THE CONVERSION OF NRU FROM HEU TO LEU FUEL

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ABSTRACT

The program at Chalk River Nuclear Laboratories (CRNL) to develop and test low-enriched uranium fuel (LEU, <20% U-235) is reviewed, and the status of the conversion of the NRU reactor from highly enriched uranium (HEU, 93% U-235) to LEU fuel is discussed. The replacement LEU fuels developed and tested at CRNL contain high-density uranium silicide particles dispersed in aluminum, in cylindrical rods. The silicides tested include U_3Si, USiAl, USi*Al and U_3Si_2 (U-3.96 wt% Si; U-3.5 wt% Si-1.5 wt% AL; U-3.2 wt% Si-3 wt% Al; U-7.3 wt% Si, respectively). Fuel elements were fabricated with uranium loadings suitable for NRU, 3.15 gU/cm^3, and for NRX, 4.5 gU/cm^3, and were irradiated under normal fuel-operating conditions. Eight experimental irradiations involving 100 mini-elements and 84 full-length elements (7x12-element rods) were completed to qualify the LEU fuel and the fabrication technology. Post-irradiation examinations confirmed that the performance of the LEU fuel, and that of a medium-enrichment uranium (MEU, 45% U-235) alloy fuel tested as a back-up, was comparable to the HEU fuel. The uranium silicide dispersion fuel swelling was approximately linear up to burnups exceeding NRU's design terminal burnup (80 at%). NRU was partially converted to LEU fuel when the first 31 prototype fuel rods manufactured with industrial-scale production equipment were installed in the reactor. The rods were loaded in NRU at a fuelling rate of about two rods per week over the period 1988 September to December. This partial LEU core (one third of a full NRU core) has allowed the reactor engineers and physicists to evaluate the bulk effects of the LEU conversion on NRU operations. As expected, the irradiation is proceeding without incident.

1. INTRODUCTION

The NRU reactor at CRNL is a multipurpose, tank-type thermal research reactor, using D_2O moderator and coolant and highly enriched uranium (HEU, 93% U-235) U-Al alloy driver fuel. Various kinds of rod assemblies are suspended in the tank on a 7 1/4 in (19.7 mm) hexagonal lattice. The reactor produces radioisotopes, provides neutron beams for research, and has special facilities for testing metallurgical specimens and advanced power reactor fuels, and for performing fuel damage (blowdown) tests.

The NRU reactor achieved criticality in 1957 and ran at 220 MW (Th) using driver fuel rods containing flat plates of natural uranium metal. It was converted to highly enriched, rod-type driver fuel in 1964, and currently runs at about 130 MW (Th). The linear-fissile loading of each 2.7 m long fuel element (twelve per rod) has risen from 0.3 g U-235/cm to the present 1.8 g U-235/cm. Roughly 90 of the 227 reactor sites are occupied by driver fuel, and these are taken to approximately 80% burnup in 11 months.

As part of an international effort to reduce the use of HEU, and the risk of nuclear proliferation, a development program was undertaken at CRNL to produce low enrichment uranium (LEU, <20% U-235) driver fuel for NRU. CRNL's progress has been reported at annual meetings sponsored by the U.S. Reduced Enrichment for Research and Test Reactors (RERTR) program [1-4]. The salient features of the LEU fuel development and testing program, and the current status of the NRU conversion program are reviewed in this paper. The LEU fuel burnup analysis has recently been completed using high-precision liquid chromatography, and where appropriate, the analyzed burnup values will be given.

2. FUEL DEVELOPMENT AND FABRICATION

The NRU fuel-rod design which consists of twelve elements, each containing an HEU-Al alloy core with finned aluminum cladding and aluminum end plugs (see Figure 1), has proven to be extremely durable and reliable. The geometry, dimensions and linear fissile loading of this 12-element fuel rod were maintained in the LEU conversion program for licensing simplicity. CRNL physics studies showed very little effect on reactor operations from using LEU driver rods, apart from the need to fuel 6 more reactor sites or reduce exit burnup by about 10%. The problem thus came down to the development of a replacement fuel with five times the HEU fuel uranium density, which would be stable to high burnup.

