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# Two Methods for Converting a Heavy-Water Research Reactor to Use Low-Enriched-Uranium Fuel to Improve Proliferation Resistance After Startup

R. S. Kemp[a]

[a] Department of Nuclear Science and Engineering, Massachusetts Institute of Technology
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# Two Methods for Converting a Heavy-Water Research Reactor to Use Low-Enriched-Uranium Fuel to Improve Proliferation Resistance After Startup

R.S. KEMP*

*Department of Nuclear Science and Engineering, Massachusetts Institute of Technology*

**Abstract:** This article demonstrates the feasibility of converting a heavy-water research reactor from natural to low-enriched uranium in order to slow the production of weapon-usable plutonium, even if the core cannot be physically reconfigured. The analysis was performed for Iran's IR-40 reactor at Arak in support of negotiations with Iran, but the methods have application to future reactors that present similar nonproliferation challenges. Two methods are considered, and both retain identical power, thermal-hydraulic, and safety profiles as the original reactor design. The conversion options can be implemented at any time during the reactor's life. The two methods have competing effects on achievable burnup, and they can be combined to produce an optimized core that matches both the fresh-core reactivity and maximum burnup of the original reactor. For the IR-40 example, the optimized design produces weapon-grade plutonium at only about 19% of the rate of the unmodified reactor for the same power level. Additionally, a reactor so converted could not be readily converted back to natural-uranium fuel without replacement heavy water, and it would retain the ability to produce medical isotopes at rates that exceed the original design through the use of LEU targets.

**Keywords:** Iran, Arak, nonproliferation, plutonium, LEU

## 1. Policy Context

Natural-uranium fueled, heavy-water moderated reactors produce plutonium quickly and have been used to support nuclear-weapon programs throughout history. Converting such a reactor to use low-enriched uranium (LEU) fuel reduces the rate of plutonium production. This happens because enriched fuels have a lower concentration of U-238, the isotope that is transmuted into plutonium by the reactor's neutrons. In February 2014, Iran's IR-40 heavy-water research reactor at Arak became the focus of nonproliferation concern. Iran initially rejected a proposal for converting the reactor to LEU on the basis that the design was committed and conversion to a new design was no longer possible. This article, initially a white paper produced in support of negotiations with Iran, demonstrated that it would be possible to convert the reactor even if the reactor could not be physically reconfigured. Iran subsequently agreed to conversion and has produced its own modified design.

*Address correspondence to: R.S. Kemp, 77 Massachusetts Ave., Cambridge, MA 02139, USA. Email: rsk@mit.edu

This study follows ground-breaking work by Thomas Mo Willig, Cecilia Futsaether, and Halvor Kippe, who performed the first study on converting the IR-40 reactor.[1] The authors suggested increasing the enrichment and power density of the core, but maintaining identical reactor power at 40 MWt, by reducing the number of active fuel tubes from 150 to 60. This modification increases the power density in the fuel region by a factor of 2.5. To maintain an unchanged temperature profile in the fuel, it would require the mass-flow rate of the coolant to increase. If the fuel geometry remained unchanged, the increased cooling requirement would require increasing the flow velocity by the same factor of 2.5. The pressure drop, under normal turbulent flow, is proportional to the velocity squared. The result is that the cooling system for the "Willig" modification would require cooling pumps with more than 6 times the pumping power planned for the original IR-40 design. It might also require larger pipework, heavier control systems, different pressurizes, and different safety strategies.

An alternative approach would be to use the Willig core design, or a similar such design, but reduce the power to no more than 16 MWt so as not to increase the demand on the cooling system. In many ways, this approach might be regarded as the best conversion strategy, as it avoids major changes of infrastructure but provides benefits in the way of increased neutron flux in target regions. However, it still requires modifications to the infrastructure in the core—specifically, the plugging of unused fuel tubes and possibly the introduction of new channels. A study

of options along these lines was undertaken in parallel to this work.[2] Such a strategy is most viable if implemented before the reactor became critical; once the reactor operates, the core infrastructure becomes radioactive and modifications become difficult. This risks creating a rationale for refusing to convert and an incentive to bring the reactor into operation early in order to quiet political demands for conversion. This study showed that a strategy of avoidance would be futile, as the methods proposed herein could be implemented at any time without any changes to the physical infrastructure.

## 2. Constraints on the Conversion of the IR-40 Reactor

The conversion of the IR-40 reactor is subject to both political and technical constraints. Politically, conversion is more likely to be accepted by Iran if it requires few design changes and maximally uses existing investments. Additionally, the international community has made clear that it prefers Iran to keep its domestic enrichment levels to less than 5% U-235. For this reason, this study will assume that the LEU fuel to be used in a modified IR-40 reactor will be enriched to 5 percent. It is worth noting that this might not be the optimal level. There is, in principle, a balance to be struck between the proliferation potential of the plutonium and uranium aspects of Iran's program, and the optimal enrichment level may be something else. Given the difficulty of trying to optimize multiple technical constraints through diplomatic talks, however, 5% is a reasonable working presumption.

