A NEUTRONIC FEASIBILITY STUDY FOR LEU CONVERSION OF THE WWR-SM RESEARCH REACTOR IN UZBEKISTAN

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DISCLAIMER

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ABSTRACT

The WWR-SM research reactor in Uzbekistan has operated at 10 MW since 1979, using Russian-supplied IRT-3M fuel assemblies containing 90% enriched uranium. Burnup tests of three full-sized IRT-3M FA with 36% enrichment were successfully completed to a burn up of about ~50% in 1987-1989. In August 1998, four IRT-3M FA with 36% enriched uranium were loaded into the core to initiate conversion of the entire core to 36% enriched fuel. This paper presents the results of equilibrium fuel cycle comparisons of the reactor using HEU (90%) and HEU (36%) IRT-3M fuel and compares results with the performance of IRT-4M FA containing LEU (19.75%). The results show that an LEU (19.75%) density of 3.8 g/cm^3 is required to match the cycle length of the HEU (90%) core and an LEU density 3.9 g/cm^3 is needed to match the cycle length of the HEU (36%) core.

INTRODUCTION

The WWR-S research reactor at the Institute of Nuclear Physics of the Uzbek Academy of Sciences is located in Ulugbek, 30 km north-east of Tashkent. The reactor was put into operation at a power level of 2 MW in September 1959. It was reconstructed in 1971-1979 to upgrade the reactor power and enhance the experiment capacity of the reactor. The reactor is designated to carry out experiments in field of nuclear physics and nuclear engineering, neutron activation analysis, solid state physics and isotope production. To perform these experiments, it originally had 9 vertical irradiation channels, a thermal column, and 9 horizontal beam tubes.

REACTOR DESCRIPTION

In course of time, the experimental needs increased, which the reactor could not meet. In September 1971 the reactor was shutdown to be reconstructed. The reconstruction project was developed by Kurchatov's Atomic Energy Institute. It was decided to increase the reactor power to 10 MW and locate 15 additional vertical irradiation channels around the core [1,2]. Metallic beryllium blocks (69x69mm, square cross-section) were installed in the core as a reflector to reduce neutron leakage. The blocks have a central hole (48mm in diameter) for vertical channels making it possible to add 15 more vertical irradiation channels around the core and thus to reach the reconstruction goal. Numerous other changes were made. The heat exchanger was replaced with one with a higher heat removal capacity. The changes in the core and power upgrade required changes in the control and instrumentation system to comply with nuclear safety regulations. The number of control rods was increased to 12. Control rod withdrawals were made to operate in step fashion. The reconstruction was fully completed in 1978. The reactor capacity to provide experiments in loop channels was increased more than 10 times, and for irradiation experiments of materials and isotope production, the capacity was increased 20 times [3]. The reconstructed reactor had the following parameters:

Core dimensions:	cylinder, 640mm diameter, 58mm high.
Critical mass:	2.5 kg U^{235}
Neutron flux: thermal in core:	$2.6x10^{14}$
thermal in reflector	$2.0x10^{13}$
Burn up:	>40%
Average power density in core:	120 kW/l
Average specific power in fuel:	2100 kW/kg U^{235}
Total worth of safety, control and regulating rods:	22.3 %Δk/k

Control, regulating and safety rods: 3 safety rods, 4 group control rods (of 2 rods each)

All control rods contain B_4C absorber in stainless steel cladding. The number of grid locations available to load fuel assemblies is 52. Two core configurations are currently used with either 24 or 28 fuel assemblies. During 1978 –1979, the reactor was operated at a nominal power of 10 MW employing IRT-2M fuel assemblies containing 90% enriched uranium. In 1979 the reactor was loaded with new, advanced IRT-3M type 6-tube and 8-tube fuel assemblies, also with 90% enriched uranium. Since 1979 the reactor has been operated at a nominal power of 10 MW, 21 full power days cycle and one week maintenance. The average operation is 5,000 hours per year over ten months.

IRT-3M AND IRT-4M FUEL ASSEMBLIES.

