# Conversion of research reactors to low-enrichment uranium fuels

by R.G. Muranaka*

There are at present approximately 350 research reactors in 52 countries ranging in power from less than 1 watt to 100 Megawatt and over. These reactors are used in every field of science and technology. The principal uses today are in fundamental research, radioisotope production, neutron-activation analysis, and materials testing.

The first research reactors built during the 1940s were low-power devices used primarily for the study of reactor physics and technology of reactors.

In the 1950s and 1960s, low-power research reactors were built around the world using fuel containing uranium enriched to 20% or less of $^{235}U$ by weight, i.e. low-enriched uranium (LEU). This value was chosen because it was considered to be a fully adequate barrier to the usability of $^{235}U$ for nuclear weapons. However, the expanded use of research reactors in fundamental research, materials testing, and radioisotope production created a demand for higher specific power and a need for greater $^{235}U$ concentrations, and led to the substitution of highly enriched uranium (HEU) reaching at least 70% – or even 90% – in place of the LEU fuel previously utilized. HEU fuel also had other benefits: it was more economical, the fuel could be used longer in the reactor core, and it had a higher specific reactivity. HEU became readily available and was used not only for high-power reactors but also for low-power reactors where low-enriched fuel would have sufficed.

In the 1970s, however, many people again became concerned about the possibility that some fuels and fuel cycles could provide an easy route to the acquisition of nuclear weapons. Since enrichment to less than 20% is internationally recognized as a fully adequate barrier to weapons usability, certain Member States have moved to minimize the international trade in highly enriched uranium and have established programmes to develop the technical means to help convert reactors to the use of low-enrichment fuels with minimum penalties. This could involve modifications in the design of the reactor and development of new fuels. As a result of these programmes, it is expected that most reactors can be converted to the use of low-enriched fuel.

* Mr Muranaka is a staff member in the Physics Section of the Agency's Division of Research and Laboratories.

The amount of plutonium in spent fuel has also caused concern, especially when the fuel has very low enrichment or is irradiated in reactors operating at high power. Both the plutonium produced and the level of enrichment of uranium have to be considered in the overall assessment of the proliferation potential of a particular reactor. For example, in a typical 2 MW reactor using 20% enriched uranium, a fuel element on discharge from the core is expected to contain about 10 grams of a plutonium. With reactors using much lower-enriched fuel the plutonium content may be 3 to 4 times greater.

### Conversion criteria

In assessing the practicability of converting existing research reactors to the use of lower-enriched fuel, three factors have to be taken into account:

- The safety margins and reliability of the fuel should not be lower than for the current design based on highly enriched fuel.
- Major modifications to the reactor should not be required.
- There should be no more than marginal loss of reactor performance (e.g. flux-per-unit power), nor increase in operation costs.

The feasibility of converting to reduced-enrichment use must be individually assessed for each reactor. There are specific applications of research reactors in basic physics studies, materials testing, or production of some isotopes which require a high flux of neutrons and this can only be met by using high-enrichment fuel.

Simply substituting lower-enriched uranium in existing fuel designs would reduce core performance and cannot meet the above criteria. Core reactivity is decreased; $^{235}U$ burn-up capability is decreased and fuel costs are increased, and/or core size must be increased, and therefore flux-per-unit power performance is decreased.

It is possible to reduce the enrichment of the fuel for most designs of research and test reactor as long as the amount of $^{235}U$ in the fuel element can be kept approximately the same, despite the decreased enrichment. In fact, it is necessary to increase the $^{235}U$ content of the fuel, to compensate for the loss of reactivity due to the increased absorption of neutrons by the $^{238}U$ present.

Matching $^{235}U$ content (i.e. maintaining the same $^{235}U$ weight in the LEU and HEU fuel elements) would result in in-core flux-per-unit power comparable to that of the unmodified reactor, but, because of the neutron-absorbing effect of $^{238}U$, would generally result in lower reactivity and reduced burn-up potential of the $^{235}U$ atoms. Burn-up potential can be matched to that of the unmodified reactor by increasing the $^{235}U$ content in the reduced-enrichment core by 12 to 15% over that of the 93% enriched case although this then decreases the in-core thermal neutron flux-per-unit power. The importance of these flux effects depends on the particular reactor, the uses to which it is put and the conversion scheme adopted. For example, the decrease in thermal flux in the reflector surrounding the core and in flux traps is generally much smaller than in-core. Costs could be reduced by increasing the fuel cycle length by further increasing the $^{235}U$ content.