The primary technical constraint is that the conversion process must result in a safely operating reactor. The use of LEU will increase reactivity, and it must be safely controlled. This study assumes that the fresh core reactivity should equal that of the unmodified core. A second consideration—albeit more of an economic factor—is that the core life should not be significantly compromised. Again, conversion to LEU is deemed possible if the modified design can provide the same core life as the original design. It is this principle of equivalency that establishes feasibility, not the quantitative parameters of any particular design.

A further constraint is that the reactor must be capable of fulfilling whatever scientific requirements it has been designed to execute. For example, the reactor may need to produce medical isotopes at a certain rate, and that will necessitate some minimum neutron flux in the channels in which isotope-producing targets are to be irradiated. One of the consequences of converting from natural to low-enriched uranium fuel is, for the same geometric buckling, the neutron flux will go down at any given power density. However, this can be compensated for by raising the enrichment of the isotope-producing targets. If the original reactor was designed for natural-uranium targets, then the use of LEU targets can compensate for the lower neutron flux, leaving medical-isotope production rates unchanged or improved.[3,*]

Finally, the reactor is subject to thermal-hydraulic constraints. These include the performance of the primary cooling loop during full-power operation as well as the safety systems designed to

remove decay heat in the event the cooling system fails. Changes to the power distribution within the core can have a significant impact on these systems. This study considers only conversion options that preserve or reduce the cooling requirements of the reactor.

## 3. Methods of Conversion

The purpose of conversion is to reduce plutonium production, which requires replacing natural uranium with LEU. The higher fraction of fissile U-235 in LEU must be compensated for to keep reactivity within manageable bounds. There are two basic methods for doing this that leave the core geometry unchanged.

1. *Reduce the density of fissile material* in the core to approximately what would have been present if the reactor were fueled with natural uranium. Such a reduction can be effected by using a dispersion fuel, in which uranium dioxide is dispersed within a neutronically inactive filler, such as pure aluminum. Dispersion fuels are commonplace in research reactors, and Iran's Tehran Research Reactor already uses dispersion fuel. The method of compensating for changes in enrichment by changing the fissile-material density is also common, and it has been the mainstay of the RERTR (Reduced Enrichment for Research and Test Reactors) program, which has converted over 40 reactors to use LEU fuel worldwide.[**]

2. *Absorb neutrons to control excess reactivity* arising from the increased enrichment. Because LEU introduces extra fissile material relative to natural uranium, the extra reactivity could instead be absorbed by neutron poisons, bringing the fresh-core reactivity in line with the maximum allowed value. This is routinely done by adding burnable poisons such as boron or gadolinium. In the case of the IR-40 reactor, it is interesting to consider an unusual but effective non-burnable poison: a small amount of chemically identical (and therefore perfectly compatible) light water added to the heavy water used as coolant and moderator. Light water acts as a poison because the protium-hydrogen absorbs neutrons. Unlike burnable poisons, however, light water will not extend the core life—generally considered to be a negative. However, light water has the advantage of making it difficult to convert the reactor back to natural-uranium fuel without first performing the time-consuming and highly visible step of flushing the moderator and replacing it with a fresh batch of high-purity heavy water. Furthermore, if heavy-water production is terminated, converting the reactor back may be infeasible. These factors could be significant nonproliferation advantages.

Both of the above methods will enable the reactor to use LEU fuel. Both can be implemented at any time during the reactor's life, without any changes to hardware, and for very low cost. It remains for this study to show that the two conversion methods

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*Iran had no recourse to enriched uranium when the reactor was originally designed an previously published a study on using natural-uranium targets for medical isotope production.

**Normally, the RERTR program is reducing enrichment from high-enriched uranium or HEU (defined as uranium containing more than 20% $^{235}$U) to low-enriched uranium or LEU (less than 20% $^{235}$U) by increasing the density; in the case of IR-40 reactor, the conversion goes the other way.

will produce reasonable technical outcomes, specifically (1) that the reactivity can be brought within manageable bounds through practicable aluminum-dispersion or light-water dilution levels; (2) that the impact of conversion on core lifetime can be made comparable to the core life of the original reactor; and (3) that the result of conversion will reduce plutonium production rates under a proliferation scenario.
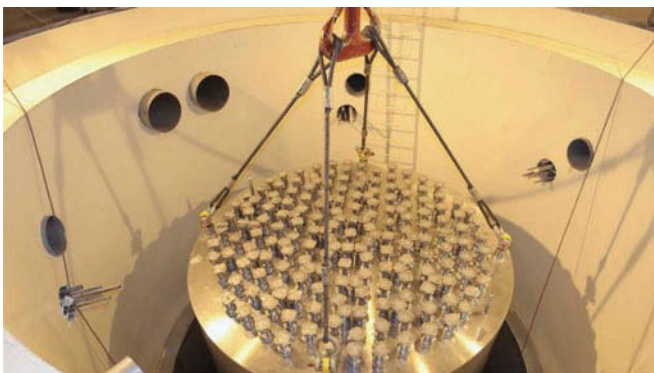
## 4. Design of the IR-40 Reactor

This analysis seeks to understand the impact of the proposed conversion strategies in general terms. Nonetheless, parts of the analysis will require certain design-specific assumptions to be made, specifically with regard to the fuel design and the size of the core. For this purpose, the parameters or Iran's IR-40 reactor are used.

