NEUTRONIC PERFORMANCE OF SEVERAL LEU FUEL ASSEMBLY DESIGNS FOR THE WWR-SM RESEARCH REACTOR IN UZBEKISTAN

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ABSTRACT

The 10 MW WWR-SM research reactor in Uzbekistan currently uses HEU (36%) IRT-3M 6-tube fuel assemblies manufactured by the Novosibirsk Chemical Concentrates Plant in Russia. Recent 4x4 core configurations reflected by beryllium have been operated at 8 MW. The Institute of Nuclear Physics plans to convert the reactor to LEU (19.7%) fuel as soon as a suitable LEU fuel assembly is qualified. This study compares the neutronic performance of the reactor and its experiments using LEU pin-type and LEU tube-type fuel assembly designs with the current HEU (36%) reference fuel assembly.

Both 3D Monte Carlo and 3D diffusion theory calculations were performed to analyze a critical core configuration with partially-burned HEU fuel assemblies in order to establish the credibility of the analytical methods and computer models used to describe the reactor and its experiments. Results based on these techniques are in reasonable agreement with the measured data.

An LEU pin-type design (164 pins, 4.5 g U/cm^3, 375g ^{235}U) or an LEU tube-type design (IRT-3M, 6-tube, 5.4 gU/cm^3, and 364g ^{235}U) with U9Mo-Al fuel meat could operate with about the same cycle length and experiment load as the reference HEU (36%) IRT-3M fuel. The annual fuel assembly consumption would be nearly the same in these HEU and LEU cores. For the LEU pin-type design, fast (thermal) fluxes would be reduced by 2.5% (14%) for experiments located at the center of the fuel assemblies and by 0.5% (4%) for experiments located in experiment channels in the beryllium reflector. For the LEU tube-type design, fast (thermal) fluxes would be reduced by 3.5% (15%) for experiments located at the center of the fuel assemblies and by 1.2% (5%) for experiments located in experiment channels in the beryllium reflector.

If the ^{235}U content of the LEU pin-type fuel assemblies were increased to 480g (using pins similar to those planned to be tested in the WWR-M reactor at Gatchina, Russia in 2003 and 2004), the annual consumption of fuel assemblies would be reduced by about 35%. Fast (thermal) fluxes would decrease by about 3.5% (27%) in the experiment positions in the core and by about 1.5% (8%) in the experiment positions in the beryllium reflector. However, for this 4x4 core configuration the LEU IRT-4M 6-tube fuel assemblies with UO_2-Al fuel meat, ~3 gU/cm^3, and 265g ^{235}U would increase the annual fuel assembly consumption by over 300%.

INTRODUCTION

The WWR-SM research reactor in Uzbekistan currently uses HEU (36% enrichment) IRT-3M fuel assemblies (6-tubes, UO_2-Al, 2.5g U/cm^3, 309g ^{235}U/FA) fabricated by the Novosibirsk Chemical Concentrates Plant in Russia. The Institute of Nuclear Physics (Tashkent) plans to convert the WWR-SM reactor to LEU (19.7%) as soon as a suitable fuel assembly is qualified. This study compares the reactor performance for LEU pin-type and LEU tube-type fuel assemblies with the HEU (36%) IRT-3M reference fuel. The LEU fuel meat consists of a U-Mo alloy (9 wt% Mo) dispersed in aluminum. Good irradiation behavior for high-density U-Mo dispersion fuels with Mo weight fractions in the 6-10% range has been reported in Ref.'s 1-2. Most of these neutronic studies used a preliminary design of 164-pin LEU fuel assemblies with ^{235}U loadings of 375g and 480g. The LEU IRT-3M fuel assembly has the same geometry as the reference fuel but a ^{235}U loading of 364g. Two LEU pin-type fuel assemblies are scheduled to be irradiation-tested in the WWR-SM research reactor.

THE WWR-SM RESEARCH REACTOR

Reactor Description:
The WWR-SM reactor is located at the Institute of Nuclear Physics in Ulugbek, 30 km NE of Tashkent, Uzbekistan. The reactor first reached criticality in September 1959 and since then has been upgraded from 2 to 10 MW. Since July 1999 the reactor has used a beryllium-reflected compact core of 16 (sometimes 18) Russian-supplied 6-tube IRT-3M (36%) fuel assemblies. The control system consists of 6 (sometimes 8) shim rods, 3 safety rods and one automatic regulating rod (AR). All control rods are 23 mm in diameter and consist of a 600 mm column of B_4C absorbers (1.7 g/cm^3) inside a 1 mm thick stainless steel tube. An aluminum alloy follower rod, 23 mm in diameter and 520 mm long, is located below each control rod except for the AR rod. For a fully withdrawn control rod, the bottom of the B_4C column is 124 mm above the top of the fuel meat. The active core height is 580 mm. Recently, the reactor has operated at 8 MW with a beryllium-reflected 4x4 core configuration.