The replacement LEU fuel elements consist of a core, containing high-density uranium silicide particles dispersed in an aluminum matrix, with the same finned aluminum cladding and end plugs. The fuel is made by melting uranium and silicon in a high-frequency induction furnace. Cast billets are heat-treated to transform the as-cast structure to U_3Si, then the billets are reduced to powders via a series of comminution processes. Uranium silicide powder is mixed with aluminum powder and hot-extruded into cores. The finned aluminum cladding is extruded onto the cores in a semi-continuous process, with the cores acting as a floating mandrel. The cladding is welded to the end-plugs to hermetically seal the elements.

The high-density silicides tested were U_3Si, U_3Si alloyed with 1.5 wt% Al and 3 wt% Al, and U_3Si_2 (U-3.96 wt% Si; U-3.5 wt% Si-1.5 wt% AL; U-3.2 wt% Si-3 wt% Al, and U-7.3 wt% Si, respectively). Fuel elements were fabricated with uranium loadings suitable for NRU, 3.15 gU/cm^3, and for NRX, 4.5 gU/cm^3. Following the decision to shut down NRX, testing of NRX-type fuel was discontinued.

Industrial-scale fuel-production equipment has been installed in temporary facilities at CRNL, and is currently being used to manufacture prototype fuel rods to familiarize the operators with the manufacturing process, and to confirm that the production rate is high enough to meet CRNL's annual fuel requirements. To date, more than 90 kg of LEU has been processed into NRU rods (see section 4) and the production line is running satisfactorily. Most of the process variables and parameters established during the development program have been used in the production line and no problems have resulted from scaling up.

A new building has been constructed for LEU fuel fabrication. The manufacturing equipment is scheduled for installation and re-commissioning in late 1989. It is anticipated that full-scale production of NRU LEU fuel will commence early in 1990 in the new building, and the first production fuel rods will be loaded into NRU later that year. NRU should be completely converted to LEU fuel within a year of the first production of LEU fuel rods in the new building.

3. LEU SILICIDE DISPERSION FUEL TESTING

3.1. Test Elements

The test vehicle for irradiating silicide dispersion fuel in most tests has been the mini-element. The mini-element fuel-core diameter (5.5 mm) and clad wall thickness (0.76 mm) are the same as in full-size NRU elements. Mini-elements are, however, only 184 mm long compared with 2.9 m for NRU elements. The mini-elements also resemble NRU elements in that they have six cooling fins at 60° intervals around the cladding, the fin width being 0.76 mm and fin height 0.96 mm. To date, approximately 100 mini-elements have been successfully irradiated to burnups in the range 56-93% U-235 depletion in the NRU and NRX reactors.

3.2. Irradiation Conditions

The LEU silicide dispersion fuels were irradiated in NRU. A medium-enrichment uranium (MEU, 45% U-235) fuel tested as a backup was irradiated in NRX. The mini-elements were irradiated in a fuel carriage made from an aluminum cylinder with four holes bored axially through it at 90° intervals. An aluminum liner containing a string of four mini-elements was inserted into each hole or flow channel. The mini-elements were located centrally in the flow channels by four-pronged anodized spiders located on the end spigots of the mini-elements. The assembly could be loaded in any of the normal driver fuel positions in NRU.

The mini-elements were irradiated at linear powers representative of a typical NRU driver fuel rod, ranging from 40 to 112 kW/m. The typical neutron flux was approximately 1.1 x 10^{18} $n/m^2/s$, the heavy-water coolant flow approximately 7.28 L/s, and coolant velocity approximately 10.9 m/s. The coolant inlet temperature ranged between 30 to 37°C and the coolant outlet temperature between 40 to 45°C during the mini-element irradiations.

3.3. Mini-Element Test Results

The fuel test matrix is shown in Table 1. The detailed results were reported elsewhere [1-4], therefore only the salient features of the irradiations will be reported here.