According to data compiled by Willig and an Iranian reactor-safety analysis,[4] the IR-40 is a tank-type reactor (see Fig. 1) with pressure tubes, using heavy-water coolant and moderator, and 150 RBMK-1000-style 19-pin fuel bundles in a hexagonal lattice with a bundle-to-bundle pitch of 26.5 cm.[***] The reactor's design parameters are given in Table 1. In addition to these provided data, this analysis assumes that the pressure tubes and major structural elements of the RBMK fuel bundles are made of Zircaloy-2. It is also assumed that the center tube, which forms a key structural element of the fuel bundle, has the same inside and outside dimensions as the fuel-cladding tubes and is filled with coolant.

## 5. Methods of Analysis

To demonstrate the feasibility and utility of the two conversion concepts, the following analyses are performed.



**Fig. 1.** Photograph of tank of the IR-40 reactor showing fixed pressure tubes and dimensional ratios consistent with data collected by Willig. Photo credit: Yasaman Hashemi, PressTV, Iran.

***RBMK or *Reaktor Bolshoy Moshchnosti Kanalnyy* is a class of Russian power reactors.

**Table 1.** Key parameters of the IR-40 reactor.

| | |
|---|---|
| Power | 40 MWt |
| Coolant and moderator composition | $D_2O$ (99.75%) |
| Coolant and moderator inlet pressure | 3.5 bar |
| Coolant and moderator inlet pressure | 3.5 bar |
| Coolant average inlet temperature | 50°C |
| Coolant average exit temperature | 70°C |
| Pressure-tubes with fuel | 150 |
| Pressure-tube lattice pitch | 26.5 cm |
| Pressure-tube inner radius | 4.0 cm |
| Pressure-tube wall thickness | 0.4 cm |
| Fuel composition and density | 10.4 g $(UO_2)/cm^3$ |
| Fuel outer radius | 0.5740 cm |
| Clad inner radius | 0.5965 cm |
| Clad outer radius | 0.6815 cm |
| Clad and assembly material | Zircaloy |
| Fueled rods per bundle | 18 |
| Empty (water filled) rods per bundle | 1 (center) |
| Active fuel length | 340 cm |
| Approximate core radius | 166 cm |
| Power density in fuel (calculated) | 4.58 W/g(U) |
| Temperature of fuel (estimated) | 100°C |

### 5.1 Excess Reactivity Control in the Fresh Core

The fresh core, without neutron poisons, has the greatest level of reactivity of all possible core configurations for a given fuel type. While it is likely that higher levels of fresh-core reactivity might be desired and safely managed through the use of better control rods or burnable poisons, estimating just how much acceptable reactivity is allowed is a matter subject to debate. Therefore, to establish feasibility in general, the conservative assumption is made that if the reactivity can be made to match the original design, then reactivity cannot be considered an impediment to conversion. A search of the parameter spaces for the two conversion options (adjusting fissile-material density for method 1; and the $H_2O:D_2O$ ratio for method 2) until solutions that gave the same fresh-core $k_{eff}$ as the original IR-40 design was undertaken, and results compared to the original design until a configuration that yielded identical $k_{eff}$ was found (at 0.1 MWd/kg, equivalent to a fresh core with xenon poison equilibrium).

The fuel effects for all configurations, both the converted and original design, were computed using a lattice physics code, which gives $k_\infty$ and the migration length $M^2$ in an infinite lattice. The $k_{eff}$ for a finite reactor can then be estimated by computing the probability of non-leakage for a finite core with diffusion theory, and the resulting values compared to the value for the original core. Any recourse to diffusion theory necessitates that the medium should be weakly absorbing (scattering dominates the interactions), that the flux should depend only weakly on the angular direction of the neutrons, and that the reactor be reasonably homogenous from the perspective of neutrons. These conditions hold for IR-40-like reactors. Specifically, heavy water is less neutron absorbing than light water; scattering off of deuterium in heavy water is more isotropic than scattering off light hydrogen; and the neutron mean free paths are large compared to the fuel geometry but small enough that leakage at the boundary is small. Although small errors in the computed probability of

non-leakage will still result, primarily from geometry considerations, those errors will be nearly the same for both the modified and unmodified core. Since feasibility depends only on showing that the two fresh-core $k_{eff}$ can be made to match, the effect of these errors will tend to cancel. Finally, it is important to bear in mind that both the fuel composition and the finite-core geometry are essentially approximations. Ultimately, greater flexibility is possible with LEU, and reactivity and cycle length can be better managed using fuel shuffling and burnable poisons. The purpose, however, is not to find a precise or optimal solution but to establish feasibility even under stringent limitations.

