

**A NEUTRONIC FEASIBILITY STUDY FOR
LEU CONVERSION OF THE IR-8 RESEARCH REACTOR**

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ABSTRACT

Equilibrium fuel cycle comparisons for the IR-8 research reactor were made for HEU(90%), HEU(36%), and LEU (19.75%) fuel assembly (FA) designs using three dimensional multi-group diffusion theory models benchmarked to detailed Monte Carlo models of the reactor. Comparisons were made of changes in reactivity, cycle length, average ^{235}U discharge burnup, thermal neutron flux, and control rod worths for the 90% and 36% enriched IRT-3M fuel assembly and the 19.75% enriched IRT-4M fuel assembly with the same fuel management strategy. The results of these comparisons showed that a uranium density of 3.5 g/cm^3 in the fuel meat would be required in the LEU IRT-4M fuel assembly to match the cycle length of the HEU(90%) IRT-3M FA and an LEU density of 3.7 g/cm^3 is needed to match the cycle length of the HEU(36%) IRT-3M FA.

INTRODUCTION

The IR-8 reactor, located at the Russian Research Center "Kurchatov Institute" in Moscow, has utilized IRT-3M FA containing HEU(90%) since 1981.¹ An IRT-3M FA with HEU(36%) is also available, but has not been used. The main objective of the reactor is to provide a high thermal neutron flux density in the large beryllium reflector. The purpose of these calculations is to compare the performance of the reactor with IRT-3M FA containing HEU(90%) and HEU(36%) fuels and to determine the uranium densities with LEU(19.75%) fuels in IRT-4M FA that would be needed to match the cycle length of the present IRT-3M HEU(90%) fuel and, potentially, the IRT-3M HEU(36%) fuel. All of the FA were modeled in the IR-8 reactor core using the same fuel management strategy. The calculations also predicted the relative thermal neutron fluxes in selected irradiation positions after equilibrium core conditions were reached. Control rod worths were calculated for the key cases.

CORE AND FUEL ASSEMBLY DESCRIPTIONS

The active core consists of 16 IRT-3M six-tube FA arranged in a 4x4 array. The core has a large 30 cm Be reflector on all radial sides as shown in Fig. 1 (reproduced here from Ref. 1). The central hole of the four corner FA, which is used for sample irradiations, was assumed to be water filled. The remaining 12 FA each have control rods located in the center. Each control rod consists of a B_4C absorber section followed by an aluminum (SAV-1) displacer which is present in the core when the absorber is withdrawn. The core has 12 beam tubes positioned along the core mid-plane in the stationary Be reflector.