The reactor is used for radioactive isotope production, neutron activation analyses, and experiments in nuclear physics, solid-state physics, and nuclear engineering. To carry out these measurements the reactor has 9 horizontal beam tubes, a graphite thermal column, and numerous vertical irradiation channels in both core and reflector regions. Ref. 3 gives a description of the WWR-SM research reactor.

Fuel Assemblies:
Figure 1 shows horizontal cross sections of the 6-tube and 8-tube IRT-3M (36%) fuel assemblies, recently tested (Ref. 4) 6-tube and 8-tube IRT-4M (19.7%) fuel assemblies, and a proposed (Ref. 5) 164-pin LEU IRT-MR fuel assembly. The corner radius of the fuel tubes is 4.2 mm. Note that the IRT-4M FA's have an outer dimension of 69.6 mm. The control rods move within guide tubes located at the center of the 6-tube and pin-type fuel assemblies. Irradiation experiments inside guide tubes are located at the center of those FA's without control rods and in beryllium reflector holes. The pitch of the fuel assemblies and the beryllium blocks is 71.5 mm. Table 1 summarizes characteristics of the fuel assemblies analyzed in this study.

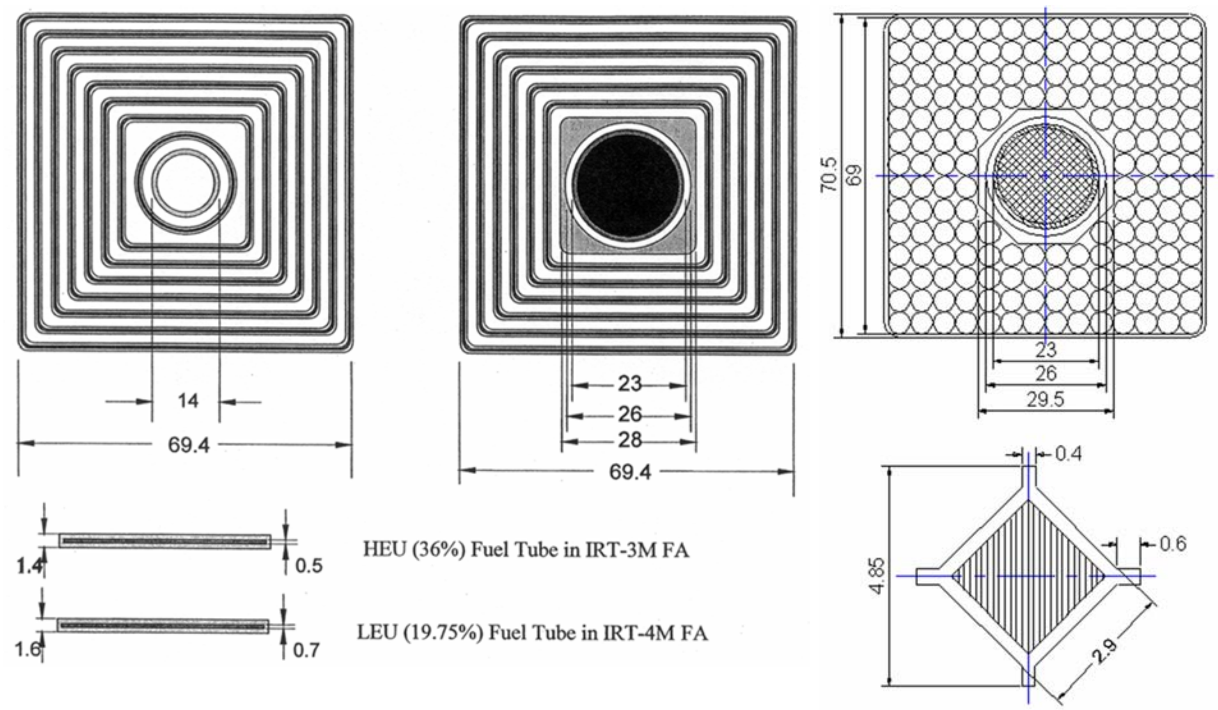


Figure 1. Horizontal Cross Sections of 6-Tube and 8-Tube FA's and a Proposed LEU 164-Pin Fuel Assembly

Table 1. Fuel Assembly Parameters

Parameter	6-Tube IRT-3M	6-Tube IRT-3M	6-Tube IRT-4M	8-Tube IRT-4M	164-Pin IRT-MR	164-Pin IRT-MR
Dispersant	UO_2-Al	U9Mo-Al	UO_2-Al	UO_2-Al	U9Mo-Al	U9Mo-Al
Wt. % ^{235}U	36.0	19.7	19.7	19.7	19.7	19.7
$(gU/cm^3)_{meat}$	2.51	5.40	2.79	2.79	4.54	5.81
VF^D, %	28.5	35.1	31.7	31.7	29.5	37.8
$^{235}M/FA$, g	309	364	265	300	375	480
H_{meat}, cm	58.0	58.0	58.0	58.0	58.0	58.0
T_{meat}, mm	0.50	0.50	0.70	0.70	2.1 x 2.1	2.1 x 2.1
T_{clad}, mm	0.45	0.45	0.45	0.45	0.40	0.40
$T_{coolant}$, mm	2.05	2.05	1.85	1.85		
Vol_{meat}, cm^3	342	342	481	545	419	419