To increase the amount of uranium in each fuel element one can either increase the volume of the part of fuel element occupied by uranium, or one can increase the amount of uranium packed into the available volume.

Increasing the space within the fuel element containing uranium may require redesign of the fuel element. Three options are open: decreasing the thickness of the cladding; decreasing the volume of the channel through which the coolant flows; or increasing the thickness of the fuel plate and thereby decreasing the number of plates per element. One of the functions of the fuel cladding is to retain fission products within the fuel element and prevent them being dispersed throughout the coolant. The thickness of the cladding cannot be reduced below the minimum needed for fission product retention. The volume left in the fuel element for the coolant to flow through cannot be reduced too much either if an excessive pressure drop across the core is to be avoided and if the neutron flux in the core is to be adequately moderated. The reduction in the number of plates may be limited by the minimum heat-transfer surface needed to prevent undesirable boiling at a given reactor power.

These limitations may make it difficult to significantly increase the uranium-containing volume within the fuel element in some high-performance reactors that are designed to operate close to their thermal-hydraulic limit. In most operating research and test reactors, however, and especially in the low-power ones, this volume can be increased above current values.

Increasing the amount of uranium packed into the fuel element without changing its thickness has only negligible effects on the thermal-hydraulic properties of the core, and therefore does not normally require redesign of the fuel element. (Only in some very rare cases might it be desirable to increase the coolant volume to moderate the neutron spectrum, hardened by the increased uranium content). The only limitation to this approach is posed by the highest uranium density feasible with the most advanced fuel fabrication technology. This approach can be immediately applied to all those research and test reactors in which the uranium density in the fuel is less than current technology allows. If the approach is to be applied in reactors which already use the most advanced fuel fabrication technology then new fabrication techniques yielding even greater uranium densities will have to be developed. Argentina, Canada, France, the Federal Republic of Germany, Japan, the USA, and the UK are developing such new fabrication techniques. But it will take several years to achieve the desired fuel properties.

For Triga $U\text{-}ZrH_x$ fuel (see box), enrichment reduction is achieved by an increase in the uranium concentration in $UZrH_x$ alloy. The geometry of the fuel element remains identical to the highly enriched version replaced.

**The primary fuel materials currently used in research reactors can be characterized as follows:**

Alloy of uranium-aluminium (U-Al) with uranium densities up to 1.1 g/cm$^3$ but often less than 1 g/cm$^3$;

Dispersion of uranium-aluminide powder in aluminium matrix ($UAl_x$-Al) with densities up to 1.7 g/cm$^3$;

Dispersion of uranium oxide powder in aluminium matrix ($U_3O_8$-Al) with densities up to 1.7 g/cm$^3$;

Alloy of uranium and zirconium hydride ($U\text{-}ZrH_x$) with densities up to 1.3 g/cm$^3$ – commonly referred to as *Triga reactor fuel;*

Uranium oxide ($UO_2$) fuel with densities up to 9.1 g/cm$^3$.

New fuel material under development is uranium silicide ($U_3Si$) in bulk or dispersed in aluminium ($U_3Si$-Al). Development to 12 and 8 g/cm$^3$ respectively is anticipated.

**Fuel development**

The main objective of these fuel development programmes is to develop existing and new fuels for research and test reactor to their maximum feasible uranium loading with the aim of improving the performance of reduced-enrichment reactors. The development effort has been divided into two areas; the extension of currently utilized fuels to their maximum uranium loading; and the development of new high-density fuels.

The extent to which it might be possible to increase the density of uranium and reduce the enrichment of current plate-type fuels is summarized in the Table on page 20. It is expected that U-Al alloy fuel which contains 1.6 to 1.9 g U/cm$^3$ can be developed; however, this development will only allow low-power reactors to be converted to low-enrichment fuel. Aluminide and $U_3O_8$ dispersion fuels could reach uranium densities