A cross section of the IRT-3M and IRT-4M fuel assemblies are shown in Fig. 3. The main parameters of the 6-tube FA are described in Table 1. A burnup test of three IRT-3M 36% enriched FA was carried out at the WWR-SM reactor during 1987-1989. Two of the FA had 6-tubes and one had 8-tubes. The fuel assemblies remained in core for a total of 10,300 hours. It was concluded that the FA under test are reliable and operable up to more than 50% burnup [5].

The IRT-3M 36% enriched and 90% enriched fuel assemblies have the same outer dimensions and the same inner diameter of the inner tube. Thus there is no need to change core size or control and safety rods to convert the reactor to 36% enriched FA.

Fig1. WWR-S Vertical Section

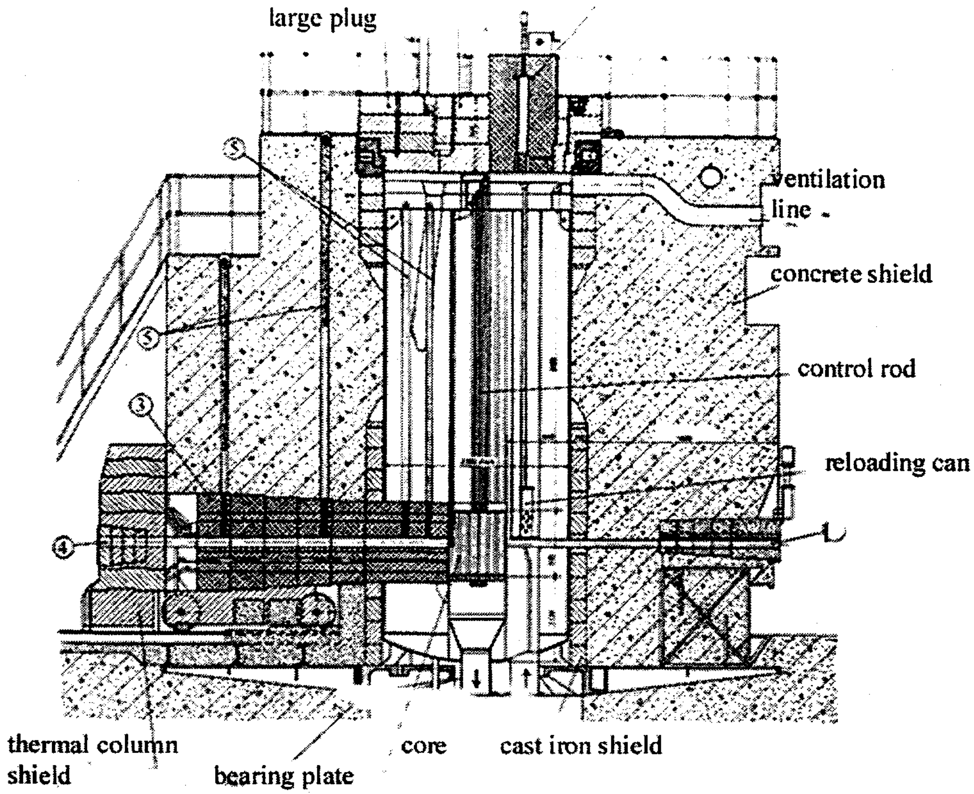


Fig2. WWR-SM Horizontal Section

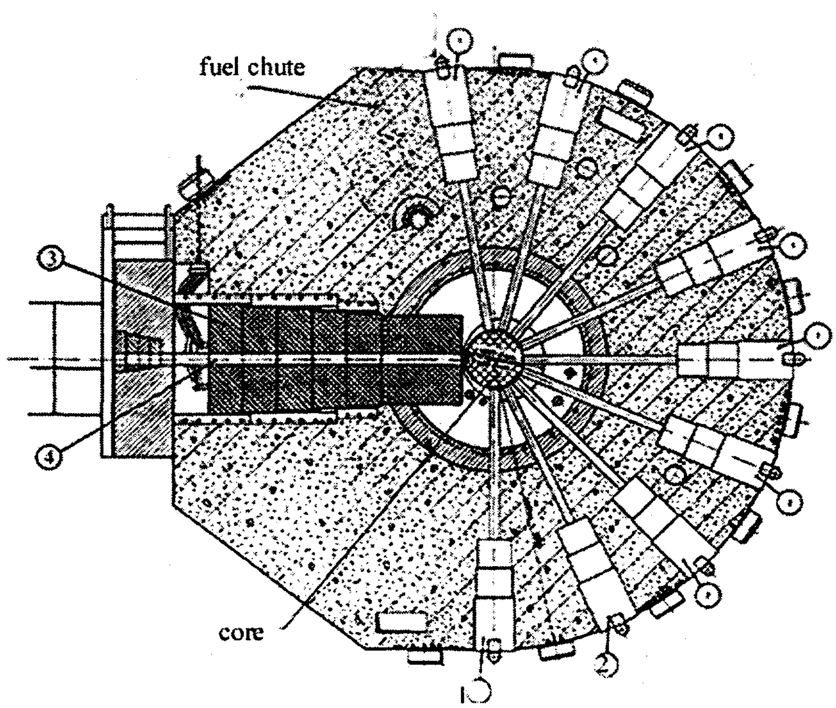


Fig. 3. IRT-3M Fuel Assembly

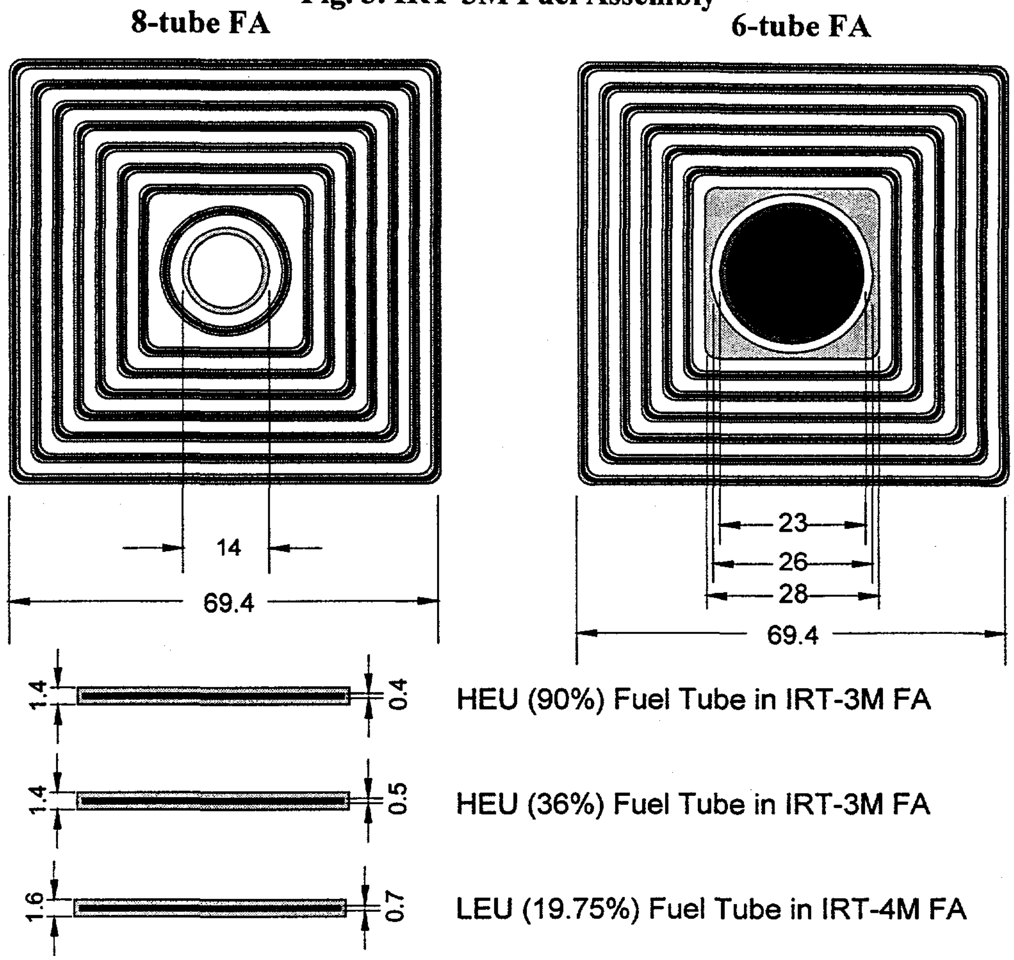


Table 1. Main parameters of the IRT-3M and IRT-4M 6-tube FA [4].