NEUTRONIC METHODS AND CODES

Neutron Cross Sections:
Neutron cross sections for both diffusion theory and Monte Carlo calculations are based on ENDF/B-VI data[6]. Multi-group cross sections for use in static and burnup diffusion theory calculations were generated with the WIMS-ANL code (Ref. 7) and its 69-group library. These cross sections were collapsed into 7 broad groups with energy boundaries (eV) of 1.0E+7, 8.21E+5, 5.53E+3, 4.0, 6.25E-1, 2.50E-1, 5.8E-2, and 1.0E-5.

Since WIMS is a one-dimensional code, tube-type fuel assemblies were represented by a series of concentric circular regions of coolant, clad and fuel meat materials. Region radii were chosen to preserve the area of each region (see Fig. 1). Cross sections were also generated for non-depletable regions in the reactor including beryllium, water and graphite reflectors; control rods and control rod followers; and in-core and ex-core experiment regions.

Diffusion and Burn-up Calculations:
Three-dimensional XYZ models (without beam tubes) of the WWR-SM research reactor were analyzed with the diffusion theory DIF3D code[8] and with the fuel cycle analysis code REBUS-PC[9] using the WIMS-generated 7 group microscopic cross sections. Results from REBUS equilibrium cycle and non-equilibrium fuel depletion calculations were needed in the analyses of the WWR-SM research reactor.

Monte Carlo Calculations:
Because of the limitations of diffusion theory, Monte Carlo calculations with the MCNP code[10] have been used to obtain more detailed and accurate results for the WWR-SM reactor. Results from 3D MCNP models of the WWR-SM reactor include beam tube effects, reactivities, fast and thermal neutron fluxes, isotopic reaction rates, and shutdown margins. Where possible, MCNP Monte Carlo results are compared with REBUS/DIF3D diffusion theory results.

Transport Calculations:
Group-dependent internal boundary conditions (i.e. current-to-flux ratios) were calculated (P_1S_8) at the SS tube surface of the B_4C control rods using the transport code TWODANT[11]. These internal boundary conditions (IBC's) are needed in diffusion theory calculations to account for strong absorption effects of fully and partially inserted control rods. Since the IBC's were calculated for cylindrical control rods, their values were multiplied by a factor of $\pi^{1/2}/2$ when applied to the square rods (of equal cross sectional area) used in the XYZ diffusion theory model of the reactor.

CRITICAL CORE CONFIGURATION

To test our analytical methods, a recent WWR-SM core configuration, with partially burned fuel and with measured control rod elevations for the critical condition at startup, was analyzed. Results from these calculations are discussed below.

Description:
Figure 2 shows the critical configuration of the xenon-free core at startup on August 20, 2001. The beryllium-reflected 4x4 core consists of 12 6-tube IRT-3M (36%) FA's, two 6-tube IRT-4M (19.75%) FA's, and two 8-tube IRT-4M (19.75%) FA's. Figure 1 shows a horizontal cross section of these FA's and Table 2 gives the location, the initial ^{235}U mass, and the ^{235}U % burn-up of each FA in the core. The control system consists of 3 safety rods (3-5, 6-4, 6-6), 6 shim rods (4-3, 4-5, 4-6, 5-3, 5-4, 5-6), and one automatic regulating rod (7-6). Except for the automatic rod, each control rod has a SAV-1 follower rod of the same diameter. The experiment (Exp) locations shown in Fig. 2 were used in this core for radio-isotope production.

Analytical Methods:
To approximate the reactivity effect of the burned FA's given in Table 2, a non-equilibrium REBUS depletion calculation was done for this core configuration (Fig. 2)

beginning with all fresh fuel. The total cycle length was divided into many burn steps so that atom densities in the burned FA's could be selected with ^{235}U burn-ups nearly equal to the Table 2 values.

Fast neutron ^{9}Be (n,α) ^{6}Li reactions in the beryllium reflector lead to the build-up of ^{6}Li and ^{3}He poisons. An accurate calculation of this poisoning effect requires the detailed irradiation history of the Be blocks and this information is not available. Therefore, the estimate of these poison concentrations given in Ref. 3 was used in this study.

The initial set of IBC's over-predicted the worth of the shim rods because it did not take into account the depletion of ^{10}B in the outer layers of the rod resulting from long exposures to neutrons. Because detailed information concerning the irradiation history of the WWR-SM shim rods is unknown, an approximate method was used to estimate the ^{10}B depletion in the shim rods from many years of use. This method is outlined in Table 3. Based on these estimates for the depleted ^{10}B concentrations in the inner, middle and outer layers of the rod, a second set of IBC's was calculated and applied to the SS tube surface of the boron carbide shim rods. Since the automatic rod was replaced shortly before this criticality measurement and since the safety rods are fully withdrawn into the upper reflector while the reactor is at power, it was assumed that the ^{10}B depletion in these rods was negligible.