**Uranium density and enrichment — reduction potential of candidate fuels for research and test reactors with plate-type fuels**

| Fuel-type | Current uranium loading ($g/cm^3$) | Short-term uranium loading ($g/cm^3$) | Long-term uranium loading ($g/cm^3$) | Current/short-term/long-term enrichment reduction potential (%) | | |
|---|---|---|---|---|---|---|
| | | | | Low-power reactors | High-power reactors | Very high-power reactors |
| U-Al alloy | 1.1 | 1.3 | ~ 1.6 | < 20 | 70/45/45 | 93 |
| $UAl_x$-Al | 1.7 | 2.2 to 2.6 | 2.6 to 2.8 | < 20 | 45/20/20 | 93/45/45 |
| $U_3O_8$-Al | 1.7 | 2.2 to 3.3 | 3.3 to 3.8 | < 20 | 45/20/20 | 93/45/45 |
| $UO_2$ | 9.1$^+$ | – | – | < 20 | < 20 | < 20$^{++}$ |
| $U_3Si$-Al | – | 4.2 to 6.0 | 7.0 to 8.0 | < 20 | 93/20/20 | 93/45/20 |
| $U_3Si$ (bulk) | – | – | ~ 11 | < 20 | 93/93/20 | 93/93/20 |

$^+$ 8.7 if the zircaloy spacers are smeared within the fuel meat. The density of the $UO_2$ is 10.3 $g/cm^3$.
$^{++}$ For very high-power reactors, $UO_2$ would have to be fabricated in very thin sections to provide proper heat removal.

as high as 2.8 and 3.8 $g/cm^3$, respectively, which would make it possible to use low-enrichment fuel in high-power reactors. This assumes that fuels containing 60% by volume of the dispersed uranium are possible. If, as is more likely, only 50% dispersion is possible, then the aluminide dispersion might not permit the conversion of high-power reactors without some changes to the fuel thickness and fuel element geometry. This may be true for $U_3O_8$ dispersions as well. There is little or no high burn-up experience with these extensions of currently utilized fuels; however, much positive experience exists for lower uranium loadings. This suggests that, if uniform dispersion of highly loaded fuel can be successfully fabricated, there is a strong likelihood that they will exhibit satisfactory irradiation behaviour.

Some limited irradiation experience exists for the new high-density fuels. As far as uranium loading is concerned, these materials are more than adequate, and the existing technology of dispersing uranium in the fuel can be utilized. However, the compatibility of the fuel compound with the matrix must be assessed. Uranium silicide ($U_3Si$) for example, must be stabilized so that its slow in-core reaction with aluminium does not cause large volume increases. Such dispersions with 50 or more volume per cent of dispersed fuel would allow conversion of even the highest-power research reactors.

It is likely that $U_3Si$ can be utilized as a dispersant in aluminium. Its increase in volume upon reaction to form $UAl_3$ can be reduced or slowed by alloy addition. As shown in the Table, $U_3Si$ provides a much higher uranium loading than do equal volumes of $U_3O_8$ or $UAl_X$. Adding an alloy would not greatly reduce the uranium loading. The use of aluminium dispersion would mean that present technology could be utilized, rather than some advanced technology which might require cladding with zircaloy or other stiffer cladding material. However, the great density difference between $U_3Si$ and aluminium would require greater care in the blending of powders to maintain homogeneous distribution of fuel particles.

### Safety analysis revisions and licensing

The use of new fuel elements in a research reactor would require that the reactor's current safety analysis report be revised to assess the new balance of safety factors. The amount of work needed will depend on the particular features of the reactor, on the changes caused by the conversion, on the details of the existing safety analysis document, and on the requirements of the licensing authority. Principal issues will include the effect of changes in enrichment and fuel technology on temperature and void coefficients of reactivity, thermal-hydraulic safety criteria, fission product retention, and control system effectiveness. Also, the plutonium build-up in the fuel elements is an issue in safety, safeguards, and licensing and must be included in the revision of the safety analysis. The primary responsibility for the safety analysis report must rest with the reactor organizations. The IAEA is preparing a new guidebook to address the issues of safety and licensing raised by core conversion.

### Effects on utilization

It is important that the effect of the conversion on the planned utilization of the reactor be fully evaluated. For instance, for each individual conversion it is important to assess how planned reactor programmes for irradiation, isotope production and neutron-beam research, may be affected. In this manner trade-offs may be identified, and the conversion can be designed to match existing plans in the best possible way.

**Results of calculations**

A consultant's group from IAEA Member States has been meeting regularly since 1978 to discuss the technical and safety issues of core conversion. The technical issues are discussed in the guidebook IAEA-TECDOC-233 *Research reactor core conversion from the use of highly enriched uranium to the use of low-enriched uranium fuels*, Vienna (1980).

