

LESSONS FROM THE ATOMS-FOR-PEACE RESEARCH REACTOR FUELS PROGRAM, 1955-1965

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ABSTRACT

This paper describes relevant findings from 1950-1965 work on both high-density LEU fuels and high-performance HEU fuels and how these findings relate to current work being performed to develop and qualify LEU fuel for the high-performance research reactors. The problems experienced in fabricating aluminum fuels with heavy uranium loading are reviewed. The adverse impacts of aluminide formation and densification observed in this early work are summarized. The effect of adding silicon and other elements to minimize or eliminate the problem in aluminum matrix fuels, which was studied and patented by Oak Ridge National Laboratory staff in 1958-60, is assessed. In the early 1960s, ORNL also fabricated aluminum clad HEU fuels using niobium coated uranium-zirconia particles and evaluated their irradiation performance at very high burn-up (65 atom percent). Some comments are also made on equilibrium versus non-equilibrium phase formation in a radiation environment.

1. Introduction

Oak Ridge National Laboratory (ORNL) under funding from the National Nuclear Security Administration (NNSA) has initiated a feasibility study of the potential conversion of the High Flux Isotope Reactor (HFIR) from high enriched uranium (HEU) fuel to low enriched uranium (LEU) fuel. There will be many facets to this feasibility study including reactor physics and thermal-hydraulic analyses but the practicability of fabricating and qualifying an economic high-density LEU fuel suitable for HFIR is a materials challenge for which the solution will require the best efforts of many organizations within the Department of Energy (DOE) complex. As part of this feasibility assessment, we are reviewing the long history of developing aluminum (Al) based nuclear fuel technology to make sure that key lessons are recognized and incorporated in current efforts that may impact HFIR's future.

From the 1940s until the 1960s, ORNL played a key role in the development of Al-clad fuels for research reactor applications including the fabrication of the first LEU core for the Atoms for Peace Program later called the Foreign Research Reactor Fuels Program. Stemming from the Atoms for Peace initiative and from the need for higher power density cores using HEU, ORNL pioneered fabricating and testing high-density alloy and ceramic fuels for research reactors. ORNL research into powder fuel fabrication techniques eliminated several options such as uranium dioxide due to its reactions with Al and uranium carbide due its susceptibility to water corrosion. Dating back to improving the performance of the Al clad for uranium metal fuel slugs in the Graphite Reactor in the late 1940s, ORNL identified, tested and later patented the use of silicon (Si) and other solutes as a mitigating alloying agent to minimize the formation of undesirable uranium aluminides (UAl_4). ORNL also developed fabrication procedures for hot and cold rolling of higher strength Al clad (6061) replacing the casting and rolling of the previously-used high purity Al clad material.

In addition to work at ORNL, key parallel efforts during this same period, in which ORNL often collaborated, were performed at Battelle Memorial Institute (BMI) in the development of methods for applying protective coatings (diffusion barriers) to uranium alloys and uranium ceramic compounds envisioned for use as nuclear fuels. Also the Atomics International (AI) Division of North American Aviation (NAA) tested diffusion barriers in its short-lived attempt to develop less expensive Al clad for uranium-molybdenum (U-Mo) alloy fuels to be used in organic moderated and cooled power reactors. There were also efforts in Europe in this period to find diffusion barrier materials to prevent U-Al interactions in nuclear fuel applications.

The lessons learned from these parallel programs are relevant to today's efforts to develop and qualify high-density alloy fuels to allow reliable and economic conversion of high-performance research reactors such as HFIR to use LEU.

2. ORNL research in the Atoms-for-Peace Program

The scientific approach used by ORNL was based on detailed characterization of the fabricated test articles both before and after other testing and on the use of out-of-pile separate effects testing (such as heating and water corrosion effects on the fuel meat and clad) to complement and supplement data from irradiation testing and post-irradiation examinations. The fabrication of test articles was performed with production-sized equipment to ensure the technology could be transferred to industrial deployment of applications. ORNL also utilized diverse measurement techniques for any parameter being quantified to validate methods and results. Demonstrating reproducibility of fabrication methods was a key part to this approach. Findings from this period such as the inability of x-ray radiography to quantify fuel meat segregation may be a function of the technology available at the time, but assuring diversity and redundancy in measurement technologies remains paramount for assuring success in research, development and deployment of materials and associated technologies for quality assurance in production. In today's environment, the availability of data bases (such as phase diagrams) and analytical tools to model and predict the results of fabrication operations provides resources for the current program that were not available during the Atoms-for-Peace Program.