3.3.1. Comparison of HEU-Al and LEU Silicide Dispersion Fuel Performance

In Exp-FZZ-905, mini-elements containing Al-21 wt% U alloy fuel (93% enriched U), Al-37 wt% U alloy fuel (45% enriched U) and Al-61.5 wt% USiAL (20% enriched U) silicide dispersion fuel, with 0.63 $gU\text{-}235/cm^3$, the fissile loading suitable for use in NRU (and for the new MAPLE-X reactors), were irradiated to 57% U-235 burnup (analyzed). The elements were in excellent condition after irradiation; they appeared as they did in the as-fabricated condition except for a dull oxide layer on the cladding. Post-irradiation examinations (PIE) showed that the performance of the uranium silicide dispersion and the MEU alloy fuel was comparable to the HEU alloy fuel. All element diametral changes were less than 1.6% and length changes were within 0.2% after 57 at% burnup. As shown in Figure 2, the Al-HEU fuel microstructure remained essentially unchanged after irradiation but the LEU silicide particles reacted with the aluminum matrix material, forming a thin interfacial layer, probably UAl_3 with dissolved Si, around the fuel particles. Fission-gas bubbles ranging in size up to 5 μm were contained in the kernels of the fuel particles. Considerably less fission-gas bubbles were retained in the interfacial layers. Fuel-core swelling ranged between 3.4 to 4.5 vol% at the exit burnup. From the results it appears that all three materials swell at roughly the same rate, up to 57 at% burnup.

3.3.2. High-Burnup Confirmation

In the Exp-FZZ-909B irradiation, mini-elements containing the USiAl and USi*Al dispersions, with 3.15 gU/cm^3, were irradiated up to 89 at% burnup (analyzed). Immersion density measurement indicated that the cores had swollen by 5.92-7.63 vol% after 77 at% burnup and by 6.57-7.76 vol% after 89 at% burnup, see Figure 3. These results showed swelling was approximately linear right up to 89 at% burnup, and confirmed that NRU-composition silicide dispersion fuels could exceed the design terminal burnup (80 at%) without exceeding the threshold of breakaway swelling.

PIE showed that both dispersions behaved similarly, i.e., the uranium silicide reacted with the aluminum matrix and fission-gas bubbles formed in the fuel particles. The interfacial layers were thinner near the fuel-core periphery and their edges were more sharply defined than at the fuel-core centre. The fission-gas bubbles were about the same diameter (up to 5 μm) in both locations but thicker interfacial layers and more particle coalescence had occurred at the fuel-core centre.

3.3.3. Dispersions with High U Loading and Fine Particles

In the FZZ-909A experiment, Al-USiAl and Al-USi*Al dispersions containing the higher loading required for NRX (4.5 gU/cm^3) were tested. Fuel performance was good and the materials behaved similar to the dispersions containing 3.15 gU/cm^3; however, swelling was marginally over 1 vol% per 10 at% burnup, at the terminal burnup of 74 at%.

In the Exp-FZZ-910 experiment, dispersions with a high loading of fine particles showed greater swelling compared to fuel with coarser particles at similar burnup. Mini-elements containing USiAl and USi*Al dispersions with fine particle-size distributions and 4.5 gU/cm^3 loading exceeded the swelling envelope of approximately 1 vol% per 10 at% burnup of the previous mini-element irradiations. The results suggest that particle size must be closely controlled to ensure good performance at high U loadings.

3.3.4. In-Reactor Corrosion

In-reactor corrosion behaviour of uranium silicide dispersion fuels has been investigated in the FZZ-911 and the FZZ-915 irradiations. The mini-elements contained Al-USiAl and Al-USi*Al, 3.15 gU/cm^3. In the FZZ-911A experiment, four mini-elements had 1.2 mm diameter holes drilled in the cladding mid-section, and were irradiated in the linear power range 60-87 kW/m in NRU. The first and second mini-elements were removed from the reactor after reaching 19 and 32 at% burnup, respectively, and the remaining two after 48 at% burnup (analyzed).

Post-irradiated metallography and neutron radiography revealed that ellipsoidal cavities had developed beneath the holes in the cladding. The cavity size increased with increasing burnup. These cavities correspond to 1.1 mg and 3.2 mg of U-235 lost to the coolant after 19 and 32 at% burnup, respectively, and 9.8-48.0 mg U-235 after 48 at% burnup. These results indicate that the corrosion rate of the purposely defected fuel elements is acceptably low.

The FZZ-915 experiment was similar to the FZZ-911 experiment, except that the mini-elements were pre-irradiated to burnups in the range of 23 to 79 at% (analyzed) before the 1.2 mm diameter holes were drilled in the cladding. These elements were further irradiated in the NRU reactor to evaluate the performance of the defective fuel, and during the additional 38 full-power days no increase in activity in the coolant above the normal background was detected. Neutron radiography and metallographic examinations revealed that the cavities were typically 0.7 mm deep by 1.3 mm across, i.e., only marginally larger than the original cavity made by the drill tip. These results indicate that the corrosion resistance of the LEU fuel is possibly increased by previous burnup.