Fuel Assembly	IRT-3M	IRT-3M	IRT-4M
Uranium enrichment, %	90	36	19.75
Thickness of the FE cladding, mm	0.5	0.45	0.45
Thickness of the meat, mm	0.4	0.5	0.7
Water gap between FE, mm	2.05	2.05	1.85
Length of meat in FE, mm	580	580	580
U-235 loading in FA, g	265	309	351.8
Heat transfer surface, m^2	1.37	1.37	1.37
Specific surface of heat transfer, m^2	0.462	0.462	0.462

On this basis, it was decided to convert the WWR-SM research reactor to use IRT-3M fuel assemblies with UO_2-Al fuel meat and 36% enriched uranium. The conversion will be performed gradually by stepwise loading 36% enriched FA. In August 1998, the first four IRT-3M 36% enriched 6-tube fuel assemblies were loaded into the core.

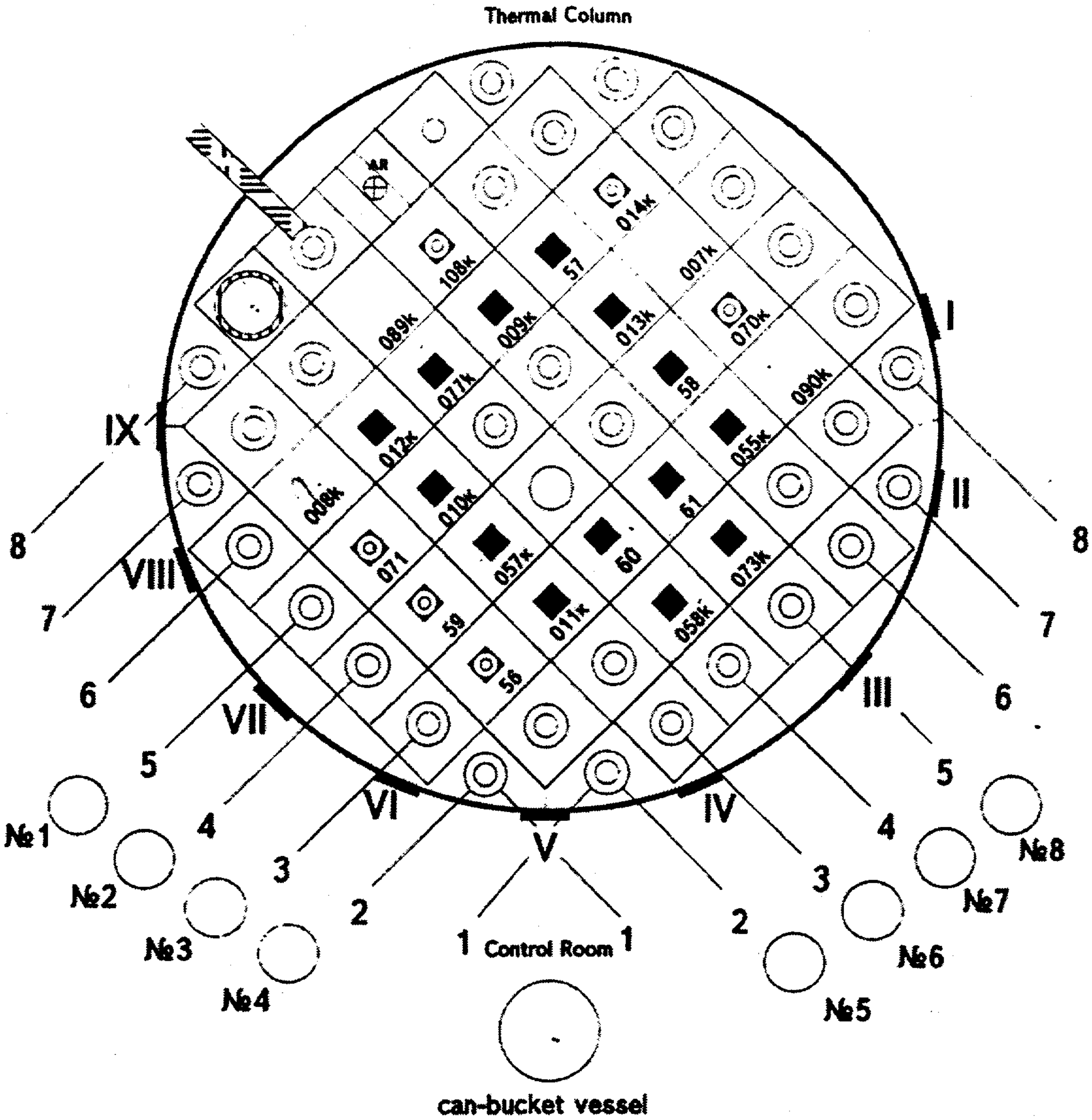
REACTOR NEUTRONIC AND FUEL CYCLE MODEL

Neutronic calculations have been performed for the core configuration of 24 FA shown in Fig. 4. Neutron cross section were generated using the WIMS-ANL [6,7] code and a 69 group library based on ENDF/B-VI nuclear data. Cross sections with 7 broad groups were generated for use in diffusion theory burnup calculations using the REBUS-3 code [8]. The IRT-3M FA has square fuel tubes with rounded corners. For the WIMS-ANL code, this geometry was transformed into an annular model with the same tube and water channel volumes as the real FA. The reference HEU (90%) core contained 20 6-tube FA with 265 g U^{235} and 4 8-tube FA with 300 g U^{235}. Each 6-tube FA remained in the core for 10 cycles. One 8-tube FA was replaced every 24 cycles. Two fresh 6-tube FA were loaded into the core near the center, moved to the opposite corner and then gradually moved to the exterior locations before discharge from the core. The 8-tube FA was loaded into an exterior location and moved around outer core locations until it achieved its end of life. The radial model of FA was represented with either a central control rod follower and guide tube or with a water hole and guide tube surrounded by a