The evolution of these efforts and their findings in the Atoms-for-Peace Program can be tracked in the documentation given in the periodic progress reports of the ORNL Metallurgy Division starting in 1948 and after 1961 of the ORNL Metals and Ceramics Division. Prior to 1959, many of these reports were originally classified and subsequently declassified.

In 1950, ORNL completed fabrication of the first HEU Al-clad plate fuel for the 3 MW(thermal) Low Intensity Test Reactor (LITR), which served as a full-scale mock-up at ORNL for the 30 MW(thermal) Materials Test Reactor (MTR) then under construction in Idaho. Subsequently, ORNL fabricated both plate-type and concentric-tube-type Al-clad fuel for various research reactors [1]. One of the early problems faced with MTR fuel in 1952-53 was the less than expected core excess reactivity. ORNL initially addressed this by increasing the uranium loading, but, in performing complete destructive chemical analyses of fuel plates, ORNL found that neutron poisons (lithium and chlorine) were being trapped in the side plates by the flux used

in brazing the Al clad fuel plates to the Al side plate. Fabrication procedures were modified and tested to correct the problem.

In 1952-53 in initial efforts to reduce fuel enrichment in research reactors, ORNL began fabricating higher density fuels than could be achieved by casting of U-Al alloys. Casting of high density LEU fuel had shown segregation (multi-phase inhomogeneities) in the product as the uranium loading is increased [2] and the rolled fuel plates developed cracking at the edges of the fuel meat. Using metallic uranium powder in fabrication was considered too difficult due to fire safety considerations. Initial fabrication of fuel compacts with uranium dioxide (UO_2) and Al powders showed unacceptable dimensional changes during sintering of the compacts both in air and in other gases due to what was then thought to be the formation of U_3O_8 [3]. This was an important finding since the fabrication process involved a number of hot operations in air, but the initial conclusion was revisited later. Compacts made with U_3O_8 and Al showed much improved dimensional stability during sintering. During 1953, a number of batches of fuel compacts and rolled plates with varying U_3O_8 loadings from 20 to 47 weight-percent in Al were fabricated and checked for segregation using several different measurement techniques. Up to four percent variation occurred in uranium loading along the plates. The earlier runs also showed that contaminants such as carbon, which had been coated on the fuel meat to avoid bonding to the clad during rolling to facilitate post-rolling testing of the fuel meat, can reduce U_3O_8 to UO_2 thus changing the fuel composition.

In 1955, in less than four months, ORNL fabricated the fuel plates for the first LEU core for the display reactor at the 1955 Geneva Conference on the Peaceful Uses of Atomic Energy [4]. Initial efforts for fabricating UO_2 fuel elements were unsuccessful due to what was later proven to be the high-temperature reactions between the UO_2 and Al that occur during certain fabrication steps (hot-rolling at 590°C , flux-annealing at 510°C , and brazing at 610°C) and can cause plate warping and blistering. To resolve this problem, the Geneva Reactor was fabricated using an ORNL-developed and specially prepared high-ceramic UO_2 powder (bipyramidal, very dense, free of clinging agglomerates and much less prone to forming UAl_3 , UAl_4 and Al_2O_3). Lacking irradiation data for this fuel, two Geneva Reactor fuel elements were fabricated and irradiated, one in LITR to five percent burn-up of the ^{235}U and another in MTR to 23 percent burn-up of the ^{235}U . Both irradiations demonstrated excellent dimensional stability of the fuel plates when a high-ceramic and more expensive form of UO_2 is used and irradiated at the relatively cool temperatures of these pool-type reactors.

Following the Geneva Reactor work, ORNL continued the search for acceptable high-density research reactor fuel options. Initially work focused on the problem of segregation in LEU-Al cast or extruded alloys containing 25 to 45 weight-percent uranium [5] and of fuel-meat cracking and clad thinning (so called “dog bone”) during fuel plate rolling. Work also began on roll-cladding using extruded fuel-meat bars (not foils) instead of castings and on developing boron-Al burnable poisons for research reactors. Rolling technologies were developed by ORNL for higher strength clad materials such as Al 6061, and this reduced the clad thinning problem. An MTR fuel element with up to 48 weight-percent LEU (~ 1.95 grams of uranium per cm^3) in each of the 18 full-sized plates was successfully irradiated in MTR to a burn-up of 25 percent of the ^{235}U with no adverse effects [5]

Ultimately, five high-density LEU fuel options were evaluated in terms of fabricability, fuel-meat water corrosion resistance, and irradiation performance for loadings of 45 to 55 weight-percent uranium in Al. These options were U-Al alloy, UC and UC₂ dispersions in Al [6], UO₂ dispersions in Al, U₃O₈ dispersions in Al, and UAl₂ dispersions in Al.