3.3.5. Fuel-Core Surface Imperfections

The objective of the FZZ-911B experiment was to evaluate the performance of intact mini-elements having slight as-fabricated or deliberately introduced imperfections (machined grooves) in the core surface. However, the mini-elements contained the same fine particles that caused enhanced swelling in Exp-FZZ-910. Therefore, examinations were also carried out to determine the effects of the fine particle size on the core swelling when the loading is 3.15 gU/cm^3.

Post-irradiation examinations revealed that the aluminum cladding had flowed into and filled the surface defects. More importantly, the core volume of the FZZ-911B mini-elements had only increased by approximately 4.9% after approximately 60 at% burnup compared with 7.0 to 17.5% swelling in the FZZ-910 cores at about the same burnup. Swelling was 7.1 vol% after 80 at% burnup. These results indicate that core surface defects had no detrimental effect on fuel performance, and fuel swelling was acceptable at the lower silicide loading required for NRU, even when fine particles were used.

3.3.6. Effect of Particle Size on Core Swelling

In the Exp-FZZ-918 experiment, 16 mini-elements containing Al-61.4 wt% uranium silicide were irradiated in NRU to help establish limits on the particle-size distribution to be used in the manufacturing specifications. The mini-elements were divided into 4 groups, each group containing progressively lower fractions of fines (particles less than 44 μm in size).

Fuel swelling was linear with burnup, and to a first approximation proportional to the percentage of fines contained. After 89 at% burnup (analyzed) the mini-elements containing a high proportion of fines swelled by 6.6 to 6.8 vol% while the mini-element with a low fraction of fines swelled by 5.8 vol%.

3.3.7. AL-U_3Si_2 Dispersion Fuel

We have recently expanded the program to include U_3Si_2 dispersions to complement the U_3Si line. Twelve Al-U_3Si_2 mini-elements (3.15 gU/cm^3) were fabricated with a variety of particle-size distributions and installed in NRU in 1988 June. The assembly was removed for interim post-irradiation examinations after reaching approximately 60 and 80 at% burnup, then it was returned to the reactor to continue the irradiation to 93 at% burnup. Metallographic examinations showed that the U_3Si_2 fuel behaved similarly to the U_3Si dispersions. An interfacial layer formed around the fuel particles, and fission-gas bubbles, ranging in diameter up to 10 μm, could be seen in the fuel particles. The U_3Si_2 was not heat treated, and contained 4 wt% free uranium, the highest level expected from local non-uniformity in full-size castings. This appeared to have no detrimental effect on fuel performance. However, no swelling dependence on particle-size distribution was observed, and the Al-U_3Si_2 dispersion fuel swelling was lower than Al-U_3Si at similar burnup. The results are shown in Figure 4, and are compared with data from Al-U_3Si mini-elements (Exp-FZZ-918).

3.4. Thermal Ramp Tests: Post-Irradiation Heating Tests

Thermal ramping tests were conducted in hot cells to determine the effects of temperature excursions on the dimensional stability and fission-product activity release from previously irradiated silicide dispersion fuel. Whole mini-elements and short segments of mini-elements with the fuel meat exposed were chosen, having fuel burnups of either 23 or 93 at%. Half the samples contained Al-61.5 w/o USiAl and half contained Al-62.4 w/o USi*Al.

The test conditions were: Argon gas flow rate - 1.66 mL/s; Heating rate - 0.2 and 0.4°C/s; Temperature - in the range 530 to 720°C; Holding time - temperature held constant (±4°C) for an hour.

In the thermal ramp tests, a whole mini-element irradiated to 93 at% burnup developed small localized blisters, some with pinhole cracks releasing fission products (^{85}Kr and ^{137}Cs) after 0.25 h at 530°C. This behaviour prevented gross pillowing or ballooning. A mini-element irradiated to 93 at% burnup and ramped to 640°C developed radial cracks, which tore the cladding and released fission products from the core. Even at this high temperature the element maintained considerable structural integrity. This behaviour is interpreted to mean that coolant channels would not become blocked even if the fuel was subjected to some hypothetical abnormal event with the potential to cause overheating of the core, e.g. 530°C compared with the normal operating maximum of 200°C.