Results:

Table 4 summarizes results obtained for the WWR-SM August 20, 2001 critical configuration. Multiplication factors calculated by MCNP and by REBUS are in close agreement and are near unity if depleted ^{10}B concentrations (obtained by the method outlined in Table 3) are used for the shim rods.

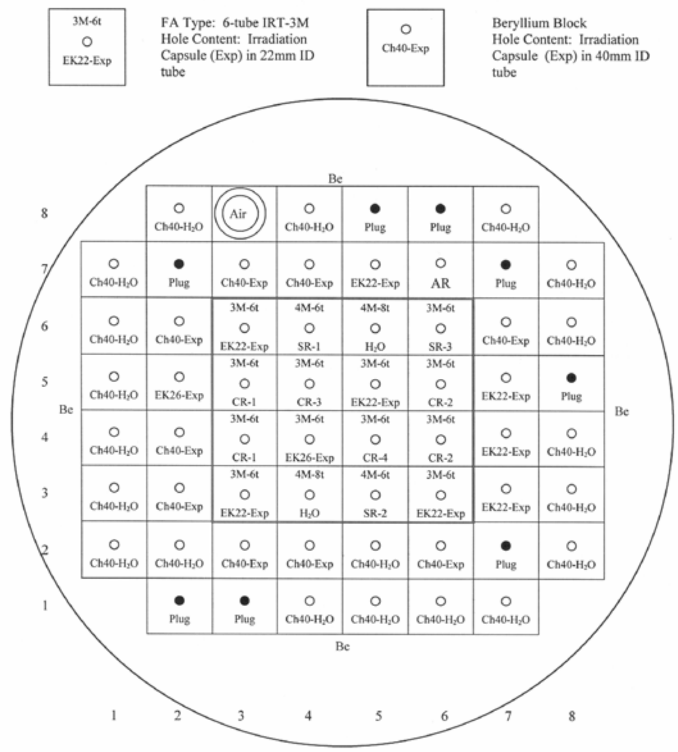
Control rod worths from MCNP and REBUS calculations are in reasonable agreement. Unfortunately, rod worths were not measured for this critical configuration. Measured values given in Table 4 are for a similar core configuration (November 23, 2001) but with two additional IRT-3M (36%) FA's located in positions 4-7 and 5-2.

REBUS/MCNP fast and thermal neutron flux ratios were calculated at several irradiation locations. Diffusion theory cannot accurately calculate the group 1 fluxes because the transverse dimensions of the irradiation regions are small relative to the group 1 transport mean free path. Note that for those energy groups where the region radius divided by the transport mean free path (R/λ) is greater than about 3, MCNP and REBUS fluxes agree.

Conclusion:

MCNP Monte Carlo and REBUS diffusion theory results are in reasonable agreement with each other and with the observed critical condition of the August 20, 2001 core configuration. Because of the limitations of diffusion theory, Monte Carlo methods are needed to calculate absolute fast neutron fluxes and fast neutron activation rates. Nevertheless, the results show that these analytical methods can produce reliable neutronic performance indicators for the WWR-SM beryllium-reflected 4x4 core configuration.

Model of the WWR-SM 01.08.20 Critical Configuration



SR = Safety Rod, CR=Shim Rod, AR=Automatic Rod

Note: The cold, xenon-free critical condition of the core (Fig. 2) corresponds to the following rod insertions (cm): CR-1(4-3)=58.0, CR-1(5-3)=56.5, CR-2(4-6)=55.5, CR-2(5-6)=57.5, AR(7-6)=30.0, CR-3(5-4)=CR-4(4-5)=SR-1(6-4)=SR-2(3-5)=SR-3(6-6)=0.0.

Figure 2

Table 2. Fuel Assemblies in the August 20, 2001 Core

Fuel Assembly Type	^{235}U Enrichment, %	FA Location	Initial ^{235}U Mass, g	^{235}U Burn-up, %
IRT-3M, 6 tube	36.0	6-3	304.2	36.3
IRT-4M, 6 tube	19.75	6-4	264.5	36.8
IRT-4M, 8 tube	19.75	6-5	297.8	36.4
IRT-3M, 6 tube	36.0	6-6	306.5	25.0
IRT-3M, 6 tube	36.0	5-3	301.8	19.1
IRT-3M, 6 tube	36.0	5-4	306.1	0.0
IRT-3M, 6 tube	36.0	5-5	303.3	2.5
IRT-3M, 6 tube	36.0	5-6	302.4	11.4
IRT-3M, 6 tube	36.0	4-3	301.3	11.5
IRT-3M, 6 tube	36.0	4-4	301.6	5.5
IRT-3M, 6 tube	36.0	4-5	304.6	0.0
IRT-3M, 6 tube	36.0	4-6	306.8	19.0
IRT-3M, 6 tube	36.0	3-3	304.3	27.5
IRT-4M, 8 tube	19.75	3-4	299.5	37.6
IRT-4M, 6 tube	19.75	3-5	260.3	38.0
IRT-3M, 6 tube	36.0	3-6	303.3	36.7