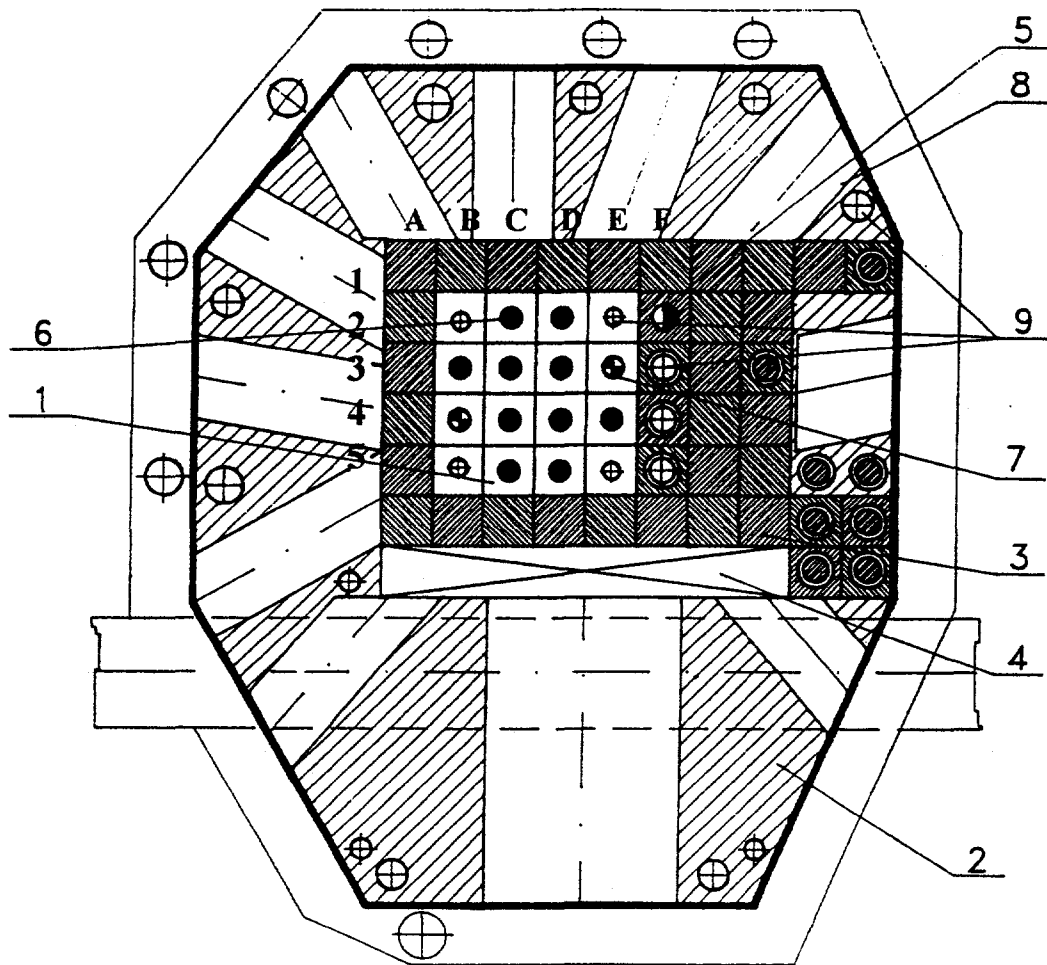


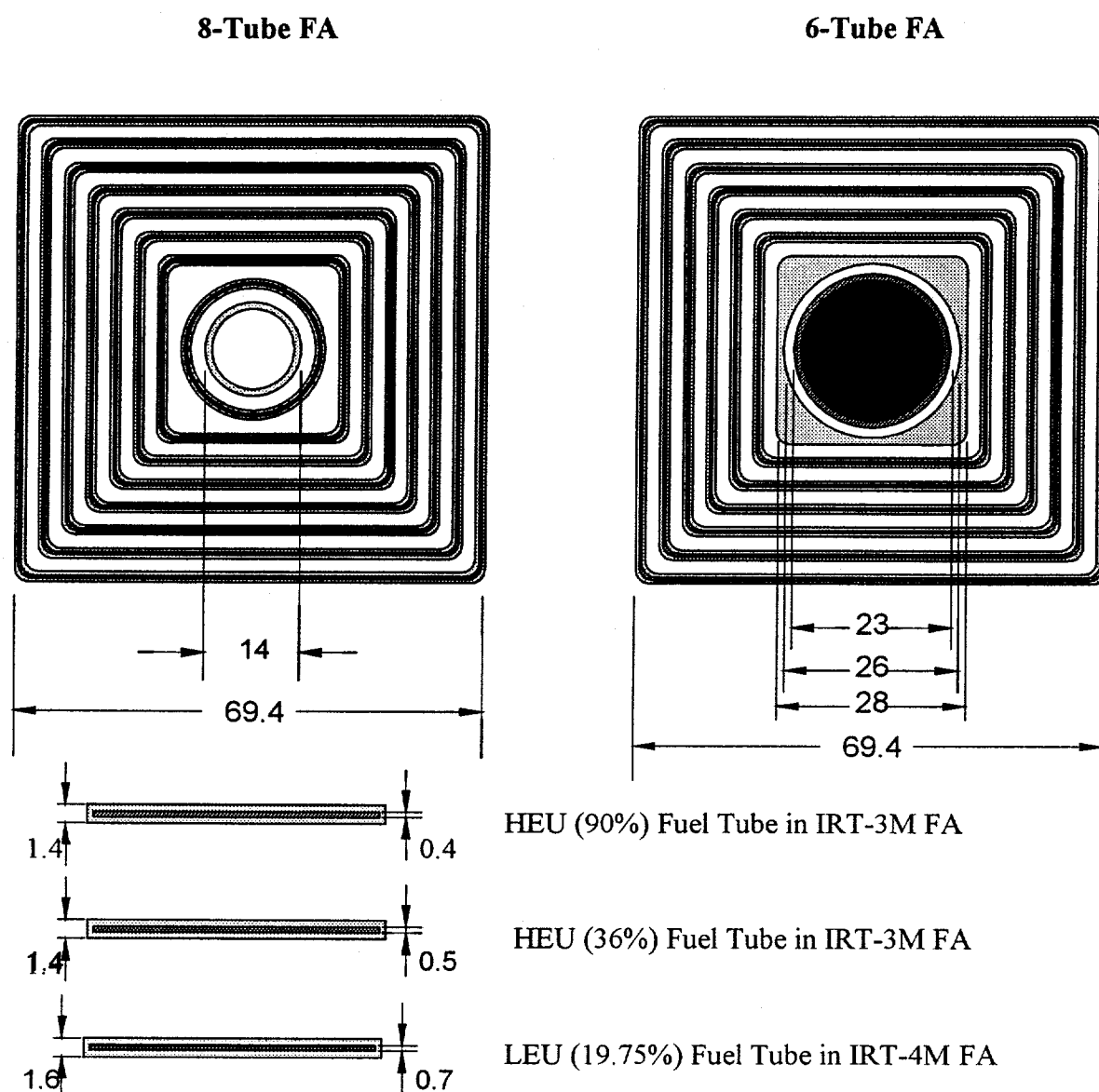
Figure 1. Load of the IR-8 Reactor Core.

1 - 6-tube FA; 2 - Blocks of stationary beryllium reflector; 3 - Removable beryllium block; 4 - Lead shield; 5 - Channel with automatic regulating rod of CPS; 6 - Channel with shim-safety rod of CPS; 7 - Channel with safety rod of CPS; 8 - Beam tube; 9 -

Vertical experimental channel

The reactor currently uses the IRT-3M FA with 90% enriched uranium and a U-235 as-built loading of 272 g per 6-tube FA. A radial slice through the active fuel zone is shown in Fig. 2 for the 8-tube FA and companion 6-tube FA. All fuel tubes have rounded corners with an inner tube radius of 2.80 mm. The uranium density in the fuel meat of the HEU(90%) FA is 1.1 g/cm³. The water channel thickness between fuel tubes is 2.05 mm. The fuel tubes are 1.4 mm thick with a 0.4 mm thick fuel meat region. The 36% enriched IRT-3M FA maintains the same control rod specifications and fuel tube dimensions, except that the fuel meat thickness is 0.5 mm and the clad thickness is 0.45 mm. The uranium density in the 36% enriched IRT-3M FA is 2.51 g/cm³ which results in a loading of 309 g ²³⁵U/ 6-tube FA.

Fig. 2 IRT-3M Fuel Assembly Cross Section



The IRT-4M FA design has the same number of fuel tubes as the IRT-3M FA but the fuel tube thickness is increased² from 1.4 mm to 1.6 mm. The fuel meat thickness is 0.7 mm and the clad thickness is 0.45 mm. The coolant channel between fuel tubes is reduced to 1.85 mm. Calculations have been performed for LEU ²³⁵U loadings of 330g (3.48 gU/cm³) and 352g (3.71 gU/cm³) with UO₂-Al dispersion fuel.