homogenized fuel-clad-coolant zone. The axial direction was divided into four depletion zones over the 29 cm core half-height. Beam tubes were not included in the diffusion theory burnup model. However, the reactivity effects have been corrected by comparison with MCNP Monte Carlo models [9], which include all known core components. Approximate beryllium poisoning effects have been incorporated. No experiments or control rods were modeled in the fuel cycle calculations.

BERYLLIUM POISONING EFFECT

Beryllium poisoning effects have been accounted for only approximately in the fuel cycle calculations. The beryllium poison (He^3, Li^6) concentrations were computed [10] assuming that the blocks were in the core for 10 operating cycles and two months maintenance each year for 20 years. The values of the He^3 and Li^6 concentrations were 8.853×10^{-7} and 1.382×10^{-6}, respectively for the beryllium blocks in the center of the core. These poison concentrations reduced the EOEC excess reactivity by 1.5% $\Delta k/k$. An MCNP calculation gave for Be poisoning 1.43% $\Delta k/k$, which is in agreement with the REBUS-3 diffusion theory result. It should be noted that the history of the Be blocks used in these calculations leads to underestimation of the Be poisoning because the blocks were actually out of the core for several prolonged periods that are currently not known and thus were not used in the calculation of the Be poisoning. It is expected that a more detailed Be block history will increase the poisoning effect. The longer the blocks were out of the core, the larger will be the poison concentrations.

Fig. 4. The WWR-SM core configuration



FUEL CYCLE RESULTS

Results of the equilibrium fuel cycle calculations that were performed for IRT-3M type FA with 90% and 36% enriched uranium and for IRT-4M type FA with 19.75% enriched uranium are shown in Table 2. All fuel assemblies contain UO_2-Al fuel meat with different uranium densities.