Table 3. Estimate of ^{10}B Depletion in the WWR-SM Shim Rods

B_4C Region	Radius (cm)
Inner/Middle/Outer	0.75/0.90/1.15

For each region, $N(t)/N_o \approx e^{-(\sigma\phi)t}$, where

t = the effective time at power for the 8 MW 4x4 core configuration
$\sigma\,\phi = \Sigma_g\, \sigma(n,\alpha)_g\, \phi_g$
$\phi_g \approx [\phi_g\,(B_4C) / \phi_g\,(\text{Hmg'd Fuel})]_{\text{TWODANT}} \times [\phi_g\,(\text{Hmg'g Fuel})]_{\text{REBUS}}$
$\sigma\,(n,\alpha)_g$ = ^{10}B cross section for group g from WIMS-ANL

Thus, $(\sigma\,\phi)_{inner}$ = 1.9188E-09 sec^{-1}, $(\sigma\,\phi)_{middle}$ = 5.3917E-09 sec^{-1}, $(\sigma\,\phi)_{outer}$ = 1.9362E-08 sec^{-1}.

The operating history of the WWR-SM reactor is:

Time (yrs) =T	FA's in Core=N	Power (MW)=P	Hrs at Power/yr=H	t=(16/8)(PT/N)H Hrs [a]
10	28	10	5000	3.571E+04
10	24	10	5000	4.167E+04
2	16	8	5000	1.000E+04

[a] This assumes ϕ is proportional to [Power/No. of FA's in the core] and so t ≈ 8.74E+04 hours. Thus, $N(t)/N_o$ becomes (N_o = 1.475E-02 atoms/barn-cm): Inner/Middle/Outer = 5.468E-1/1.834E-1/2.263-3. Based on these concentrations group and burn-up-dependent internal boundary conditions were calculated for the WWR-SM shim rods. The same ^{10}B concentrations and radii were used in the Monte Carlo calculations.

Table 4. Results for the August 20, 2001 Critical Configuration

Multiplication Factors and Control Rod Worths

Parameter	MCNP	REBUS/DIF3D	Exp.
Multiplication Factor	0.99179±0.00022	0.99872	1.00000
Control Rod Worths, %δk/k [a]			
CR-1 + CR-2	6.39±0.03	6.12	5.26
CR-3 + CR-4	4.73±0.03	4.64	4.65
AR	0.61±0.03	0.45	0.41
SR-1	1.73±0.03	1.76	1.86

[a] The measured values are for a core (November 23, 2001) very similar to the August 20, 2001 critical configuration but with two additional IRT-3M (36%) FA's located in positions 4-7 and 5-2.

Relative Power Distributions in the WWR-SM
(Top Value-MCNP, Bottom Value-REBUS)

	3	4	5	6
6	0.0553 0.0528	0.0546 0.0535	0.0604 0.0612	0.0613 0.0604
5	0.0547 0.0540	0.0793 0.0792	0.0776 0.0780	0.0586 0.0583
4	0.0572 0.0570	0.0773 0.0772	0.0793 0.0799	0.0560 0.0564
3	0.0595 0.0596	0.0595 0.0617	0.0538 0.0548	0.0556 0.0560

Table 4. (Continued)
REBUS/MCNP Neutron Flux Ratios at Several Irradiation Locations

Location	Group 1 (0.821-10.0 MeV)		Group 7 (1.0E-5 – 0.058 eV)	
	R/λ_{tr}	Flux Ratio	R/λ_{tr}	Flux Ratio
2-4 (refl)	0.3088	1.369	3.906	1.008
4-2 (refl)	0.3088	1.367	3.906	0.980
4-4 (core)	0.1463	1.210	0.7867	0.860
5-2 (refl)	0.2973	1.325	4.632	0.990
5-5 (core)	0.1321	1.186	0.4608	0.766
7-5 (refl)	0.3254	1.285	4.388	1.008

REBUS/MCNP fast and thermal neutron flux ratios were calculated at several irradiation locations. Diffusion theory cannot accurately calculate the group 1 fluxes because the transverse dimensions of the irradiation regions are small relative to the group 1 transport mean free path. Note that for those energy groups where the region radius divided by the transport mean free path (R/λ) is greater than about 3, MCNP and REBUS fluxes agree.