REACTOR CORE NEUTRONICS MODEL DESCRIPTION

The reactor core and ex-core materials were modeled using XYZ multi-group diffusion theory and continuous energy Monte Carlo methods. The Monte Carlo reactor model was constructed using MCNP³ with an ENDF-B/VI cross section library. A detailed geometrical model of each core component was made in the MCNP model, except in the axial reflector where some materials were homogenized. Results of calculations with MCNP models using fresh fuel were used to check REBUS3⁴ diffusion theory results. A reactivity difference was computed between the two models and applied to all REBUS3 fuel cycle results.

All REBUS3 fresh core and fuel cycle burnup models assume that the core is symmetrical about the core midplane. The neutron cross sections for the core materials were generated using the WIMS-ANL code^{5,6} and a library with 69 energy groups based on ENDF-B/VI data and collapsed to seven broad energy groups for use in REBUS3. The annular model of the FA in WIMS-ANL was based upon each concentric square tube volume having rounded corners. Each fuel tube was depleted based upon its unique neutron spectrum in the WIMS-ANL FA model. Macroscopic cross sections for the two-group two-dimensional IRT-2D/PC⁷ code used by RRC(KI) were calculated using the URAN-AM code⁸.

The seven broad group microscopic cross sections were polynomial fitted as a function of burnup for use in REBUS3. The REBUS3 fuel depletion chains included production of five Pu isotopes, ²⁴¹Am, and ²³⁷Np. Each FA was modeled using four axial depletion zones over the 29 cm core half-height. The radial model of each FA was divided into a central control follower or water hole and guide tube surrounded by a homogenized fuel-clad-coolant zone. All removable Be blocks have water holes and water gaps modeled in separate regions. Beryllium poisoning or beam tubes have not been included in REBUS3 models.

FRESH CORE COMPARISONS OF EXCESS REACTIVITY AND FLUXES

Comparisons of fresh core reactivity and fluxes in two experiment positions for each FA design and enrichment loaded into the 16 FA IR-8 core are presented in Table 1. Three different neutronic models were used to calculate the fresh core reactivities. The first method by ANL used the MCNP model, the second method by ANL used a REBUS3 diffusion theory model, and the third method by the Russian Research Centre "Kurchatov Institute" used the diffusion theory code IRT-2D/PC. Both the REBUS3 and IRT-2D models overpredict the reactivity of fresh core loaded with HEU(90%) IRT-3M FA by 1.0 % $\Delta k/k$ compared to the MCNP model. The REBUS3 models of HEU(36%) and LEU(19.75%) fresh core reactivities were overpredicted by about 2.1% $\Delta k/k$.

The peak midplane thermal fluxes were calculated at the centers of the water holes in position B-5, a corner FA, and position F-3, adjacent to the core in a beryllium reflector block with a central water hole irradiation channel. The thermal fluxes are presented as the product of the core k_{eff} and the computed flux in Table 1 and Fig. 3. This product adjusts fluxes for a critical core condition that can be compared with measured fluxes. The MCNP flux standard deviations are 2.3% in location B-5 and 1.8% in location F3. The REBUS3 diffusion theory model results predict peak thermal fluxes (<0.625 eV.) from 4 to 10% less than the MCNP model for all cores. The peak midplane thermal flux in the Be block water channel was reduced by 6% in the 36% enriched core compared to the HEU core. The LEU core peak thermal fluxes at the center of the water channel in position F-3 were reduced 1 to 3% depending upon the uranium density. The LEU peak thermal fluxes were 15% lower in core position B-5 compared to HEU(90%) core fluxes.