### 5.2 Effect of Conversion on Core Life

The core life, or maximum achievable burnup and the intimately related cycle length, are ultimately design considerations driven by external factors such as fuel economics, the reactor's operating profile, and indigenous technology limitations. Nevertheless, claims about core life requirements can be laid to rest if is possible to establish that similar burnup can be obtained even after conversion. The challenge arises because the different materials used for the modified core will alter the rate at which the reactor loses reactivity. The impact of this rate of reactivity loss can be readily observed in the infinite lattice. The true effect, however, requires a finite core analysis. Conveniently, however, the lattice results show that the two conversion strategies have opposite effects on achievable burnup. This implies the existence of a hybrid conversion strategy—a blend of fuel density reduction and moderator dilution—that will provide the same burnup as the original core. As such, feasibility can be established without solving a finite core burnup model, which would normally need to incorporate a larger set of assumptions about external constraints than this general analysis is able to make.

### 5.3 Effect of Conversion on Plutonium Production

Conversion is interesting only if the converted reactor, in some configuration approximating the performance objectives of the original reactor, produces considerably less plutonium. Plutonium production is a matter of per-fuel bundle exposures. This analysis makes the assumption that all fuel bundles can be burned to the same extent through reshuffling, which is reasonable for a tank-type research reactor. It then becomes possible to study the plutonium vector as a function of bundle burnup by using a lattice physics code. Both the plutonium production rate and the isotopics of the plutonium are relevant considerations and jointly evaluated.

## 6. Computation of $k_\infty$, $M^2$, and $k_{eff}$

To compute $k_\infty$ and $M^2$, it is necessary to model accurately the geometry of the lattice described in Table 1, and to take into account effects such as parasitic neutron capture in fission products and resonance self-shielding—effects that go beyond what is possible with diffusion theory. We do this with CASMO-4E, a lattice physics code that solves the transport equation using the method of characteristics technique. The code was developed for power-reactor designers by Studsvik Scandpower, a division
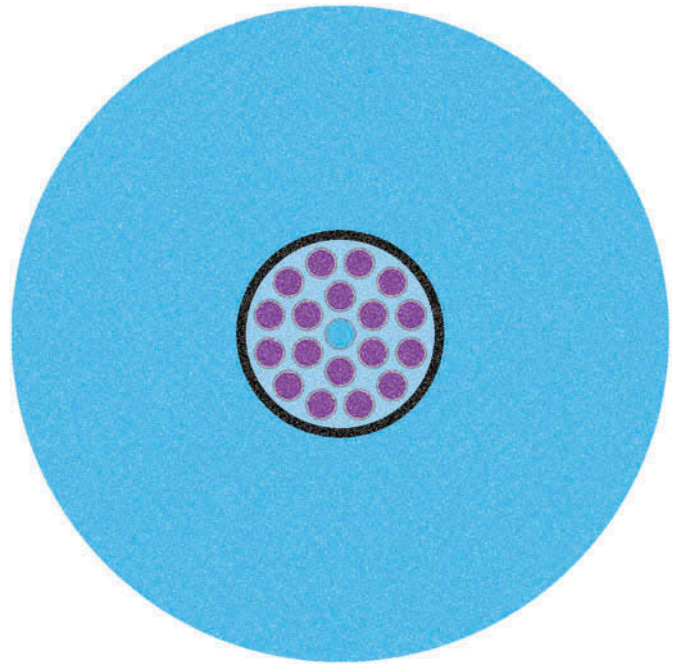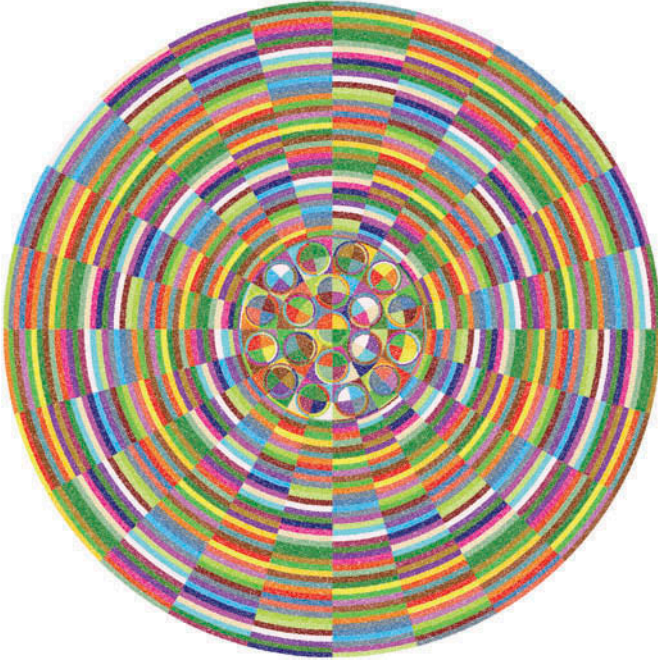


