Analyses for Inserting Fresh LEU Fuel Assemblies Instead of Fresh HEU Fuel Assemblies in the Dalat Nuclear Research Reactor in Vietnam

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ABSTRACT

Analyses were performed by the RERTR Program to replace 36 burned HEU (36%) fuel assemblies in the Dalat Nuclear Research Reactor in Vietnam with either 36 fresh fuel assemblies currently on-hand at the reactor or with LEU fuel assemblies to be procured. The study concludes that the current HEU (36%) WWR-M2 fuel assemblies can be replaced with LEU WWR-M2 fuel assemblies that are fully-qualified and have been commercially available since 2001 from the Novosibirsk Chemical Concentrates Plant in Russia.

The current reactor configuration using re-shuffled HEU fuel began in June 2004 and is expected to allow normal operation until around August 2006. If 36 HEU assemblies each with 40.2 g ^{235}U are inserted without fuel shuffling over the next five operating cycles, the core could operate for an additional 10 years until June 2016. Alternatively, inserting 36 LEU fuel assemblies each containing 49.7 g ^{235}U without fuel shuffling over five operating cycles would allow normal operation for about 14 years from August 2006 until October 2020. The main reason for the longer service life of the LEU fuel is that its ^{235}U content is higher than the ^{235}U content needed simply to match the service life of the HEU fuel. Fast neutron fluxes in the experiment regions would be very nearly the same in both the HEU and LEU cores. Thermal neutron fluxes in the experiment regions would be lower by 1-5%, depending on the experiment type and location.

Introduction

The Dalat Nuclear Research Reactor (DNRR) has operated with a nominal power of 500 kW since its reconstruction in 1984 from the previous 250 kW TRIGA-MARK II reactor. The core has gone through a long period of core size expansion from startup core size of 89 fuel assemblies (FA) in 1984 until reaching current core size of 104 FA in March 2002.

In June 2004, 16 high burnup FA from the center of the core were exchanged with 16 low burnup FA from the core periphery. This operation provided additional reactivity that will allow the core to operate for about 115 full-power-days (fpd) until ~ August 2006, based on normal DNRR operation of 50-54 fpd (1200-1300 full-power-hours) per year.

Around August 2006, additional reactivity will be required for continued operation. This could be accomplished either by further shuffling of the core or by replacement of the highest burnup HEU fuel assemblies with fresh HEU or LEU fuel assemblies. For purposes of these analyses,

we assumed that HEU FAs with the highest burnup would be replaced, as suggested for DNRR (Ref. 3), rather than further shuffling HEU fuel. Calculations were done to compare the behavior of the reactor when 36 fresh HEU assemblies or 36 LEU assemblies are incrementally inserted into the core over five operating cycles.

Also included are comparisons of neutron flux performance in the irradiation positions and beam tubes for the HEU core incrementally loaded with either 36 fresh HEU or with 36 fresh LEU fuel assemblies over operating five cycles .

Core Description

Most ex-core structures were retained from the former TRIGA reactor. These include the aluminum tank, graphite reflector, thermal column, the horizontal beam tubes, and concrete shielding (Ref. 1). Figure 1 shows a radial slice of the current core with 104 fuel assemblies, along with the beam tubes, graphite reflectors and thermal column.

Figure 2 provides an enlarged view inside the core vessel, showing the fuel, control rods and irradiation positions. A cross-sectional view of the WWR-M2 fuel assembly is shown Figure 3. Each HEU (36%) assembly contains an average of 40.2 g ^{235}U with fuel meat consisting of U-Al alloy (35 wt% U, 65 wt% Al). Each LEU (19.7%) assembly contains an average of 49.7 g ^{235}U with UO_2-Al dispersion fuel meat. In Figure 3, each of the fuel elements (tubes) in the HEU and LEU fuel assemblies has the same thickness of 2.50 mm, but the fuel meat and cladding thicknesses are different.

The core is cooled and moderated using distilled light water. The core is primarily reflected by graphite with a small amount of Be surrounding the core periphery inside the core vessel. There is also some Be reflector material surrounding the large water channel located in the center of the core. There are two dry irradiation positions and one wet irradiation position located in the core periphery. There are two safety control rods, four shim control rods and one regulating control rod.