Table 1

IR-8 Fresh Fuel Core Reactivity and Peak Thermal Flux Comparisons

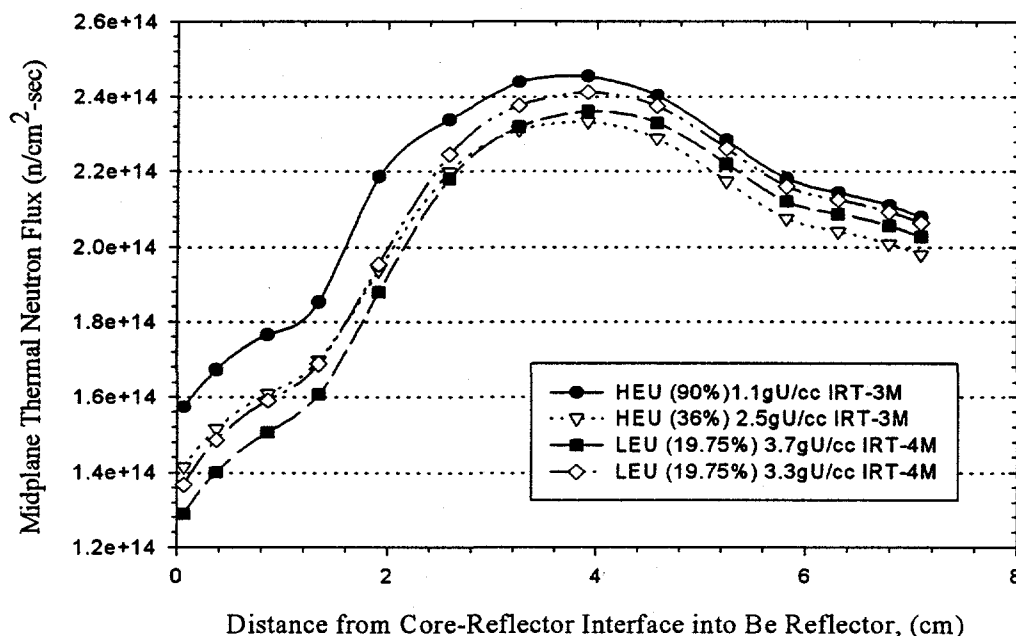
Neutronics Model	Fuel Assembly Design	Fuel Enrichment (wt %)	^{235}U -Loading 6-Tube (g/FA)	Uranium Density in Meat (g/cm^3)	Excess Reactivity (% $\Delta k/k$)	Reactivity Bias (% $\Delta k/k$)	Peak ¹ Thermal Flux in F-3 ($\text{n}/\text{cm}^2\text{-sec}$)	Peak ¹ Thermal Flux in B-5 ($\text{n}/\text{cm}^2\text{-sec}$)
MCNP ²	IRT-3M	90	272	1.1	20.57	0	2.54E+14	1.65E+14
REBUS3	IRT-3M	90	272	1.1	21.61	1.05	2.46E+14	1.53E+14
RRC(KI)	IRT-3M	90	272	1.1	21.60	1.03		
MCNP	IRT-3M	36	309	2.51	18.91	0	2.39E+14	1.47E+14
REBUS3	IRT-3M	36	309	2.51	21.00	2.09	2.33E+14	1.41E+14
RRC(KI)	IRT-3M	36	309	2.51	19.90	0.99		
MCNP	IRT-4M	19.75	330	3.48	15.71	0	2.52E+14	1.38E+14
REBUS3	IRT-4M	19.75	330	3.48	17.84	2.13	2.39E+14	1.33E+14
RRC(KI)	IRT-4M	19.75	330	3.48	21.30	5.59		
MCNP	IRT-4M	19.75	352	3.71	16.13	0	2.46E+14	1.40E+14
REBUS3	IRT-4M	19.75	352	3.71	18.28	2.15	2.36E+14	1.30E+14
RRC(KI)	IRT-4M	19.75	352	3.71	21.80	5.67		

¹ Peak Fluxes are reported as the product of the computed flux and k_{eff}

² MCNP fluxes have standard deviations of $\pm 2.3\%$ in B-5 and $\pm 1.8\%$ in F-3.