3.5. Full-Size NRU Fuel Test Results

The excellent performance of mini-elements containing the USiAl and USi*Al dispersion fuel led naturally to the fabrication and testing of full-size, 12-element NRU fuel assemblies. In experiment FZZ-913, seven assemblies were irradiated in NRU, 3 containing Al-62.4 wt% USi*Al and 4 containing Al-61.0 wt% U_3Si (identified as FL-001 to FL-007). The LEU assemblies were installed in NRU during 1984, replacing the currently used HEU alloy fuel in NRU, and were irradiated at typical driver-fuel operating conditions. The electrical conductivity of the coolant ranged between 0.25 and 0.7 μmho/cm during irradiation. The pH was not routinely measured, but ranged between 5.5 and 7. The coolant inlet temperature ranged between 30-37°C and the outlet between 60-70°C. Inlet pressure was approximately 80 psi (579 KPa) and outlet approximately 30 psi (207 KPa). Each rod occupies an average of 5 core positions during its lifetime (~340 d residence time @ 70% efficiency) as it is moved from the outside of the core (low-flux site) to the centre (high-flux site) and back to the outside. Average element linear power ranged from 40-50 kW/m, with the maximum being approximately 80 kW/m.

Burnup analysis showed that the rods were irradiated to 67 to 84 at% burnup (peak). Visual examinations showed that the fuel elements were in good condition; they were identical in appearance to the HEU elements with the normal aluminum oxide layer coating the surface. Dimensional measurements indicated that elongation during irradiation was negligible; the USi*Al and the U_3Si dispersion fuel elements were within 0.5% of their original length.

Post-irradiation metallography revealed that, as expected, the full-length elements' high burnup behaviour was similar to that of the mini-elements. The fuel particles had reacted with the matrix aluminum forming the normal aluminide interfacial layer (UAl_3 with dissolved Si). The interfacial layer which formed around the U_3Si particles was considerably thinner than that in the USi*Al dispersion. Small fission-gas bubbles were contained in the kernels of the fuel particles and ranged in size up to 10 μm in diameter. Considerably fewer fission-gas bubbles had been retained in the interfacial layers.

Evidence of the axial burnup gradient was observed in the full-length elements. In the high-burnup sections (mid-length) more particle coalescence had occurred than at the lower burnup sections (ends). There was less evidence of particle coalescence in the U_3Si dispersion. However, there was no evidence of the linking-up of fission-gas bubbles in any of the fuels examined, indicating that the fuels were well away from the onset of breakaway swelling. Immersion-density swelling measurements could not be made on the full-length elements, so estimates based on the dimensional changes from the underwater and metallographic examinations have been calculated. Core diameter increases of up to 3% and 4% have been measured at the ends and at the middle of the fuel elements, respectively. These give conservative estimates of the core's swelling by less than 1 vol% per 10 at% burnup at the terminal burnup of 84 at%.

It is clear that the high-burnup performance of USi*Al and U_3Si dispersion fuel containing 3.15 gU/cm^3--the loading required for the research reactors at CRNL--is acceptable. Factors contributing to the good performance were the suitable particle-size distribution and the superior restraint provided by the thick-walled cladding.

4. STATUS OF NRU CONVERSION

NRU was partially converted to LEU when the first 31 LEU fuel rods manufactured using industrial-scale equipment were installed during 1988 September to December. This partial (one third) LEU core allowed the reactor engineers and physicists to evaluate the bulk effects of LEU conversion on reactor operations. Table 2 shows the burnup profiles of selected Al-U_3Si dispersion fuel rods from the first campaign, as of 1989 July. As expected, the irradiation is proceeding without incident and the reactor engineers have seen no difference in fuel-rod behaviour or handling compared to HEU fuel. In a given neutron flux the reactivity worth of the LEU rods is indistinguishable from HEU rods at comparable burnup. Bulk parameters such as coolant/moderator chemistry and overall reactivity have not changed perceptibly with the partial LEU core.