Concern about high-temperature reactions of UO₂ dispersions in Al and the need for an expensive high-ceramic powder to mitigate the potential for high-temperature reactions limited the consideration of this option [3]. Problems with fabricating brittle U-Al alloys due to the presence of UAl₄ created by heated operations during fabrication were overcome by adding 3 weight-percent Si to the alloy to suppress significant UAl₄ production [7-10]. The work with uranium carbides at 50 weight-percent in Al showed that the UC dispersions are subject to significant dimensional change due to reactions with the Al during fabrication. Equivalent work with the UC₂ dispersions found that these are more stable but need the high-temperature portions of the fabrication process to be conducted in the absence of hydrogen. However, subsequent corrosion testing determined that fuel meat dispersions of UC₂ corrode catastrophically in 20 and 60°C purified aerated water after only 120 hours thus limiting any serious long-term consideration of this option. In comparison, equivalent water corrosion testing of fuel-meat dispersions of U₃O₈ out to 26 weeks showed only minor surface corrosion and that of the Si-stabilized-alloy fuel meat showed random blister formations, but the researchers report that the alloy test article was likely contaminated by inclusions of UC₂ in the alloy during fabrication.

Irradiation tests of LEU fuel were performed in MTR for multiple fabricated mini-plates containing the Si-stabilized alloy with 64 weight-percent uranium, 62 weight-percent U₃O₈ dispersions, and 60 weight-percent UC₂ dispersions [11]. This irradiation test series was designated ORNL-MTR-35. The estimated centerline fuel-meat temperature during these irradiations was 80°C. The burn-up range spanned in the tests was from 15 percent to 60 percent of the ²³⁵U atoms. Detailed destructive post-irradiation examinations were conducted at several burn-ups. Density changes as a function of burn-up reveal that dispersions of UAl₃ (the U-Si-Al alloy) and UC₂ exhibited decreases in density, whereas dispersions of U₃O₈ showed density increases. The density of dispersions of U₃O₈ increased rapidly during the early stages of irradiation and then appeared to stabilize and remain unchanged even after rather extensive burn-up of the ²³⁵U atoms. The density data are supported by dimensional measurements. The UC₂ dispersions show significant voiding with irradiation due to interactions with surrounding Al. The magnitudes of the changes in density and dimensions in the specimens containing dispersions of UAl₃ and U₃O₈ were not considered sufficient to cause deleterious effects in the performance of research reactor fuel elements in applications that require the use of LEU. Microstructure examinations of the irradiated U₃O₈ dispersion fuel indicated that reactions with Al only occurred in the finer particulates of U₃O₈. Out-of-pile heating tests of fuel elements at 600°C had shown that the U₃O₈ reacts quickly to form UO₂ at this temperature and then the UO₂ reacts with Al to form U-Al intermetallic compounds.

As noted above, work at ORNL [7-10] and elsewhere [12] in the late 1950s on U-Al alloys demonstrated the important role of solute additions such as Si to suppress UAl₄ formation by acting as a diffusion barrier. Figure 7 in Reference [12] summarizes the reaction layer rate constants determined by U-Al and U-(Al-11.7 percent Si) diffusion couples following exposures from 1 to 475 hours over a wide temperature range [12]. The interaction layer thickness was

observed to obey parabolic behavior varying with the square root of time over a wide range of conditions. It can be seen that the addition of 11.7 percent Si caused a significant reduction in the thickness of the UAl_4 interaction layer, particularly at high temperatures [12].