Figure 3

IR-8 Fresh Core REBUS3 Thermal Fluxes in Be
Reflector Position F-3 at Core Midplane for
HEU and LEU Cores



REACTOR BURNUP MODEL

The reactor burnup model in the REBUS3 code was an equilibrium flux solution in which two fresh 6-tube assemblies were loaded at the beginning of each cycle and discharged after remaining in the core for eight cycles. The fuel management strategy⁹ used in these analyses was to load the lowest burnup fuel into the central regions of the core and gradually move the fuel to the corner positions of the core before discharge. This in/out fuel management strategy increases the core reactivity performance compared with an out/in strategy by reducing the total core neutron leakage.

Since no absorber sections (B_4C) of the control rods were inserted during the depletion, the axial flux shape was assumed to be symmetric about the core midplane and only a half-height flux solution was required. The 29 cm half-height was divided into four axial depletion zones. The remainder of the axial height was modeled with a water and aluminum (SAV-1) mixture. There was a total of 171 Be reflector blocks and a Pb shield which occupied eight reflector block positions. No beam tubes were represented in this model. The end of equilibrium cycle (EOEC) excess reactivities were assumed to be about 2% $\Delta k/k$ to account for reserve excess reactivity, the absence of beam tubes, beryllium poisoning, and other reactivity losses. A no-return-current boundary condition was assigned to all exterior surfaces of the reactor model at the outer

boundary of the beryllium reflector in the radial direction and 60 cm above the core midplane in the axial direction.

EQUILIBRIUM CORE BURNUP RESULTS

A summary of the calculated results is presented in Table 2, including beginning and end of equilibrium cycle excess reactivities, fuel meat uranium densities, and cycle lengths. The same fuel shuffling pattern was used for each fuel assembly design and enrichment. Beginning and end of equilibrium cycle reactivity values include the effects of equilibrium Xe and Sm concentrations and other fission products. The ^{235}U discharge burnups listed in Table 2 are average values for the entire fuel assembly. Peak burnups are approximately 20% higher than the average burnup.

Using the data shown in Table 2, a plot of equilibrium cycle length versus EOEC core excess reactivity for all fuel cycle options is shown in Figure 4. The sensitivity to how changes in cycle length affect EOEC excess reactivity can more readily be recognized by viewing this plot. For an estimated EOEC reactivity of 2% $\Delta k/k$, an LEU density of about 3.5 g/cm³ in the fuel meat is needed to match the cycle length of the HEU(90%) case. This corresponds to a ^{235}U content of 330 g in the 6-tube IRT-4M FA. An LEU density of 3.7 g/cm³, corresponding to a ^{235}U content of 352 g per 6-tube FA, is needed to match the cycle length that would be obtained with the HEU(36%) IRT-3M FA.

Fig. 4. IR-8 Equilibrium Cycle Length vs EOEC Reactivity - Replace IRT-3M HEU(90%) with IRT-3M HEU(36%) or IRT-4M LEU(19.75%) Fuel Assembly