In summarizing the results of calculations performed by this consultants' group, the following conclusions can be drawn:

- Conversion of some reactors from 93% enriched fuel to 45% enriched fuel can be readily achieved without changes in fuel-element design simply by substituting new fuel manufactured using current fuel fabrication technology. Calculations predict very modest flux changes in the experimental regions, making such a conversion entirely feasible.

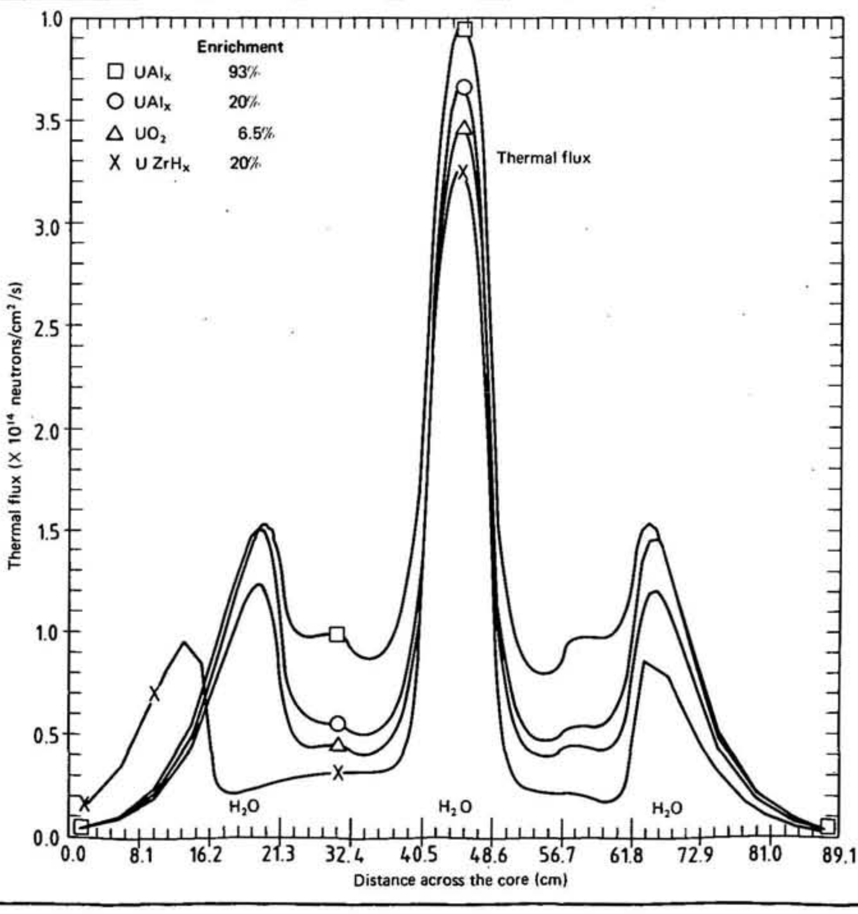
- With a uranium density in the fuel of 2.83 to 2.91 g/cm$^3$ the reactor could be converted to 20% enrichment. The length of the fuel cycle would be equal to that of the highly enriched reactor without modification of the fuel thickness, or fuel element geometry. Thus, the thermo-hydraulic conditions would be essentially unaltered.

The mass of $^{235}$U would be increased by 15% to 18%. Therefore, the control-rod effectiveness would be reduced. On the other hand, the reactivity variation during burn-up would also be smaller. Thus, no modification in the control system should be necessary.

The flux of thermal neutrons in the fuelled regions is also reduced by about the same percentage as the mass of $^{235}$U is increased. However, in the reflector region next to the fuel, the thermal flux recovers rapidly to the value in the HEU case. The flux is only few per cent lower in a typical beam-hole or irradiation position inside or outside the core (see Figure).

- The reactor could be converted to 20% enrichment with the same length of the fuel cycle but with lower uranium density requirements when the fuel thickness is increased. This can be done by reducing the coolant channel width and/or the number of fuel plates in the elements, provided thermal-hydraulic conditions allow such reduction.

It is likely that the development of new fuels now underway will provide the technical means of reducing enrichment in research and test reactors without seriously degrading their performance. Extensions of currently utilized fuels should allow the conversion of low- and medium-power reactors and the new high-density fuels should allow the conversion of higher-power research reactors.




**Comparison of thermal fluxes between the HEU reference core and cases with different LEU fuels.**