Additions of several percent of Si, zirconium (Zr), germanium (Ge) or tin (Sn) were found to be effective in suppressing UAl_4 formation during fuel fabrication [8-10]. The (U-M) Al_3 phase (M = Si, Zr, etc.) that formed instead of UAl_4 improved the workability of the alloy by reducing the intermetallic phase volume fraction by ~18% for a given uranium loading (due to the higher U loading in UAl_3). Whereas 48%U-Al exhibited significant cracking after rolling corresponding to a reduction thickness of only 10 percent, a 48%U-3%Si-Al alloy was successfully rolled to 75 percent reduction in thickness with only limited cracking near the edges of the plate [9]. Figure 10 of Reference [9] compares the optical microstructures of 48%U-Al and 48%U-3%Si-Al alloys following 75% reduction in thickness by hot rolling at 600°C. The addition of Si clearly results in a reduction in the volume fraction of the U-Al intermetallic phase. Irradiation in the MTR at low temperatures produced only a 2.2 percent reduction in density in the 48%U-3%Si-Al fuel after a burn-up of 60.9 percent of ^{235}U [10]. However, irradiation to moderately high burn-ups (>35 percent ^{235}U) resulted in the formation of an unidentified interaction layer around the periphery of the UAl_3 phase, as shown in Figure 18.2 of Reference [11]. It can be speculated that irradiation at higher temperatures than the MTR exposure (due to higher power densities in high-performance research and test reactors) could accelerate the formation of this interaction layer and lead to unstable fuel performance.

Basically, ORNL work on the Atom-for-Peace Program generated two LEU product lines, up to 65 weight-percent U_3O_8 dispersions in Al (~2.0 grams of uranium per cm^3 from an ORNL internal document) and the three percent Si-stabilized U-Al alloy (~2.2 grams of uranium per cm^3 from an ORNL internal document). In the late 1950s, ORNL transferred the alloy fabrication technologies to Babcock & Wilcox in Lynchburg for implementation, and alloy-type LEU cores were fabricated by Babcock & Wilcox for MTR and later for export to the Netherlands, Germany, and Brazil. In 1960, ORNL fabricated U_3O_8 LEU dispersion fuel for the pool reactor at the Puerto Rico Nuclear Center. The technology for fabricating U_3O_8 dispersion fuel was transferred by ORNL to Texas Instruments. After completion of the post-irradiation examination of test series ORNL-MTR-35 in 1961, ORNL was focused on the development of high-density HEU fuel for the high-performance research reactors then under development, HFIR and the Advanced Test Reactor (ATR). Much of these latter efforts focused on improved fabrication technology and on comparative irradiations for U_3O_8 dispersions in Al selected for HFIR and the alloy fuel selected for ATR.

3. ORNL Research Based on Battelle Memorial Institute (BMI) Results, 1953-1962

Starting in 1953, BMI used various techniques to apply metallic coatings to both metallic and ceramic substrates. These chemical vapor deposition (CVD) processes employed among others hydrogen reduction of metal pentachloride gases. These processes worked well for coating ceramics with a tight metal film but could not be used for coating the many alloys (such as uranium) that do not resist chloride corrosion and leads to the formation of an underlying chloride salt layer below the metal coating [13].

Based on a 1962 BMI proposal [14] and from BMI work with niobium (Nb) coating of UO_2 particles in the early 1960s [15], ORNL fabricated, destructively inspected, and then irradiated in the Engineering Test Reactor (ETR) Al-HEU fuels using Nb-coated uranium dioxide-15% zirconium dioxide particles [16, 17]. The makeup of the irradiated specimens is described as “56 weight percent Nb-coated $\text{UO}_2\text{-ZrO}_2$ with 0.16 weight percent B_4C .” The oxide particles had been coated using hydrogen reduction of Nb pentachloride gas.

Detailed destructive post-fabrication inspections of the fuel plates showed that the Nb coatings on spherical UO_2 -15 weight-percent ZrO_2 particles can be maintained during roll-bonding with a hot reduction of 84 percent at 500°C and a subsequent cold reduction in thickness of 20 percent. In the rolled plate, the coated fuel particles retained their shape and resisted degradation in the form of stringing and fragmentation. There was also no indication of Nb-Al interactions from the 500°C operations during fabrication.