**Fig. 2.** Tracks for a single fuel bundle and associated moderator used for the characteristics ray-tracing solution produced by CASMO-4E. Colors relate to different materials. The 18-pin fuel bundle (purple), cladding (gray), and pressure tube (black) are clearly visible at the center of coolant and moderator (blue). The track density provide an indication of how the transport solution sees the geometry. That the tracks appear homogeneously distributed and dense (the white speckles are the residual areas not covered by tracks) indicates that the geometry well sampled.

of Studsvik AB, Nykoping, Sweden. CASMO version 4E is designed to model hexagonal lattices like that of the IR-40 by using a "white" boundary condition in which neutrons crossing the boundary are assumed to be isotropically instead of spectrally reflected.[5] The model used here is therefore a single bundle with associated coolant and moderator, as shown in Figure 2, surrounded by a white boundary, which is why the model geometry is circular rather than hexagonal. CASMO breaks the geometry into a fine mesh, shown in Figure 3, and solves the neutron-transport equations across the mesh for 70 energy groups using ENDFB-VI cross-section libraries under the assumption that the neutron source is spatially flat and isotropic inside each mesh element. CASMO also computes burn-up isotopics, tallies the corresponding plutonium production and isotope vector, and can model extensive core physics such as a Doppler broadening and thermal expansion in materials.

The result of this model is that the IR-40 infinite lattice has a $k_\infty = 1.128$ and $M^2 = 409\,cm^2$ at the fresh-core condition after equilibrium xenon poisoning (determined at 0.1 MWd/kg burnup). Detailed results at different burners are shown in Table 2.

### 6.1 Calculation of $k_{eff}$ by Diffusion Theory

We use diffusion theory to convert $k_\infty$ of the infinite lattice to $k_{eff}$ for a finite core. Diffusion theory can be thought as the

relationship between the net neutron current (vector flux) **J** and the scalar flux-density $\phi$:

$$\mathbf{J}(r,t) = -D(r)\nabla\phi(r,t) \tag{2}$$

Where $D$ is the diffusion coefficient. In the steady-state condition, $\partial N/\partial t = 0$, so equation 1 can be rewritten as simply the balance of sources, sinks, and leakage:

$$0 = \nabla \cdot [D(r)\nabla\phi(r)] + S(r) - s(r) \tag{3}$$

Where the sources are $S$, the sinks $s$, arise from nuclear reactions in the volume in proportion to the flux density $\phi$ and the macroscopic interaction cross-sections $\Sigma$, which is typically weighted over a group of energies represented by $\phi$ and all the materials within the reactor. Writing the macroscopic absorption cross-section as $\Sigma_a$ and the macroscopic fission cross-section as $\Sigma_f$, and assuming that neutrons are produced only by fission, gives

$$S(r) = \frac{1}{k_{\text{eff}}}\nu\Sigma_f(r)\phi(r) \tag{4}$$

$$s(r) = \Sigma_a(r)\phi(r) \tag{5}$$

where $\nu$ is average number of neutrons (prompt and delayed) produced by a fission event, and $k_{\text{eff}}$ is the effective neutron multiplication between generations. Substituting $S$ and $s$ into equation 3, assuming core materials are spatially uniform, making use of the geometrical buckling $B^2 = [-\nabla^2\phi(r)]/\phi(r)$, canceling $\phi(r)$, and solving for $k_{\text{eff}}$ gives

$$k_{\text{eff}} = \frac{\nu\Sigma_f}{DB^2 + \Sigma_a} \tag{6}$$

Dividing through by $\Sigma_a$ lets $k_{\text{eff}}$ be written in terms of $k_\infty = \nu\Sigma_f/\Sigma_a$ and the migration area $M^2 = D/\Sigma_a$, both of which were found numerically using CASMO in section 6.

$$k_{\text{eff}} = \frac{k_\infty}{1 + M^2B^2} \tag{7}$$

Equation 7 gives $k_{\text{eff}}$ from $k_\infty$ by multiplication with the term $1/(1 + M^2B^2)$, which is called the probability of non-leakage, and it is the same for one or multiple energy groups. The calculation of $M^2$ differs for different energy groups, but since $M^2$ is obtained from CASMO's numerical simulation, the approximation here is primarily that of the geometric buckling, $B^2$, which remains unchanged across all cases.[****]



**Fig. 3.** The same tracks shown in Figure 2, but with the color of each track changing at the boundary between computational mesh cells. Within each mesh cell, the neutron sources are assumed to be spatially flat and isotropic.