Table 2. WWR-SM Equilibrium Reactivity Data

Fuel Enrichment (wt%)	Fuel Design	U-235 Loading 6/8 tube (g)	Uranium Density In Meat (g/cm^3)	Volume[1] Fraction UO_2 (%)	Cycle Length (fpd)	Discharge Burnup 6/8 tube (%)	Excess Reactivity[2] (%Δk/k) BOEC	EOEC	Peak[3] Thermal Flux in 26 n/cm^2-s	Peak[3] Thermal Flux in 54 n/cm^2-s
90	IRT-3M	264/300	1.07	12	21	43.1/71.6	5.96	3.85	1.95E+14	3.16E+14
90	IRT-3M	264/300	1.07	12	24	49.5/78.7	4.44	1.87	1.96E+14	3.18E+14
36	IRT-3M	309/351	2.51	27.4	21	35.9/61.6	6.73	5.13	1.85E+14	3.03E+14
36	IRT-3M	309/351	2.51	27.4	24	41.1/68.3	5.75	3.88	1.85E+13	3.04E+14
19.75	IRT-4M	352/400	3.71	40.6	21	30.8/56.0	4.70	3.30	1.82E+14	3.00E+14
19.75	IRT-4M	352/400	3.71	40.6	24	35.1/62.2	3.91	2.30	1.82E+14	3.01E+14
19.75	IRT-4M	399/452	4.20	45.9	21	27.4/50.7	8.12	6.90	1.77E+14	2.93E+14
19.75	IRT-4M	399/452	4.20	45.9	24	31.2/56.6	7.45	6.09	1.77E+14	2.93E+14

[1] $\rho_u = 9.15\ V_f^{UO2}$. The density of UO_2 was 10.4 g/cm^3 and the weight fraction of U in UO_2 is 0.88.

[2] All computed reactivities include equilibrium Xe and Sm concentrations

[3] Fluxes are quoted as k_{eff}*computed flux

The reactivity results showing fuel cycle trends for the various FA designs, enrichment, and ^{235}U loadings are shown in Fig. 5. The first result in Table 2 with the reference 90% enriched IRT-3M FA, a 21 day cycle length, and an average ^{235}U discharge burnup of 43% in the 6-tube FA approximately matches the operating experience of the reactor. The EOEC excess reactivity of 3.85 % Δk/k is larger than experienced, but as stated earlier, the burnup model does not include reactivity losses due to experiments, and the reactivity loss due to beryllium poisoning effects is likely to be larger than presently modeled.

With the 36 % enriched IRT-3M fuel, the equilibrium core would have a cycle length of 24 days with the same EOEC excess reactivity as the HEU (90%) fuel. The average ^{235}U discharge burnup in the 6-tube FA would be about 41%. Peak thermal fluxes in core positions 26 and 54 would be reduced by about 5% relative to the HEU (90%) fuel.

Calculations with IRT-4M LEU fuel were done for uranium densities of 3.7 and 4.2 g/cm^3 in UO_2-Al fuel meat for equilibrium core cycle lengths of 21 days and 24 days. The uranium densities that are needed with LEU fuel to match the 21 day cycle length of the HEU (90%) fuel and the 24 day cycle length of the HEU (36%) fuel can be obtained by interpolating the uranium density and EOEC excess reactivity data shown in Table 2. The 21 day LEU data have a slope of 0.73 % $\Delta k/k$ at EOEC per 0.1 gU/cm^3 ((6.9-3.3)/(4.2-3.71)x10). This means that the 3.85 % $\Delta k/k$ EOEC excess reactivity of the HEU (90%) core can be matched with a uranium density of about 3.8 g/ cm^3 with LEU fuel.