Post-irradiation testing of Nb-coated HEU particles demonstrated excellent retention of fission products during irradiation, no evidence of substantial reaction between the Nb and Al and no evidence of diffusion of the Al through the Nb during irradiation. However, the post-irradiation examinations of the plates did show excessive swelling (11.8 percent) at high burn-up (~65 atom-percent HEU) compared to swelling observed from the equivalent irradiation of HEU-Al fuels fabricated with uncoated U_3O_8 dispersed particles (7.3 percent swelling) and UAl_3 -alloy fuels (5.1 percent swelling). One plate showed warping attributed to an uneven distribution of coated particles. The maximum nominal clad temperatures during irradiation varied between 198°C and 268°C , and the fission density varied from 8.1 to 17×10^{20} fissions per cm^3 . Post-irradiation anneal experiments showed that the breakaway swelling temperature for a test plate with Nb-coated $\text{UO}_2\text{-ZrO}_2$ particles with nominally the same ^{235}U and ^{10}B content as the U_3O_8 and UAl_3 test articles was 260°C at a fission density of 11×10^{20} fissions per cm^3 . The breakaway swelling temperatures for the U_3O_8 and UAl_3 test articles were respectively 400°C and 428°C at a burn-up of 15×10^{20} fissions per cm^3 . The findings reported in [17] state:

“The niobium-coated UO_2 plate cores more nearly approach the anticipated swelling. As an example, Plate 0-4-844 had an observed volume change of 11.8 percent compared to an expected 12.0 percent. At first appraisal this would indicate that the $\text{UO}_2\text{-ZrO}_2$ particles had very little capacity to accommodate fission products. Examination of the microstructures revealed matrix cracks developing and particle cracking voids enlarging with increased irradiation which must be accounted for in the observed swelling. Based on the measured increase in particle diameters, 7.3 percent swelling was produced within the Nb casings. The remaining swelling ($11.8 - 7.3 = 4.5$ percent) then was produced outside the Nb shells primarily by a growth interaction of the agglomerated particles on one another.”

The swelling of the coated particles indicates that the Nb coating should be applied either to a less dense fuel kernel or to a more accommodating swelling-resistant fuel material for the target burn-up. The additional swelling of the plates indicates that an improved particle distribution is needed to avoid the agglomeration of coated particles and the resulting matrix cracks; however, Reference [17] also states in conclusion that “the individual particles swelled to the degree that matrix cracking was obtained” and “post-irradiation annealing showed that matrix cracking

permitted the fuel core to swell substantially at reactor operating temperatures.” Thus particle swelling impacts matrix cracking but more so where particles can agglomerate. The authors have not attempted to correlate the observed neutron-irradiation-induced swelling in the UO_2 - ZrO_2 Nb-coated particles with data on U-Mo swelling as a function of the fission density to see if Nb-coated particles might make sense for an all metallic fuel. Depending upon the options considered for HFIR, this attempt at correlating swelling data may be done in the future.

4. Relevant Related Research at Atomics International (AI), 1955-1966

As indicated earlier in this paper, AI was interested in developing diffusion barriers to support the use of less expensive Al clad for U-Mo alloy fuels to be used in organic moderated and cooled power reactors. A key focus of the AI program was the development and testing of diffusion barriers to prevent aluminide formation at the interface of the Al clad and the LEU-Mo alloy fuel meat. A wide range of candidate diffusion barrier coatings was evaluated, with the two most promising candidates identified to be nickel (Ni) and Nb. Starting in the late 1950s, AI focused initially on electroplated-Ni as the diffusion barrier in its fabricated fuel with Al clad. The fuel irradiations in both the experimental and demonstration versions of the organic moderated and cooled reactors were to very low burn-up (~2500 MWD/MT) at peak clad temperatures of ~400°C. These irradiation tests resulted in ~20% of the 254 micron thick electroplated-Ni diffusion barrier being consumed by aluminide formation [18, 19]. Out-of-pile isothermal soak-tests of unirradiated fuel elements at 400°C for times ranging from 1000 to 4000 hours show complete consumption of the Ni diffusion barrier at 4000 hours and the initiation of the diffusion of Al into uranium [20]. The testing demonstrates that Ni reacts readily with Al but much less so with uranium. Out-of-pile diffusion couple tests indicate that the penetration depth of Al into Ni follows a parabolic curve varying with the square root of time and decreasing with decreasing temperature. The Al-Ni penetration coefficient drops by an order of magnitude as the temperature is reduced from 400°C to 300°C. This implies that Ni may work just fine as a diffusion barrier in an Al-clad research reactor with U-Mo alloy fuel if the temperatures and exposure times are reduced compared to the AI test conditions; however, high-burn-up fuel irradiation data are lacking for Ni diffusion barriers along with a quantification of the effects of Ni aluminide formation on thermal conductivity of the clad and on possible cracking/debonding at the clad-barrier interface during irradiation. Additional testing is needed to assure the integrity of an electroplated-Ni diffusion barrier in the irradiation conditions expected in a high-performance research reactor such as HFIR.