Neutronics Models of the DNRR

The reactor core and ex-core materials were modeled using hexagonal-Z multi-group diffusion theory and continuous energy Monte Carlo methods. The MCNP (Ref. 6) code was used to perform the Monte Carlo calculations using an ENDF-B/VI cross section library. A detailed geometrical model of each reactor component was made in the MCNP model, except in the axial reflectors above and below the FA where some materials were homogenized. MCNP calculations were used to calculate control rod worths, and neutron flux performance in irradiation positions, and to compare with core criticality measurements.

All fuel cycle calculations were performed with the diffusion theory code REBUS-PC (Ref. 7) using the nodal flux solution method. The neutron cross sections for use in REBUS-PC were generated using WIMS-ANL (Ref. 8). The WIMS-ANL uses a 69 energy group library based on ENDF-B/VI data and collapsed to seven broad energy groups for use in REBUS-PC. The FA

Figure 1. The Dalat Nuclear Research Reactor

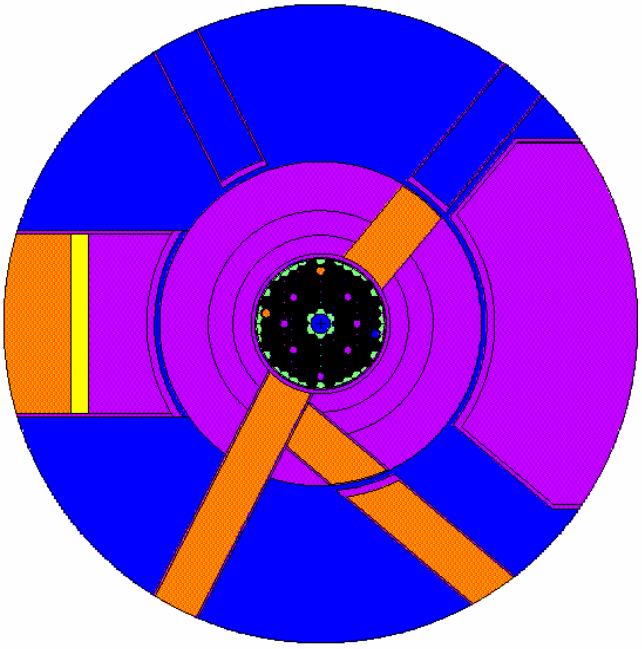


Figure 2. The Core of the DNRR Loaded with 104 WWR-M2 Fuel Assemblies

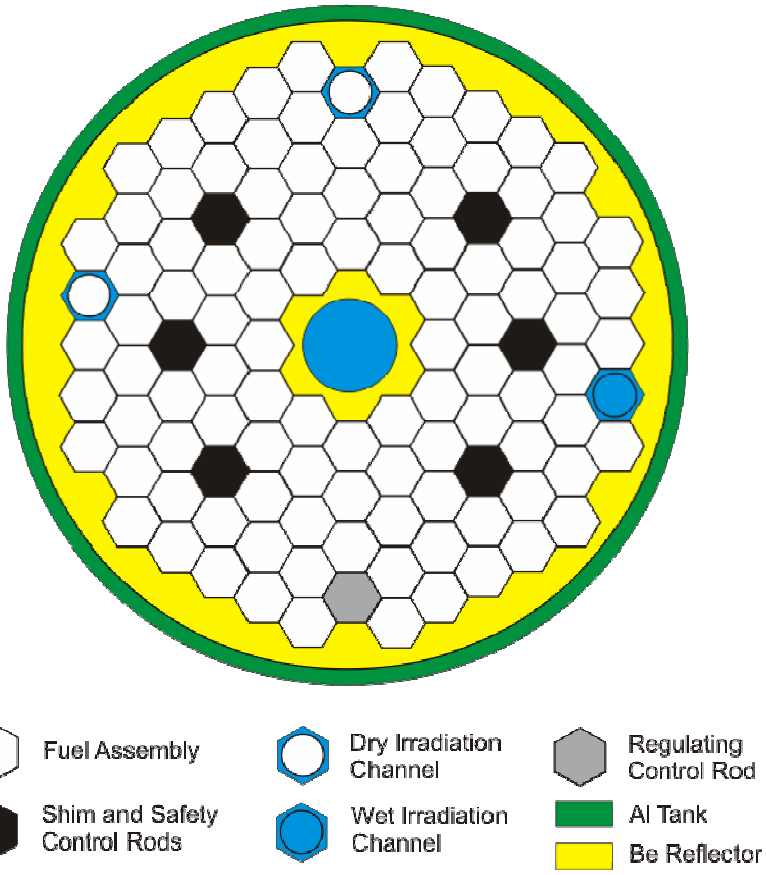
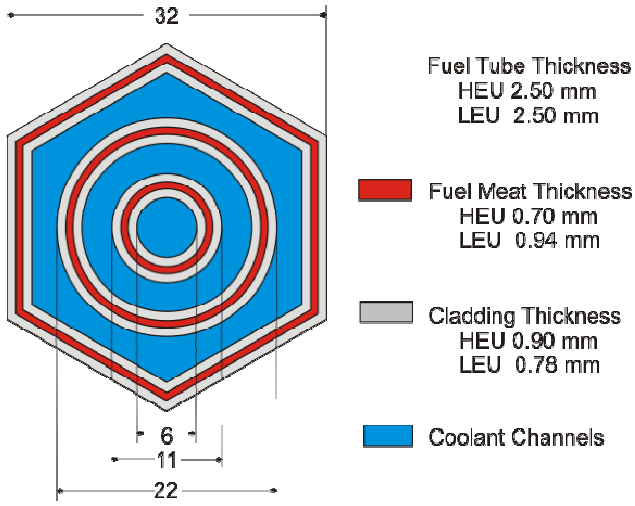


Figure 3. Cross Section of the WWR-M2 HEU and LEU Fuel Assembly.



cross sections were generated in a radial geometry with each fuel element depleted based upon its unique neutron spectrum in the WIMS-ANL model. A separate WIMS-ANL model was created to generate cross sections for the control rods, irradiation positions, and reflector materials.