In 1989 July, the first of the rods reached their exit burnup. Two of these rods were cut apart in the NRU bays for PIE. Under-water visual examinations revealed clean, straight elements with only the light oxide coating on the cladding that is normally also seen on the HEU elements. No detailed metallographic examinations were planned since the rods behaved satisfactorily during irradiation, as expected. One of the elements which was inadvertently bent during handling demonstrated good ductility after full irradiation. By the end of 1989 all of the prototype LEU fuel rods will have completed their irradiation in NRU.

Fuel production in the new facility is expected to begin in 1990. It is expected that the full conversion of NRU will commence when a stable fuel production rate exceeding the fuel-usage rate is achieved. The projected date for this milestone is 1990 September.

5. CONCLUSIONS

1. Suitable LEU silicide dispersion fuels have been developed and tested at CRNL for the conversion of NRU from HEU to LEU. Under NRU fuel operating conditions, the fuels are stable up to burnups exceeding the design terminal burnup (80 at%).

2. NRU has been partially converted to LEU; 31 prototype LEU rods containing AL-61 wt% U_3Si fuel (one third of a NRU core) were installed during 1988 September to December. The irradiation is continuing without incident, as expected. Post-irradiation examinations of the first rods to be discharged confirmed that the fuel achieved the design burnup in good condition.

3. Construction of a new fuel-fabrication facility is essentially completed and LEU fuel production is expected to begin in 1990. The complete conversion of NRU is expected to begin in 1990 September.

6. ACKNOWLEDGEMENTS

Many people, some of whom have since left CRNL, have contributed to the success of the LEU fuel development and conversion program. They are too numerous to properly acknowledge here. However, we would especially like to thank J.C. Wood, M.T. Foo, L.C. Berthiaume, L.N. Herbert, J.D. Schaefer, J.G. Goudreau, F.C. Iglesias, R.G. Barrand, J. Mitchell, J.R. Kelm, R.J. Chenier and the CRNL hot-cell metallography group, and E.J. McKee and the rod-shop group, for their contributions.

7. REFERENCES

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TABLE 1 Irradiation Program Status

EXPERIMENT	ELEMENTS	CORE MATERIAL	TEST OBJECTIVES	RESULTS	ANALYZED BURNUP (at%)
FZZ-905	8 4 4	Al-61.5% USiAl Al-21% U Al-37% U	Compare LEU dispersions ($3.15\ MgU/m^3$) with U-Al alloys	LEU fuel performance comparable to MEU and HEU alloys	57
FZZ-909A	6 6	Al-72.4% USiAl Al-73.4% USi*Al	Test dispersions containing $4.5\ MgU/m^3$	Fuel swelling marginally above 1 vol% per 10 at% burnup	73
FZZ-909B	6 6	Al-61.5% USiAl Al-62.4% USi*Al	High burnup confirmation	Fuel swelling <1 vol% per 10 at% burnup and linear up to the terminal burnup	89
FZZ-910	8 8	Al-72.4% USiAl Al-73.4% USi*Al	Test dispersions with fines	Particle size needs to be controlled to minimum swelling	51
FZZ-911	4	Al-61.5% USiAl	Drilled defects in cladding	Corrosion resistance acceptable	18-48
	8 4	Al-62.4% USi*Al Al-61.5% USiAl	Fuel core surface imperfections	Surface defects have no detrimental effect Swelling 5.7 to 6.9 vol%	88
FZZ-913[a]	36 48	Al-62.4% USi*Al Al-61.0% U_3Si	Full-size assembly irradiation	Both dispersions suitable for use in NRU. U_3Si chosen as reference	67-84
FZZ-915	6	Al-61.5% USiAl Al-72.4% USiAl Al-73.4% USi*Al	In-reactor corrosion of pre-irradiated dispersions	Prior irradiation enhances corrosion resistance	23-79
FZZ-918	16	Al-61.4% U_3Si	Define optimum particle size distributions	Swelling proportional to percentage of fines (5.8 to 6.8 vol% after 89 at% BU)	89
FZZ-921	12	Al-64% U_3Si_2	Compare to U_3Si and evaluate effect of particle size	Behaviour similar to U Si dispersions but less swelling	

[a] Full-length NRU 12-element assemblies

TABLE 2

STATUS OF SELECT LEU FUEL IRRADIATIONS (AS OF 1989 JULY 31)