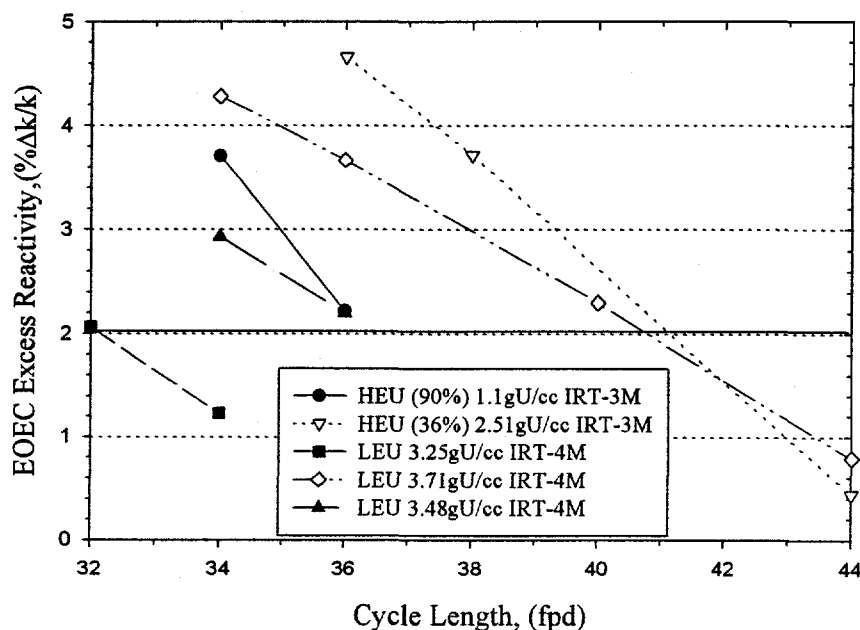


Table 2

IR-8 EQUILIBRIUM FUEL CYCLE COMPARISONS

Fuel Enrichment (wt%)	Fuel Assembly Design	U-235 Loading 6-Tube (g)	Fuel Meat Volume (cm ³)	Uranium Density in Meat (g/cm ³)	Volume Fraction UO ₂ (%)	Cycle Length (fpd) ¹	Ave. U-235 Discharge Burnup (%)	Excess ² Reactivity (% $\Delta k/k$)	BOEC	EOEC
90	IRT-3M	272	274	1.1	12.0	34	62.9	8.07	3.71	
90	IRT-3M	272	274	1.1	12.0	36	66.5	7.08	2.22	
36	IRT-3M	309	343	2.51	27.4	36	56.6	8.15	4.66	
36	IRT-3M	309	343	2.51	27.4	38	59.6	7.52	3.72	
19.75	IRT-4M	309	479	3.25	35.5	32	49.7	5.07	2.06	
19.75	IRT-4M	309	479	3.25	35.5	34	52.6	4.48	1.23	
19.75	IRT-4M	330	479	3.48	38.0	34	49.2	5.85	2.93	
19.75	IRT-4M	330	479	3.48	38.0	36	52.0	5.34	2.20	
19.75	IRT-4M	352	479	3.71	40.6	34	46.4	6.95	4.28	
19.75	IRT-4M	352	479	3.71	40.6	36	48.9	6.52	3.67	
19.75	IRT-4M	352	479	3.71	40.6	40	54.0	5.56	2.30	

¹ fpd = full power days

² All computed reactivities include equilibrium Xe and Sm concentrations

The fuel management scheme follows an in/out pattern with two fresh fuel assemblies loaded into the center of the core at the beginning of each cycle.

CONTROL ROD WORTH COMPARISONS

The MCNP Monte Carlo model at ANL and the IRT-2D diffusion theory models at RRCKI were used to compute the changes in control rod worth as a function of the fuel used in a fresh 16 FA IR-8 core. The results are summarized in Table 3. The worth of the automatic regulating control rod remains unchanged for the three fuel types calculated. The worth of the 10 shim rods and the regulating rod are reduced < 5% for the HEU(36%) FA core relative to the HEU(90%) FA core. The worth of this group of control rods was calculated with all rods inserted 1.3 cm above the bottom of active fuel (ABAF) and then with all rods inserted to 6.0 cm ABAF. Control rod positions were specified in Ref. 9. The worth of the shim rods is unchanged for the LEU(19.75%) FA core relative to the HEU(90%) FA core. The worth of the two safety rods increases by about 6% from 3.4% $\Delta k/k$ in HEU(90%) FA core to 3.6% $\Delta k/k$ in the LEU(19.75%) FA core. The

worth of the two safety rods increases from 3.4% $\Delta k/k$ in HEU(90%) FA core to 5.0% $\Delta k/k$ when the 10 shim rods and the regulating rod are inserted to 10 cm ABAF. This positioning of the shim and regulating rods is very close to the critical rod location when the safety rods are fully withdrawn.