In the same manner, the 24 day cycle length of the HEU (36%) FA can be matched with an LEU density of about 3.9 g/cm^3. Increasing the uranium density by 0.1 g/ cm^3 adds about 10 g ^{235}U to each fuel assembly. Peak thermal fluxes in positions 26 and 54 with 3.8 g/cm^3 LEU fuel would be reduced by about 7-8 % relative to the HEU (90%) fuel and by about 2% relative to the HEU (36%) fuel. Peak thermal flux profiles within irradiation position 54 are shown in Figure 6.

Fig. 5. WWR-SM Equilibrium Cycle Length vs EOEC Reactivity for Replacement IRT-3M (90%) with either IRT-3M (36%) or IRT-4M (19.75%) Fuel Assemblies.

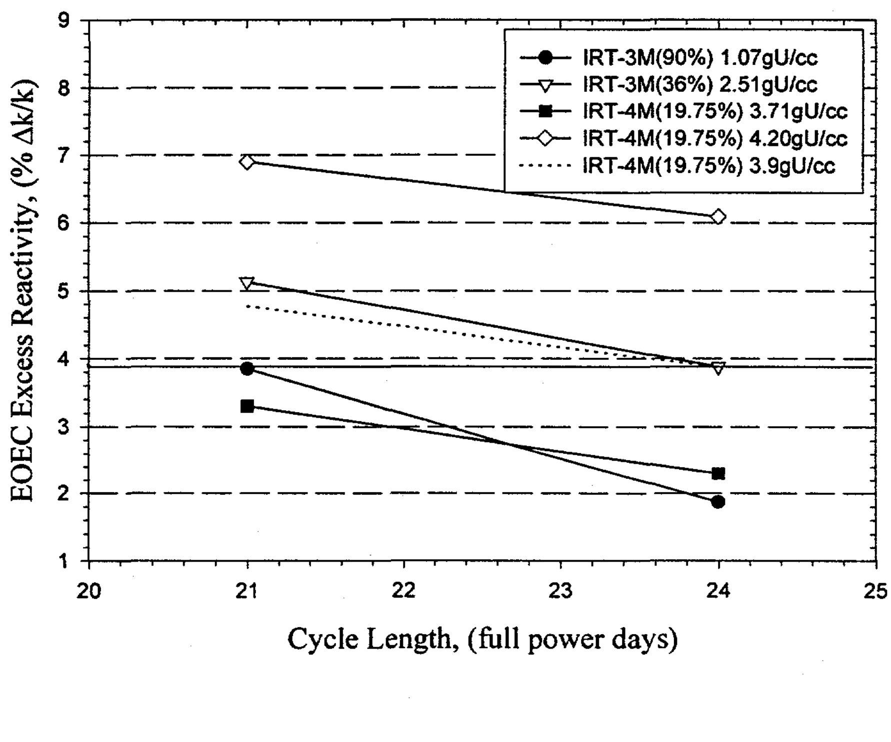
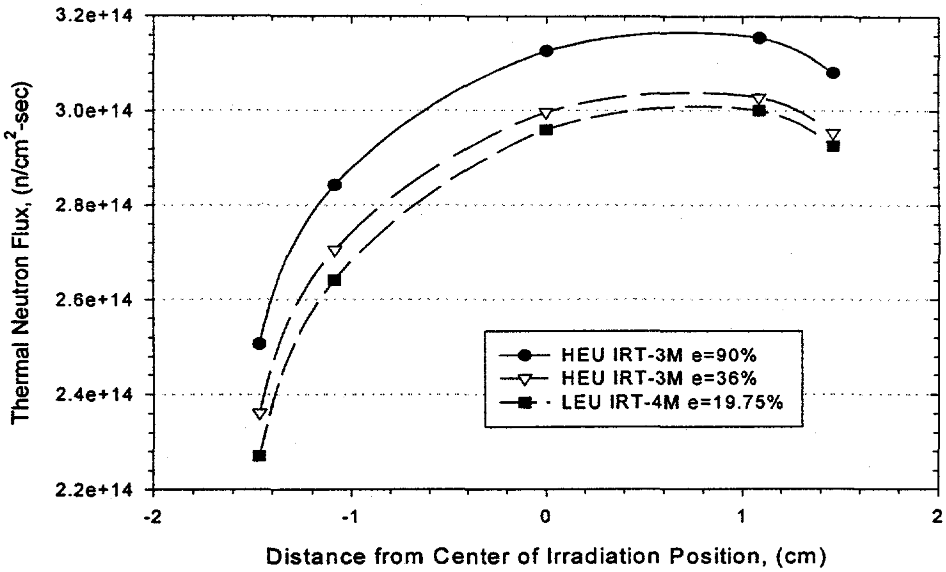


Fig. 6. WWR-SM Comparison of Thermal Flux Profiles within Irradiation Position 54.