ROD ID	INSERTION DATE	REMOVAL DATE	INITIAL U-235 (g)	BURNUP (% U-235)	POWER MWd
FL008	88 09 29	89 07 30	495.7	80	320.1
FL009	88 09 30	---	494.1	79	314.2
FL010	88 10 01	---	493.9	79	313.1
FL011	88 10 04	89 07 30	497.0	80	321.3
FL012	88 10 06	89 05 31	494.0	61	244.9
FL015	88 10 13	---	492.6	77	304.9
FL020	88 11 07	---	493.3	72	285.3
FL028	88 12 12	---	492.1	71	282.5
FL038	88 12 28	---	494.0	54	215.8

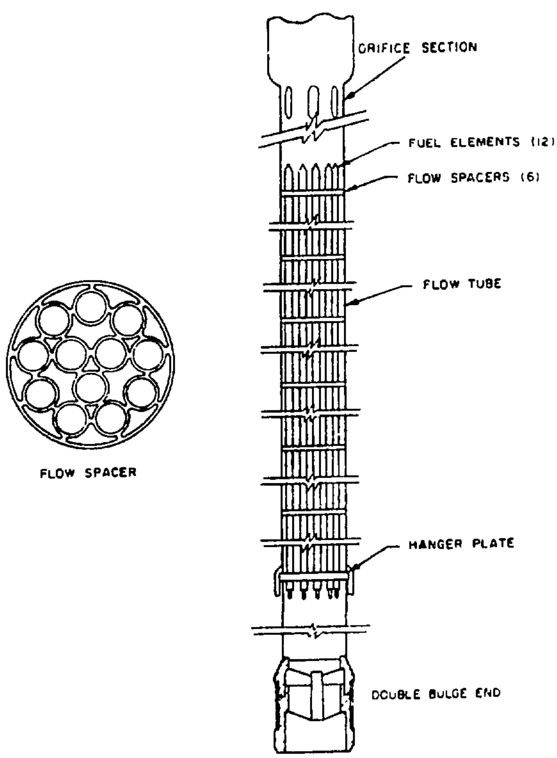

FIGURE 1
NRU FUEL ROD