**Table 2.** CASMO output for the unmodified IR-40 infinite lattice.

| Burnup MWd/kg | $k_\infty$ | $M^2$ cm² | U-235 w% | Fissile Pu w% | Total Pu w% |
|---|---|---|---|---|---|
| 0.00 | 1.144 | 404 | 0.720 | 0.000 | 0.000 |
| 0.10 | 1.113 | 397 | 0.708 | 0.007 | 0.007 |
| 0.50 | 1.113 | 340 | 0.664 | 0.038 | 0.039 |
| 1.00 | 1.112 | 391 | 0.612 | 0.072 | 0.076 |
| 2.00 | 1.101 | 387 | 0.521 | 0.126 | 0.139 |
| 3.00 | 1.084 | 384 | 0.443 | 0.167 | 0.192 |
| 4.00 | 1.065 | 382 | 0.375 | 0.199 | 0.237 |
| 5.00 | 1.044 | 382 | 0.316 | 0.223 | 0.278 |
| 6.00 | 1.022 | 381 | 0.266 | 0.242 | 0.313 |
| 7.00 | 1.001 | 381 | 0.222 | 0.257 | 0.345 |
| 8.00 | 0.980 | 381 | 0.185 | 0.268 | 0.374 |
| 9.00 | 0.961 | 380 | 0.153 | 0.277 | 0.401 |
| 10.0 | 0.943 | 380 | 0.126 | 0.285 | 0.425 |

steady-state balance between production, consumption, and leakage of neutrons for a fixed-size reactor of a homogenous material. For a given energy group, the continuity equation for an arbitrary volume is:

$$\partial N(r,t)/\partial t = -\nabla \cdot \mathbf{J}(r,t) + S(r,t) - s(r,t) \tag{1}$$

where $N$ is the neutron density in an infinitesimal volume, $\nabla \cdot \mathbf{J}$ the net neutron current across the surface enclosing that volume, $S$ is the sum of all sources internal to that volume, and $s$ is the sum of all internal sinks in that volume. Fick's law gives the

****The geometrical buckling $B^2$ for a right cylinder radius $R$ and height $H$ is:

$$B_{\text{cyl}}^2 = \left(\frac{J_{0,1}}{R}\right)^2 + \left(\frac{\pi}{H}\right)^2$$

where $J_{0,1} = 2.4048$ is the first root of the zeroth Bessel function of the first kind. To the radius and height are usually added a small extrapolation length that helps the flux profile better fit the real distribution inside the reactor.

## 7. Results

### 7.1 *Method 1: Dispersion of Fissile Material in an Aluminum Matrix*

The process described in section 6 was iterated until the aluminum-LEUO$_2$ ratio was found that produced the same $k_{eff}$ as the unmodified natural-uranium core. The solution occurs at 0.526 g/cm$^3$ uranium (heavy metal at 5% U-235) dispersed in 2.31 g/cm$^3$ aluminum.

The lattice burnup at this mix ratio is given in Table 3. Compared to the results for the unmodified lattice in shown in Table 2, the data show the core lifetime would be improved as $k_\infty$ remains higher for any given burnup.

### 7.2 *Method 2: Dilution of Heavy-water Moderator with Light Water*

The process described in section 6 was iterated until the amount of light water that, when added to heavy water, produces the same $k_{eff}$ as the unmodified natural-uranium-reactor was found. The solution occurs at 81.5 atom-percent heavy water.[†]

The lattice burnup at this dilution ratio is given in Table 4. Compared to the results for the unmodified lattice in shown in Table 2, that data show that cycle length would be compromised as $k_\infty$ remains drops more rapidly for a given burnup.

### 7.3 *Optimization: A Combined Dispersion-Dilution Conversion Strategy*

The above results demonstrate that it is technically feasible to convert the IR-40 reactor to LEU fuel by either of the two methods proposed. It is notable that the use of an LEU dispersion fuel extends the core life relative to the unmodified reactor (for the same initial $k_{eff}$), whereas the light-water dilution method severely shortens core life but offers additional nonproliferation advantages of making conversion more difficult to reverse. The two methods can be combined to achieve the same initial reactivity and same core life as the original design, while producing less plutonium and achieving the extra nonproliferation advantages afforded by the use of light water. For the infinite lattice, this occurs at approximately 95.8 atom-percent heavy water and 1.01 g/cm$^3$ uranium (heavy metal at 5% U-235) dispersed in 2.18 g/cm$^3$ aluminum. Detailed results are given in Table 5.

## 8. Non-Proliferation Consequences of Conversion

Using the plutonium tallies in the CASMO outputs given in Tables 2–5, one can estimate the plutonium production rates and plutonium-isotope vectors of the various core designs. Two immediate effects are observable: the plutonium production rates

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[†]The addition of approximately 20% light water will reduce coolant viscosity and density by less than 2%, a change that is likely too small to have a detrimental effect on any cooling system or safety profile and leaves the thermal hydraulics of the reactor effectively unchanged.