LEU CONVERSION STUDIES

Reference Core Configuration:

Since MCNP-Monte Carlo and REBUS-diffusion theory analyses gave reasonable results for the WWR-SM critical configuration, a similar reactor model was used for the LEU conversion studies. The reference core configuration is the same as Fig. 2 except that the four IRT-4M FA's are replaced with IRT-3M (36%) fuel. REBUS equilibrium calculations determined axially dependent FA actinide and fission product concentrations at the beginning of the equilibrium cycle (BOEC) for the HEU (36%) reference fuel and for the LEU FA designs. At the beginning of each cycle two fresh FA's are added at the center of the core (4-4, 5-5) and two spent FA's are discharged from the edge of the core (4-6, 5-3). The fuel-shuffling pattern used is:

Fresh→ 4-4 → 4-5 → 4-3 → 3-4 → 3-5 → 3-3 → 3-6 → 4-6 → Discharge
Fresh→ 5-5 → 5-4 → 5-6 → 6-5 → 6-4 → 6-6 → 6-3 → 5-3 → Discharge

The elevations of the control rods were as follows. The first series of REBUS equilibrium calculations determined the cycle length needed for the desired K-EFF value at the end of the equilibrium cycle (EOEC) with all control rods fully withdrawn. These cycle lengths were then used in a second set of burnup calculations to determine the desired set of atom densities with the control rods inserted (in cm) as follows: CR1=CR2=SR1=SR2=SR3=0.0, CR3=CR4=35.5, and AR=30.0

These burn-up-dependent atom densities were used in standalone BOEC problems to determine reactor performance indicators for the LEU FA designs relative to the standard IRT-3M (36%) fuel. The control rod elevations used in these problems were typical of those used in the WWR-SM research reactor at startup, namely, CR1=CR2=35.0 cm, CR3=CR4=65.0 cm, AR=30.0 cm, and SR1=SR2=SR3=0.0 cm.

Reactor Performance Parameters:

Using the methods described above, Tables 5, 6 and 7 show reactor performance parameters for LEU FA designs (see Table 1) relative to the IRT-3M (36%) reference fuel. These performance indicators included BOEC shutdown margins, annual fuel consumption, average ^{235}U discharge burn-ups, and fast and thermal neutron flux decreases in core and reflector irradiation positions. The IRT-MR pin-type FA loadings of 375g and 480g were selected to provide a range of values on which to choose a final ^{235}U loading.

Table 5. BOEC (no Xe) Shutdown Conditions for the 8 MW 4x4 Core Configuration (CR1=CR2=CR3=CR4=AR=65.0 cm, SR's=0.0 cm)

Parameter	Fuel			
Fuel Type:	6-tube IRT-3M	6-tube IRT-3M	164-rod IRT-MR	164-rod IRT-MR
Dispersant	UO_2-Al	U9Mo-Al	U9Mo-Al	U9Mo-Al
Wt % ^{235}U	36.0	19.7	19.7	19.7
ρ_{meat}, gU/cm^3	2.508	5.400	4.538	5.808
VF^D, %	28.5	35.1	29.5	37.8
^{235}M/FA, g	309	364	375	480
Code	K-EFFECTIVE			
MNCP	0.96436±.00023	0.96408±.00033	0.96624±.00020	0.97847±.00021
REBUS	0.96973	0.97083	0.96663	0.97938

Note: The shut down margin criterion for the WWR-SM is that the reactor must be subcritical by at least 1% δk/k when all control rods and the automatic regulating rod are fully inserted and the safety rods are fully withdrawn. Clearly, this condition is satisfied for all these fuels.

Table 6. Approximate Annual Fuel Consumption and Average ^{235}U Discharge Burn-up For the 8 MW WWR-SM 4x4 Equilibrium Core with Experiments

Fuel Type	Enrichment %	^{235}U/FA g	Cycle Length, D	K_{eff} [a] (EOEC)	FA/year [b]	^{235}U Discharge BU, %	
						FA-46	FA-53
IRT-3M	36.0	309	~26.97 [c]	1.0200	15.45	43.1	42.6
IRT-3M	19.7	364	~26.19 [c]	1.0200	15.91	35.1	34.6
IRT-MR	19.7	375	~26.12 [c]	1.0200	15.95	34.0	33.5
IRT-MR	19.7	480	~42.08 [c]	1.0200	9.90	42.3	41.6
IRT-4M	19.7	265	~6.57 [c]	1.0200	63.40 [d]	12.4	12.2

[a] Calculated at the end of the equilibrium cycle with all rods fully withdrawn.
[b] This calculation assumes that the reactor is at power (8 MW) for 5000 hours per year. (From Ref. 3.)
[c] Calculated by linear interpolation for K_{eff} (EOEC) = 1.0200. This estimate includes the REBUS/MCNP bias (~0.6% δk/k), the assumed cold-to-hot swing (~0.4% δk/k), and the reactivity difference for rods at the normal shutdown condition (CR1=CR2=SR's=0.0 cm, CR3=CR4≈25.0 cm, and AR=30.0 cm) and all rods fully withdrawn (0.0 cm) which amounts to about 0.95% δk/k.
[d] This calculation shows that the LEU 6-tube IRT-4M fuel assembly is not a suitable replacement fuel for the HEU (36%) fuel assemblies, at least for a core of this size.