The seven broad group microscopic cross sections were either polynomial or spline fitted as a function of burnup for use in REBUS-PC. The REBUS-PC fuel depletion chains included production of six Pu isotopes, Am-241, Np-237, and lumped fission product. Isotopic precursors of Xe-135 and Sm-149 were also included in the depletion chains so that Xe and Sm transients during periods of shutdown and startup could be modeled. Each homogenized FA was modeled using five equal volume axial depletion zones. The beam tubes were modeled using a homogenized mixture of air or concrete, graphite and aluminum.

Comparison of Measured Core Excess Reactivities with Calculated Results

Several critical core configurations using fresh WWR-M2 HEU FA have been measured at the DNRR. The core sizes range from 69 FA to 88 FA. The smaller cores have very few control rods inserted while the larger cores required more reactivity control. A representative number of the critical core configurations were modeled using MCNP. The results were tabulated in Table 1. The configuration of cores with 69 FA and 88 FA are shown in Figure 4. The uncertainty in the calculated k_{eff} was < 0.03% for all 10 critical cores modeled. The average MCNP k_{eff} was 0.99516 +/-0.00022 for the 10 critical core configurations. This result obtained from the various core configurations establishes the credibility in the MCNP model of the DNRR cores using fresh HEU fuel.

Two additional comparisons of the computed and measured excess reactivity were made for the March and June 2004 burned HEU cores. Using the burnup distribution of the March 2002 provided by the DNRR staff, REBUS and MCNP models were developed. This core was

depleted for 120.8 fpd until the March 2004 burnup distribution was reached. The June 2004 core was modeled based on moving 16 lower burnup FA from the outside of the core to the interior of the core. The excess reactivity of the March 2004 and June 2004 cores were compared and presented in Table 2. The calculated excess reactivity was 0.52% and 0.50% greater than measured for the March 2004 and June 2004 cores, respectively.