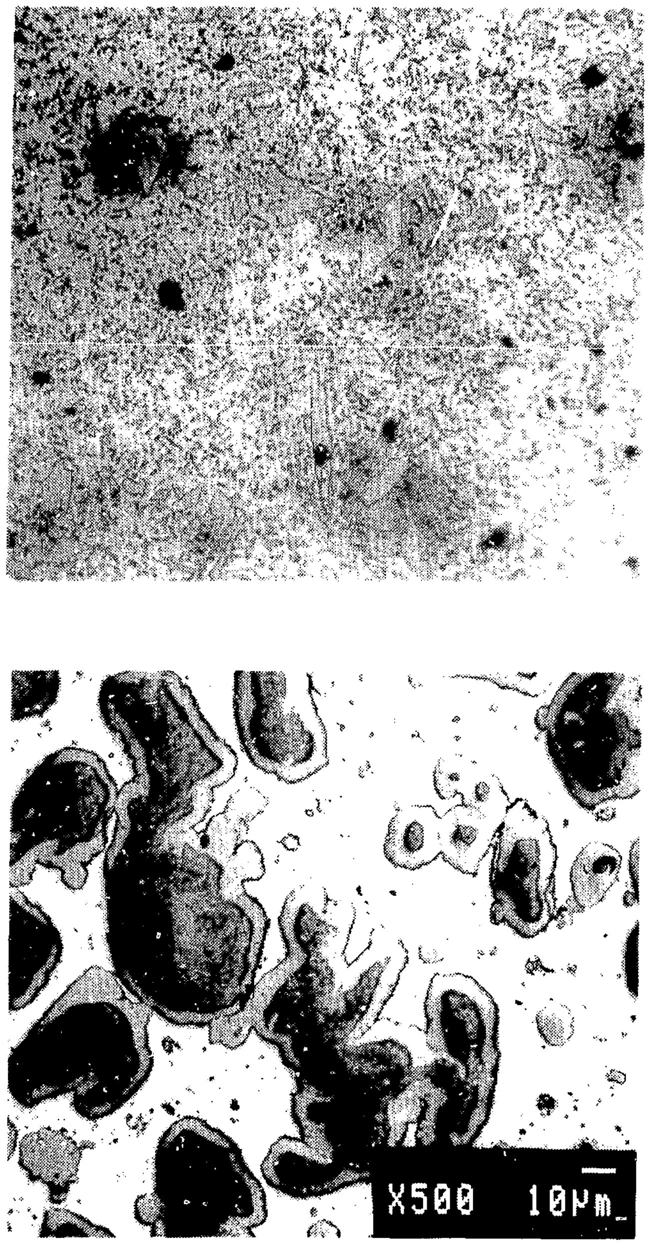

Fig. 2. Microstructure of: (a) Al-21 wt% U fuel after 57 at% burnup, UAl_4 - dark grey, Al - light grey; (b) Al-61.5 wt% USiAL after 57 at% burnup, U_3Si - dark grey, UAl_3 - light grey, Al - white.

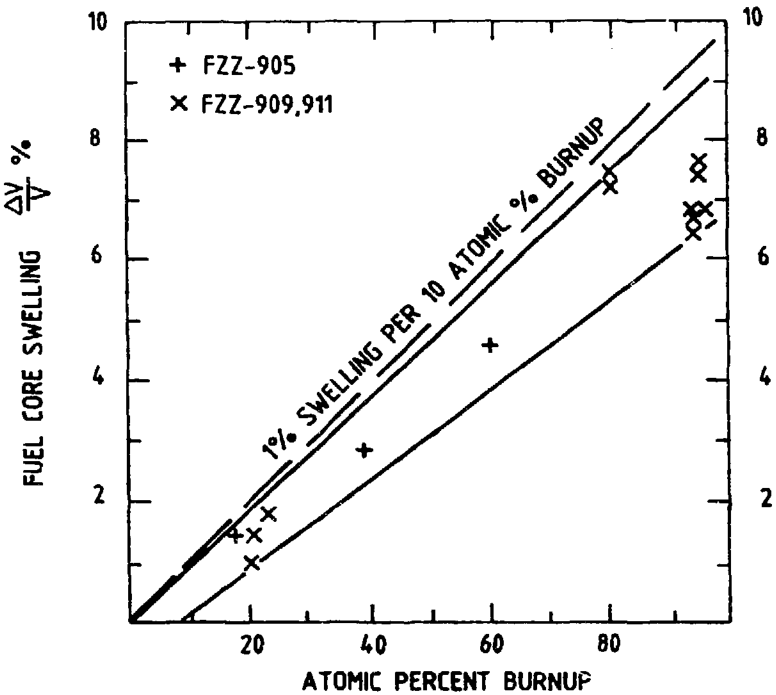

Fig. 3 Swelling of LEU fuel core containing Al-61.5 wt% USiAl and Al-62.4 wt% USi*Al.

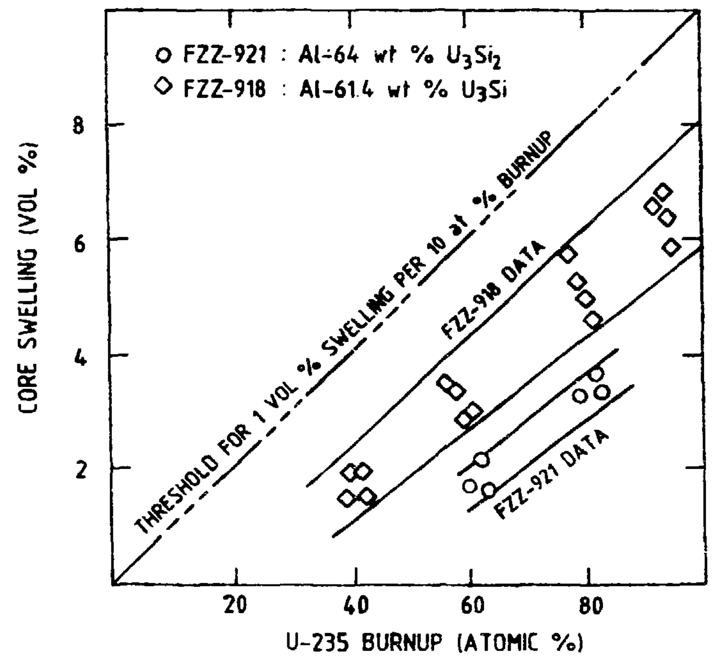

Fig. 4 Swelling of LEU fuel core containing Al-61 wt% U_3Si and Al-64 wt% U_3Si_2.