Table 7. BOEC (no Xe) Relative Neutron Fluxes in Core (5-5) and Reflector (5-7) Locations for the 8 MW WWR-SM 4x4 Core Configuration (from REBUS) (CR1=CR2=35.0 cm, CR3=CR4=65.0 cm, AR=30.0 cm, SR's=0.0 cm)

Fuel Type	% Flux Decrease in Core Location 5-5		% Flux Decrease in Refl. Location 5-7	
	Fast [a]	Thermal [b]	Fast [a]	Thermal [b]
IRT-3M, 36% 309 g	0.0	0.0	0.0	0.0
IRT-3M, 19.7%,364 g	3.48	14.8	1.20	4.88
IRT-MR, 19.7%, 375 g	2.47	13.6	0.51	3.80
IRT-MR, 19.7%, 480 g	3.50	26.8	1.47	8.26

[a] The fast flux is for neutrons with energies greater than 0.821 MeV.
[b] The thermal flux is for neutrons with energies less than 0.625 eV.

Note that experiments requiring fast neutron flux activations should be located in the core and those requiring thermal neutron flux activations should be located in the beryllium reflector. Therefore, the important neutron flux decreases are the fast flux in the core and the thermal flux in the reflector.

Fuel Assembly Powers and Power Density Factors:
Two IRT-MR pin-type FA's are planned to be irradiation tested in the WWR-SM research reactor beginning in 2003. These neutronic studies were undertaken to provide power density distributions, power peaking factors and peak power densities needed for thermal hydraulic analyses of the test assemblies. The reference core configuration, with fuel assembly burn-ups corresponding to xenon-free BOEC conditions, was used for this purpose. Control rod insertions were chosen to be typical of those used in the WWR-SM reactor at startup.

The peak power density occurs in the lower right corner of FA-56 that is adjacent to the beryllium reflector. Figure 3 shows transverse and axial normalized power density distributions for this fuel assembly where the axial distances are normalized to the height of the fuel column and the transverse distance to the width of the FA. Table 8 summarizes MCNP and REBUS power density information for a number of cases analyzed in this study. REBUS PD_{max} in Table 8 is the REBUS/DIF3D-calculated maximum power density divided by the volume fraction of meat in the homogenized fuel.

Power Density Distributions for FA-56 in the WWR-SM IRT-3M 4x4 Reference Core (Cr1=CR2=35.0 cm, CR3=CR4=65.0 cm, AR=30.0 cm, SR's=0.0 cm)

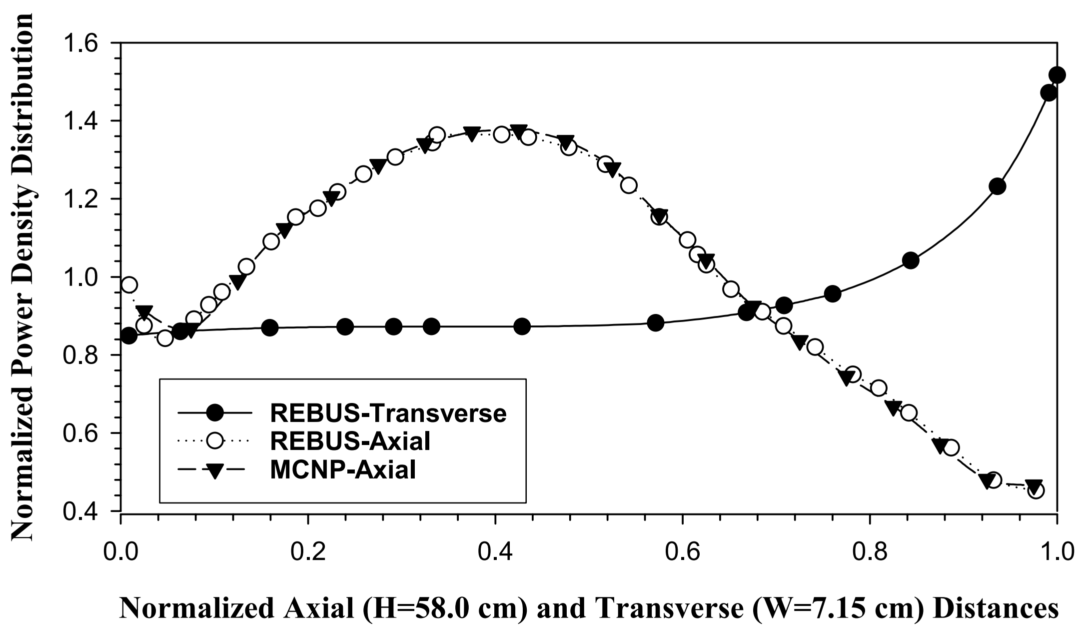


Figure 3. Transverse and Axial Normalized Power Density Distributions in FA-56

Table 8. MCNP and REBUS BOEC Power Factors in FA-56 for the Xe-Free 8 MW WWR-SM 4x4 Core Configuration with Fresh Fuel in Positions 4-4 and 5-5
(CR1=CR2=35.0 cm, CR3=CR4=65.0 cm, AR=30.0 cm, SR1=SR2=SR3=0.0 cm)