**Table 3.** CASMO output for the 5% enriched aluminum-dispersion fuel infinite lattice.

| Burnup MWd/kg | $k_\infty$ | $M^2$ cm$^2$ | U-235 w% | Fissile Pu w% | Total Pu w% |
|---|---|---|---|---|---|
| 0.00 | 1.246 | 706 | 4.94 | 0.000 | 0.000 |
| 0.10 | 1.210 | 693 | 4.93 | 0.000 | 0.000 |
| 0.50 | 1.200 | 693 | 4.88 | 0.004 | 0.004 |
| 1.00 | 1.193 | 694 | 4.82 | 0.009 | 0.010 |
| 2.00 | 1.181 | 699 | 4.70 | 0.022 | 0.022 |
| 3.00 | 1.170 | 704 | 4.58 | 0.034 | 0.035 |
| 4.00 | 1.158 | 709 | 4.46 | 0.046 | 0.047 |
| 5.00 | 1.147 | 715 | 4.34 | 0.057 | 0.059 |
| 6.00 | 1.135 | 720 | 4.22 | 0.068 | 0.071 |
| 7.00 | 1.124 | 726 | 4.10 | 0.079 | 0.083 |
| 8.00 | 1.111 | 732 | 3.98 | 0.090 | 0.095 |
| 9.00 | 1.099 | 738 | 3.86 | 0.100 | 0.106 |
| 10.0 | 1.086 | 745 | 3.75 | 0.109 | 0.118 |

**Table 4.** CASMO output for the light-water diluted IR-40 infinite lattice.

| Burnup MWd/kg | $k_\infty$ | $M^2$ cm$^2$ | U-235 w% | Fissile Pu w% | Total Pu w% |
|---|---|---|---|---|---|
| 0.00 | 1.047 | 121 | 5.00 | 0.000 | 0.000 |
| 0.10 | 1.033 | 121 | 4.99 | 0.002 | 0.002 |
| 0.50 | 1.027 | 121 | 4.94 | 0.010 | 0.010 |
| 1.00 | 1.022 | 121 | 4.88 | 0.019 | 0.019 |
| 2.00 | 1.016 | 121 | 4.76 | 0.038 | 0.039 |
| 3.00 | 1.010 | 121 | 4.65 | 0.056 | 0.057 |
| 4.00 | 1.004 | 121 | 4.53 | 0.073 | 0.075 |
| 5.00 | 0.998 | 121 | 4.41 | 0.089 | 0.093 |
| 6.00 | 0.992 | 121 | 4.30 | 0.105 | 0.110 |
| 7.00 | 0.985 | 121 | 4.19 | 0.120 | 0.127 |
| 8.00 | 0.979 | 121 | 4.07 | 0.134 | 0.144 |
| 9.00 | 0.972 | 121 | 3.96 | 0.148 | 0.160 |
| 10.0 | 0.966 | 121 | 3.85 | 0.161 | 0.175 |

**Table 5.** CASMO output for reactivity and core-life matched IR-40 infinite lattice with extra proliferation resistance.

| Burnup MWd/kg | $k_\infty$ | $M^2$ cm$^2$ | U-235 w% | Fissile Pu w% | Total Pu w% |
|---|---|---|---|---|---|
| 0.00 | 1.114 | 367 | 4.94 | 0.000 | 0.000 |
| 0.10 | 1.086 | 363 | 4.92 | 0.000 | 0.000 |
| 0.50 | 1.078 | 363 | 4.87 | 0.005 | 0.005 |
| 1.00 | 1.071 | 363 | 4.81 | 0.012 | 0.012 |
| 2.00 | 1.061 | 365 | 4.69 | 0.025 | 0.025 |
| 3.00 | 1.051 | 366 | 4.57 | 0.038 | 0.039 |
| 4.00 | 1.041 | 368 | 4.45 | 0.050 | 0.052 |
| 5.00 | 1.030 | 370 | 4.33 | 0.062 | 0.064 |
| 6.00 | 1.020 | 372 | 4.21 | 0.074 | 0.077 |
| 7.00 | 1.009 | 374 | 4.10 | 0.085 | 0.089 |
| 8.00 | 0.998 | 376 | 3.98 | 0.095 | 0.101 |
| 9.00 | 0.987 | 378 | 3.86 | 0.106 | 0.113 |
| 10.0 | 0.975 | 380 | 3.75 | 0.116 | 0.125 |

**Table 6.** Weapon-grade plutonium production in the IR-40 infinite lattice.

| Configuration | Burnup Target for 7% Pu-240 [MWd/kg] | Total Pu at Target Burnup [w%] | WgPu Rate at 40MWt [gPu/day] | Relative Production Rate |
|---|---|---|---|---|
| Unmodified | 1.55 | 0.103 | 26.6 | 1 |
| Optimized | 9.86 | 0.124 | 5.04 | 0.19 |
| Moderator Dilution Only | 8.64 | 0.155 | 7.20 | 0.27 |
| Dispersion Fuel Only | 10.3 | 0.121 | 4.72 | 0.18 |

in the LEU cores are much lower, but the isotopic vectors in the LEU reactors retain a more weapon-favorable fraction of fissile isotopes at higher burnups. This means there is a tension between overall plutonium production rates and the weapon usability of the plutonium.[††]

