluminum matrix drastically altered the composition of the (U-Mo-Al-Si) zone, reducing irradiation-induced growth. However, the thickness of the interaction layer grows more slowly than in UMo/Al dispersions.

The growth rate of the interface depends on temperature under irradiation. Besides, U<sub>3</sub>Si<sub>2</sub>/Al dispersions are free of porosity formation, in contrast to UMo/Al. Although U<sub>3</sub>Si is unsuitable for a plate-type geometry because of unstable swelling, it is still applicable for fuel rods. Both U<sub>3</sub>Si<sub>2</sub> and USi are brittle U<sub>3</sub>Si is rigid and relatively soft.

The microstructure of U<sub>3</sub>Si<sub>2</sub>/Al dispersions fuel plates submitted to a high heat flux suffer unwanted changes. At severe conditions performed using a maximum heat flux of 520 W.cm<sup>-2</sup>, leading to a cladding surface temperature of 180 - 200°C and fuel meat temperatures of 220 - 240°C. As a result, these experiments found degradations on the outer aluminum surfaces of the U<sub>3</sub>Si<sub>2</sub> coupled with higher fission products released at maximum fuel burnup of 25% U-235. Equation (C) represents an empirical correlation for UxMo fuel given as a function of fission and fraction of molybdenum in weight.

$$y^2 = 2.2443 \times 10^{-19} \cdot (1.625 - 6.25w_{Mo}) \cdot (f^{0.75}) \cdot \Delta t \cdot \exp\left\{\frac{-10000}{RT}\right\} \quad (C)$$

where  $y$  is the interaction thickness (cm),  $w_{Mo}$  is the weight fraction of Mo in the fuel alloy,  $f$  is the fission rate density (fissions.cm<sup>-3</sup>.s<sup>-1</sup>),  $\Delta t$  is time (seconds), and  $R$  represents the gas constant (1.987 cal.mole<sup>-1</sup>.K<sup>-1</sup>), and  $T$  is the temperature (K).

The presence of Mo in U introduces a complexity to the interdiffusion behavior of U–Mo. Aluminum reacts with molybdenum, forming intermetallic or ternary phases, such as U<sub>6</sub>Mo<sub>4</sub>Al<sub>43</sub> and UMo<sub>2</sub>Al<sub>20</sub>, U<sub>6</sub>Mo<sub>4</sub>Al<sub>43</sub>. However, adding some wt% Si to the Al matrix can retard the formation of the UMo/Al interaction layer and change the properties of the interaction phase. Since silicon has a higher chemical affinity for U than Al, this way, it forms a silicon-rich interface layer between UMo particles and Al, which retards UMO/Al interdiffusion but does not avoid it.

## CONCLUSION

This research was based on the physical properties of aluminum dispersions fuel, such as U/Al, U3O8/Al, U3Si2/Al, and U10Mo/Al. Many behaviors analyzed strongly depend on thermal conductivity, thermal expansion, fission-induced swelling, and interaction layer growth. Besides, the results used are over time databases produced in international research programs, such as RERTR, SELENIUM, and FUTURE, resulting from a large irradiation spectrum. Studying the thermal properties, such as thermal conductivity and expansion, can produce an understanding of fuel plate deformation behavior. The thermal properties fundamental for a simulating mechanical response have been a critical safety aspect. The production of an interaction layer is an unwanted effect that reduces the life of fuel plates.

Over the years, a rapid evolution in the manufacturing route of dispersion fuels has occurred, and many results reported do not offer a perfect conclusion of values, showing significant uncertainties. However, its model for fuel meat swelling is based on published in the open literature. The mathematical models got from swelling results from diversified experimental data and theoretical results. There are a few interaction layer growth models, partly because of the vast gamma of nuclear facilities employed. Besides, each laboratory uses a particular manufacturing route and works with different fuels composition and other aluminum cladding. Although the maximum IL thickness can achieve 1.68 mm, it decreases the thermal conductivity of the fuel meat, especially under the significant fission rate and long irradiation time.

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