Core Configuration	Code	K_{eff}	Power %	PD_{ave} W/cm^3	PD_{max} W/cm^3	Axial PF	^{235}U BU-%
Reference Core	MCNP	1.00822	6.45	1509	2966	1.376	-
16 IRT-3M (36%) FA's	REBUS	1.01154	6.34	1484	3249	1.364	11.2
Mixed Core							
14 IRT-3M (36%) +	MCNP	1.00475	6.40	1497	2985	1.374	-
2 IRT-MR (19.7%, 375g)	REBUS	1.00648	6.29	1471	3262	1.365	11.2
Mixed Core							
14 IRT-3M (36%) +	MCNP	1.00781	6.28	1469	2983	1.376	-
2 IRT-MR (19.7%, 480g)	REBUS	1.00976	6.17	1443	3241	1.365	11.2
16 IRT-MR (19.7%, 375g)	MCNP	1.00765	6.37	1216	2784	1.347	-
All pin-type fuel	REBUS	1.00605	6.25	1193	2767	1.369	9.53
16 IRT-MR (19.7%, 480g)							
All pin-type fuel	REBUS	1.01591	6.32	1207	2960	1.352	11.7
16 IRT-3M (19.7%, 364g)							
All tube-type fuel	REBUS	1.01003	6.26	1464	3435	1.368	9.48
Mixed Core	Fresh IRT-MR (19.7%) FA in Location 5-5						
14 IRT-3M (36%) +	MCNP	1.00475	7.48	1428	2230	1.328	-
2 IRT-MR (19.7%, 375g)	REBUS	1.00648	7.38	1409	2236	1.316	0.0
Mixed Core							
14 IRT-3M (36%) +	MCNP	1.00781	8.09	1544	2394	1.326	-
2 IRT-MR (19.7%, 480g)	REBUS	1.00976	7.97	1522	2412	1.316	0.0

SUMMARY AND CONCLUSIONS

The Institute of Nuclear Physics (Tashkent, Uzbekistan) plans to convert the WWR-SM reactor to LEU (19.7%) as soon as a suitable fuel assembly is qualified. This study compares the neutronic performance of the reactor using LEU pin-type and LEU tube-type fuel assembly designs with the current HEU (36%), IRT-3M 6-tube, reference fuel. Since two IRT-MR pin-type fuel assemblies are to be irradiation-tested in the WWR-SM reactor, mixed HEU/LEU cores were also analyzed to provide information needed to determine thermal-hydraulic safety margins for the test assemblies.

Three-dimensional Monte Carlo and 3D diffusion theory methods were used to analyze a recent WWR-SM critical configuration with partially burned fuel and with partially inserted control rods. Eigenvalues and control rod worths calculated by the two methods were found to be consistent with each other and with measured values.

A beryllium-reflected 4x4 core configuration, similar to the critical core configuration, was used to determine reactor performance parameters for LEU FA's relative to the HEU (36%) IRT-3M reference fuel. For U9Mo-Al LEU (19.7%) fuel assemblies of the IRT-MR pin-type (375g ^{235}U) and the IRT-3M tube-type (364g ^{235}U) the number of FA's used per year would be nearly the same as that for the reference fuel. For the more heavily loaded IRT-MR (480g ^{235}U) FA's, however, the number of fuel assemblies needed per year would be reduced by about 36%. Shutdown margins calculated by Monte Carlo and by diffusion methods were found to be consistent with each other and well within the safety requirements of the WWR-SM reactor for the HEU reference fuel and for all the LEU fuels analyzed in this study.

Relative to the HEU (36%) reference fuel, the LEU pin-type design (IRT-MR, 375g ^{235}U/FA) would reduce fast (thermal) neutron fluxes by about 2.5% (14%) for experiments located at the center of fuel assemblies and by about 0.5% (4%) for experiments located in the beryllium reflector. For the LEU tube-type design (IRT-3M, 364g ^{235}U/FA) the reduction in the fast (thermal) neutron fluxes is about 3.5% (15%) in the core locations and about 1.2% (5%) in reflector locations. If the ^{235}U content of the IRT-MR FA is increased from 375g to 480g, the annual consumption of fuel assemblies would be reduced by about 35%. Fast (thermal) fluxes would decrease by about 3.5% (27%) in the experiment positions in the core and by about 1.5% (8%) in the experiment positions in the beryllium reflector. Since experiments requiring fast neutron activations are irradiated in the core and those requiring thermal neutron activations are irradiated in the beryllium reflector, the LEU flux losses are not very significant.

Peak power densities and power density distributions were evaluated for the HEU reference fuel and for the U9Mo-Al LEU FA designs in the beryllium-reflected 4x4 equilibrium core. In addition, power densities were calculated for several mixed cores consisting of two IRT-MR pin-type fuel assemblies and fourteen 6-tube IRT-3M (36%) FA's. These studies showed that the maximum power density and the limiting thermal-hydraulic conditions in occur in a fuel assembly next to the beryllium reflector. Based on this finding the two LEU IRT-MR pin-type test assemblies should be irradiated at the center of the core until ^{235}U burn-ups of about 25-30% have been achieved.

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