CONTROL ROD WORTH COMPARISONS

The MCNP Monte-Carlo model was used to compute the reactivity worth of several control rod configurations as a function of fuel used in a fresh core of the WWR-SM reactor. The results are summarized in Table 3.

Table 3. WWR-SM Reactivity Worth of Control Rods

Inserted Control Rods[1]	Worth of Inserted Control Rods (% Δk/k)		
	HEU (90%) core	HEU (36%) core	LEU (19.75%)core[2]
CR-1 and CR-2	5.81 ± 0.04	5.47 ± 0.04	5.45 ± 0.04
CR-3 and CR-4	4.86 ± 0.05	4.58 ± 0.04	4.56 ± 0.04
All Three Safety Rods	4.04 ± 0.05	3.70 ± 0.04	3.81 ± 0.04

[1] All inserted control rods are moved to 1.5 cm below the bottom of the active fuel
[2] Loading of LEU (19.75%) is 3.7 g/cm^3

The reactivity worth of each combination of control rods in Table 3 was calculated with all other rods out of the core. The inserted control rods were moved to 1.5 cm below the bottom of the active fuel. In all three control rod configurations, the reactivity worth of the control rods in the HEU (36%) core is about 0.3% Δk/k less than their worth in the HEU (90%) core. Between the cores with HEU (36%) fuel and LEU (19.75%) fuel, the control rod worths are very nearly the same.

CONCLUSION

The WWR-SM research reactor began conversion in August 1998 from IRT-3M FA containing 90% enriched uranium to IRT-3M FA containing 36% enriched uranium. The uranium density in the UO_2-Al fuel meat of the 36% enriched fuel is 2.5 g/cm^3. Burnup test of three fuel assemblies with 36% enrichment had been successfully completed to a burnup of about 50% in 1987-1989.

Equilibrium fuel cycle calculations for the core with HEU (90%) IRT-3M FA and a cycle length of 21 days gave a ^{235}U discharge burnup of about 43% in the 6-tube FA. This agrees with reactor operating experience. The calculated excess reactivity at the end of the equilibrium cycle was higher than expected. Approximate concentrations of ^{3}He and ^{6}Li in the beryllium reflector blocks that were included in the calculations had a negative worth of 1.5% Δk/k. Inclusion of a more detailed history of the beryllium blocks is expected to increase the poisoning effects because the blocks were actually out of the core for several prolonged periods that are currently not known.

Equilibrium burnup calculations for the core with HEU (36%) IRT-3M FA predict a cycle length of 24 days, an increase of 3 days over the HEU (90%) core. Peak thermal neutron fluxes in two of the key experiment positions were calculated to be lower by about 5% . Control rod worths were computed to be about 0.3% Δk/k lower in the HEU (36%) core than in the HEU (90%) core. Shutdown margins are satisfactory.

With LEU (19.75%) UO_2-Al fuel in the IRT-4M FA geometry, a uranium density about 3.8 g/cm^3 is needed to match the 21 day cycle length of the HEU (90%) core, and a uranium density of about 3.9 g/cm^3 is needed to match the anticipated 24 day cycle length of the HEU (36%) equilibrium core. Peak thermal fluxes in key experiment positions with 3.8 and 3.9 g/cm^3 LEU fuel would be reduced by 7-8% relative to the HEU (90%) fuel and by about 2% relative to the HEU (36%) fuel. Control rod worths in the LEU cores were calculated to be nearly identical to those for the HEU (36%) core.

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