Table 3. IR-8 Reactivity Worth of Control Rods

Inserted Control Rod(s)	Position of Other Control Rods	IRT-3M(90%) (% $\Delta k/k$)	IRT-3M(36%) (% $\Delta k/k$)	IRT-4M(19.75%) (% $\Delta k/k$)
Automatic Regulating Control Rod	10 Shim Rods and Two Safety Rods Out	0.58 ± 0.04 ¹ 0.5 ²	0.56 ± 0.05 ¹ 0.51 ²	0.64 ± 0.05 ¹
10 Shim Rods and Reg. Rod to 1.3 cm ABAF ³	Two Safety Rods Out	23.2 ± 0.04 25.1 ²	22.3 ± 0.08 24.5 ²	23.3 ± 0.06
10 Shim Rods and Reg. Rod to 6 cm ABAF	Two Safety Rods Out	22.1 ± 0.07	21.2 ± 0.08	22.2 ± 0.06
Two Safety Rods to 1.3 cm ABAF	10 Shim Rods and Reg. Rod Out	3.4 ± 0.06	3.3 ± 0.06	3.6 ± 0.05
Two Safety Rods to 1.3 cm ABAF	10 Shim Rods & Reg. Rod Inserted to 10 cm ABAF ⁴	5.0 ± 0.08		

¹ MCNP at ANL with \pm the standard deviation

² IRT-2D/PC at RRC(KI) ⁷⁻⁹

³ ABAF = Above Bottom of Active Fuel

⁴ Very close to the critical position of the 10 shim rods and regulating rod with two safety rods out

CONCLUSIONS

The fuel cycle calculations showed that a uranium density of 3.5 g/cm³ in the UO₂-Al fuel meat would be required in the LEU IRT-4M fuel assembly to match the cycle length of the HEU(90%) IRT-3M FA and an LEU density of 3.7 g/cm³ is needed to match the cycle length of the HEU(36%) IRT-3M FA. The equilibrium cycle length will be increased from about 36 days to about 41 days if the HEU(90%) fuel were replaced with either the HEU(36%) fuel or the LEU(19.75%) fuel with 3.7 gU/cm³. The annual FA consumption rate could be reduced by as much as 14% in both cases. The peak thermal neutron fluxes in the ex-core locations are nearly the same in the LEU cores and the HEU(90%) core. The peak thermal flux in the center of the corner FA with LEU fuel will be reduced by about 15% relative to the HEU(90%) fuel and by about 5% relative to the HEU(36%) fuel. Calculated changes in the reactivity worth of the control rods were not significant after replacement of HEU(90%) fuel with either HEU(36%) or LEU(19.75%) fuel.

REFERENCES

1. Ryazantsev, E.P., Egorenkov, P.M., and Yashin, A.F., "The IR-8 Reactor Operation." First International Topical Meeting of Research Reactor Fuel Management, Bruges, Belgium (February 1997).
2. Aden, V., Research and Development Institute of Power Engineering Report (January 1997).
3. Briesmeister, J.F., "MCNP - A General Monte Carlo N-Particle Transport Code, Version 4A", LA-12625-M (November 1993).
4. Toppel, B.J., "A User's Guide for the REBUS-3 Fuel Cycle Analysis Capability", ANL-83-2 (March 1983).
5. Deen, J.R., Woodruff, W.L., and Costescu, C.I., "WIMS-D4M User Manual, Rev. 2", ANL/RERTR/TM-23 (September 1998).
6. Woodruff, W. L., Argonne National Laboratory, USA and Leopando, L. S., Philippine Research Institute, "Upgrades to the WIMS-ANL Code and Library", (these Proceedings).
7. Nasonov, N. A., Tsipulin, V. N., Karpukhin, A. A., Mitrofanov, V. N., " Calculation Precision of Critical Loads with the IRT-2M type FA (36% enrichment) by IRT-2D/PC code". Preprint AEI-5259/4. M.. 1990.
8. Arkhangelsky, N. V. and Nasonov, N. A., "URAN-AM Code for Reactor Cylinder Cell Neutron Calculation Considering Isotopic Change under Burn-up". Preprint AEI-3861/5, M., 1983.
9. Egorenkov, P. M., private communication.