Weapon-grade plutonium is defined as plutonium having less than 7 weight-percent Pu-240.[‡] Plutonium production rates and isotope vector trends were fitted to CASMO data to deduce the effective rate at which each configuration would produce weapon-grade plutonium. The results show the reduced plutonium-production rate arising from the use of LEU fuel substantially outweighs the negative effect of the improved isotope vector. The maximum rates at which weapon-grade plutonium can be produced by each configuration are shown in Table 6.

Table 6 shows the optimized and dispersion-fuel cores have similar properties in terms of weapon-grade plutonium production potential; however, the combined dispersion-dilution core, being moderated by 95.8 weight-percent heavy water, will not sustain a chain reaction if refueled with natural-uranium fuel ($k_{\text{eff}} = 0.95$ for the infinite lattice without xenon). This small amount of light water in the moderator prevents Iran from converting the reactor back to a more proliferation-capable reactor without first producing a fresh batch of moderator and coolant. Because heavy-water production is visible, slow, and easily monitored, this would not go unnoticed.[‡‡] Better still, Iran's existing heavy-water production facility could be dismantled as a confidence-building measure with no negative impact on Iran's nuclear program (a small reserve of 95.8 weight-percent heavy water could be kept on hand for topping up the IR-40 reactor

as needed). The combined dilution-dispersion strategy provides a meaningful additional barrier to proliferation.

Finally, it is important to note that although plutonium production has been slowed by roughly a factor of five, eventually the accumulated spent fuel will contain a weapon-quantity of plutonium that could be used to make a weapon quickly. It will, therefore, be necessary to export spent fuel on an occasional basis. The frequency of export depends on the power and duty cycle at which the reactor is operated, but the modifications suggested here will significantly ease the export requirement. For example, if Iran used its 20% enriched uranium as targets for medical isotope production, the reactor could be operated at a continuous average power below 10 MWt and still produce medical isotopes at a rate equal to or above that of the original IR-40 design with natural-uranium targets. Under this arrangement, the reactor would produce plutonium at less than 5% the rate of the original design, and fuel would need to be exported no more often than once every 4 years (assuming a maximum allowed plutonium inventory in spent fuel of 4 kg). In practice, with additional isotope production ongoing at Iran's TRR reactor, the IR-40 reactor might operate at a lower duty cycle, and plutonium production and exports would be proportionally lower and less frequent.

## 9. Conclusions

This article demonstrates the viability of converting the IR-40 reactor at Arak to use low-enriched uranium by two methods that can be effected after the reactor becomes critical and is therefore not sensitive to the timing of the negotiations. By combining the two methods, it is possible to produce a 5% enriched LEU core with the same fresh-core reactivity and core life as the original, unmodified reactor, but which produces weapon-grade plutonium at only 19% the rate of the original reactor design. This optimized core has the additional advantage of preventing Iran from easily converting the reactor back to natural-uranium fuel.

The foregoing analysis was constrained by the assumption that the LEU fuel had to be enriched to 5% U-235. Higher enrichments, if tolerable within the context of the enrichment program, would result in even less plutonium. Conversely, lower enrichments could be used to reduce the threat from the enrichment program at the cost of somewhat greater plutonium production rates. The optimal configuration depends on the size of the enrichment program and policies regarding the accumulated LEU it produces.

This analysis did not consider other reactor of fuel-cycle optimization strategies. For example, core life could be extended by the use of burnable poisons and greater initial fissile loading. Neutron flux across the reactor could be flattened or tailored using different enrichments or U:Al dispersion ratios. To model such effects would require from Iran a clear definition of the reactor's purpose and performance requirements, as well as a finite-reflected core model. These analyses are best performed jointly by Iran and its negotiating partners and beyond the scope of this analysis. Nevertheless, this paper demonstrates the feasibility and tremendous flexibility that exists in converting Iran's existing IR-40 reactor to a more peaceful configuration.

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[††]Modern weapon designs are not neutronically sensitive to plutonium isotopics, although heat from the decay of the higher plutonium isotopes may present a more practical, but not insurmountable, design challenge. The sophistication of a posited Iranian weapon design is not known to the author, but some evidence in the public domain suggests Iran may possess design knowledge that exceeds historical first-generation weapons.[6,7]

[‡]This analysis assumes CASMO's non-fissile plutonium tally (which is the sum of all even numbered isotopes) is effectively equivalent to a Pu-240 tally. The two will be very nearly equal at the low burnups considered here.

[‡‡]The Heavy Water Plant in Iran is rated at 16 tons/year. The Arak reactor needs about 100 tons of heavy water.

## Acknowledgment

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