Table 1. Calculated DNRR HEU Critical Configurations

Configuration	Calculated (MCNP) k_{eff}	Excess Reactivity, % Δk/k
69 FA (Fig 2.1, Ref 2)	0.99772 ± 0.00029	- 0.28 ± 0.03
72 FA (Fig 2.2, Ref 2)	0.99679 ± 0.00029	- 0.32 ± 0.03
74 FA (fig 2.3 Ref 2)		
Configuration #1	1.00028 ± 0.00028	+ 0.03 ± 0.03
Configuration #2	0.99734 ± 0.00017	- 0.27 ± 0.02
Configuration #3	0.99519 ± 0.00019	- 0.48 ± 0.02
Configuration #4	0.99710 ± 0.00019	-0.29 ± 0.02
75FA (Fig 2.4, Ref 2)	0.99478 ± 0.00017	-0.52 ± 0.02
86 FA (Fig 2.5, Ref 2) – Configuration #3	0.99124 ± 0.00018	- 0.88 ± 0.02
88 FA (Fig 2.6, Ref 2) – Configuration #1	0.99022 ± 0.00018	- 0.99 ± 0.02
88 FA (Fig 2.7, Ref 2) – Configuration #1	0.99096 ± 0.00022	- 0.91 ± 0.02
Average of 10 Calculated Critical Configurations	0.99516 ± 0.00022	- 0.49 ± 0.02

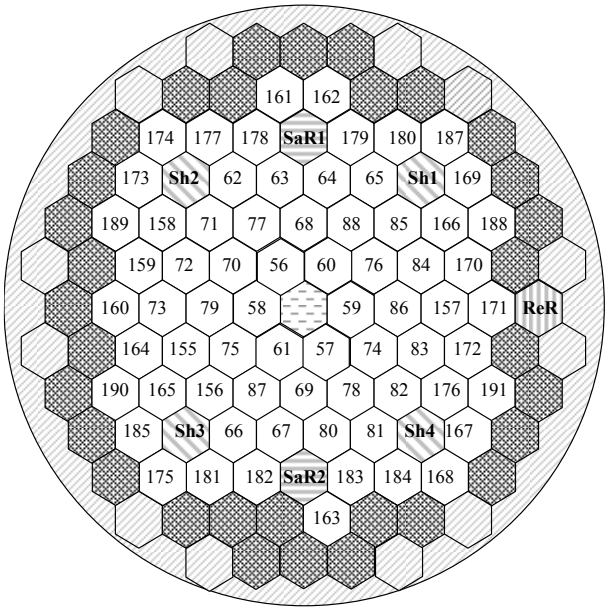
Table 2. Calculated Results for Recent DNRR Cores With 104 Fuel Assemblies

Configuration	Measured. Refs. 2-3	MCNP	REBUS
Fresh HEU Core			
With Beam Tubes	N/A	1.09741 ± 0.00013	1.09774
Without Beam Tubes	N/A	1.11153 ± 0.00015	1.11330
Worth of Beam Tubes		-1.16%	-1.27%
March 2004 Burned Core w/Beam Tubes			
Calculated k_{eff}		1.02814 ± 0.00012	1.02802
Excess Reactivity	2.72 β	3.38 β	3.36 β
Reactivity Difference (Calc-Meas)		0.53%	0.52%
June 2004 Burned Core w/Beam Tubes			
Calculated k_{eff}		1.03500 ± 0.00015	1.03514
Excess Reactivity	3.57 β	4.18 β	4.19 β
Reactivity Difference (Calc-Meas)		0.49%	0.50%
June 2004 Burned Core w/Beam Tubes			
Worth of All Shim Rods	10.44 β	12.08 β --- C/M* = 1.15	N/A
Worth of Regulating Rod			
a) All shim Rods In	0.51 β		
b) All shim Rods Out		0.52 β	
c) Average of a) and b)		0.38 β	
		0.45 β --- C/M = 0.88	