No irradiation tests are reported by AI for Nb diffusion barriers [21]. The Nb work followed the initial work with Ni and was focused primarily on the development of a bonding mechanism to the LEU-Mo alloy and Al clad. AI did report that, after reviewing the literature at the time on Nb-Al, “Diffusion couple studies subsequently indicated insignificant interdiffusion between aluminum and niobium, and that high-strength bonds could be attained.” These findings are consistent with the results of both the ORNL fabrication work and the ETR irradiation tests of the Nb-coated UO_2 - ZrO_2 particles discussed above. Several techniques were tested by AI to apply Nb to the U-Mo alloy. The most promising techniques for Nb diffusion barrier application reported by AI are electron beam vacuum deposition (that is, physical vapor deposition or PVD) and isostatic pressure bonding where in the latter case Nb foil is bonded to the surfaces of a U-Mo plate by applying gas pressure at elevated temperatures. The isostatic

pressure bonding was performed after vacuum degassing all components at 900°C and acid-pickling the Nb and U-Mo components. The isostatic pressure bonding was then performed between steel sheaths in a vacuum at 1050°C and 82.5 MPa (12,000 psi) for one hour.

Apparent good bonding of Nb to U-Mo was achieved in the cases of both Nb PVD and isostatic pressure bonding. A contaminant inclusion caused a small debonding of the Nb and U-Mo on one plate produced using isostatic pressure bonding. However, the subsequent application of Al clad by roll-bonding left discontinuities and pinholes in both sets of Nb diffusion barriers. Accelerated heated soak tests at 540°C for 198 hours showed diffusion of Al through the Nb barrier discontinuities leading to the formation of a UAl_x layer inside the diffusion barrier where the aluminide was considered by the AI researchers likely to have been UAl_4 .

Since AI reported no further work in this area, additional research is needed to develop better techniques for applying a tight layer of Nb to uranium and for avoiding the production of discontinuities during roll-bonding of the clad. Alternately, for metallic-coated particle fuels, the PVD technique or another coating mechanism would need to be developed to the point of providing high-quality leak tight Nb-coated particles assuming that electroplated-Ni is insufficient or undesirable for the application.

The AI reports also cite the contemporaneous European work that was being performed on the development of diffusion barriers; however, the patent [22] emerging from the European work, which looked at vanadium and chromium as well as Ni and Nb, applies to fuel rods and is not particularly relevant to the fuel configuration in HFIR or in the other U.S. high-performance research reactors.

5. Conclusions

The impetus behind producing this paper is to make sure that the vast amount of historically relevant technical information is made available for consideration and use during the upcoming studies of the feasibility of HFIR conversion to use LEU. This information provides useful scientific, engineering and manufacturing details on the experience and approaches that may have been abandoned in the past because the technology at the time lacked the ability to execute the realization of concepts with merit or to quantify key aspects of those concepts. An example of progress is in the current synergism that exists among the materials, computational and manufacturing sciences which have made great advances since the 1950s and 1960s. This is especially true for the ability to model fabrication processes and to model equilibrium and non-equilibrium phase formation in either a fabrication or a radiation environment. Phase formation is most important since understanding the formation dynamics of UAl_x in terms of the variations in such parameters as density and thermal conductivity and the behavior in the presence of phase-formation-impeding elements such as Si directly impacts the ability to predict fuel performance. In an irradiation environment, the diffusion of metal atoms driven by neutron knock-on or fission product recoil collisions as well as temperature directly impacts the timing for non-equilibrium phase formation just as time and temperature at pressure drive the diffusion and phase formation in fabrication. The Atoms for Peace results from work performed at ORNL and elsewhere suggest that 1 to 10 weight-percent alloying additions of solutes such as Si, Zr, Ge and Sn can be effective in improving the fabricability and irradiation performance of U-Mo

dispersed fuels, but high flux (high nuclear heating rate) conditions may increase atomic mobility and lead to breakaway swelling problems. These historical results suggest for dispersed U-Mo fuels that multiple approaches such as additions of the listed alloying elements to the Al matrix and use of particle diffusion barrier coatings such as Ni or Nb should be investigated in order to obtain a more radiation resistant fuel form for the demanding conditions of high-flux LEU reactors.

6. References

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