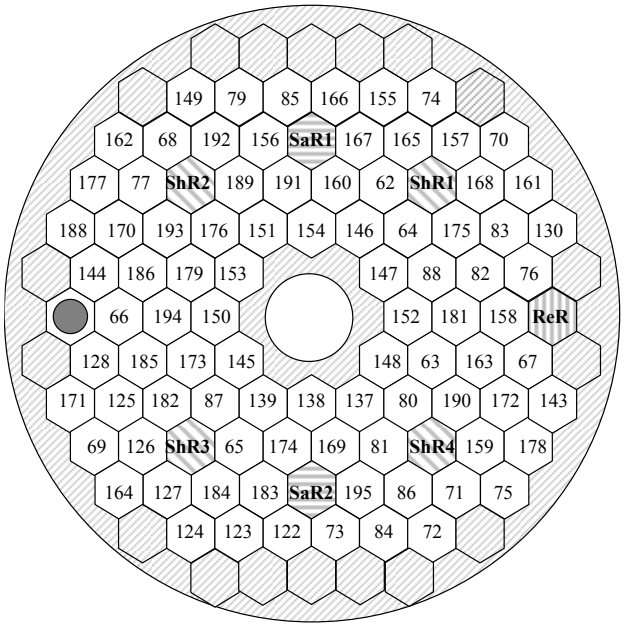
Note: β = 0.81% (Ref. 4)

* Calculated/Measured

Figure 4. Horizontal Cross Sections of Two of the Ten DNRR HEU Critical Configurations That Were Calculated in Table 1.



69 FA Core From Fig. 2.1, Ref. 2



88 FA Core From Fig. 2.7, Ref. 2

Fig 2.1 (Ref. 2) Critical configuration without neutron trap
(69 Fuel Assemblies, 12 Beryllium rods and 32 Aluminum chock rods)

Identification of each Fuel assembly is written on fuel lattice cell

1. Position of Control Rods (**From the Top of Reactor Core**): Shim Rod1=0 (cm); Shim Rod2=0 (cm); Shim Rod3=0 (cm); Shim Rod4=6 (cm); Regulating Rod=0 (cm); Safety Rods=0 (cm)
2. Mass of Uranium 235 = 2781,3 (gram)

Fig 2.7 (Ref. 2) Critical configuration with neutron trap
(88 FAs, 18 Be Rods. Regulating Rod is Stainless Steel)

Position of Control Rods with 6 critical configurations (**cm** Unit) in January, 23rd, 1984

1/ ShR1=36.8	ShR2=65.0	ShR3=65.0	ShR4=36.8	ReR=65.0	2SaR=0
2/ ShR1=37.5	ShR2=65.0	ShR3=65.0	ShR4=37.5	ReR=40.0	2SaR=0
3/ ShR1=39.2	ShR2=65.0	ShR3=65.0	ShR4=39.2	ReR=40.0	2SaR=0
4/ ShR1=65.0	ShR2=37.5	ShR3=37.5	ShR4=65.0	ReR=65.0	2SaR=0
5/ ShR1=65.0	ShR2=38.0	ShR3=38.0	ShR4=65.0	ReR=40.0	2SaR=0
6/ ShR1=65.0	ShR2=39.5	ShR3=39.5	ShR4=65.0	ReR=20.0	2SaR=0

Estimated Core Lifetime Using 36 Additional HEU or 36 LEU FA

When the cycle that began operation in June 2004 is completed around August 2006, additional reactivity will be required for continued operation. This could be accomplished by further shuffling of irradiated HEU fuel in the present core or by replacing the highest burnup fuel with fresh fuel. For purposes of this report, we assumed that HEU fuel assemblies with the highest burnup would be replaced, as suggested by DNRR personnel (Ref. 3), without further fuel shuffling. Calculations were done to compare the behavior of the reactor when 36 fresh HEU assemblies or 36 fresh LEU assemblies are incrementally inserted without further fuel shuffling over five operating cycles.

The REBUS-PC model was used to estimate that the cycle length of the June 2004 core would be 115 fpd. The excess reactivity at the end of cycle (EOC) of the June 2004 core was equal to the EOC excess reactivity calculated for the March 2004 core. The REBUS-PC model was used to predict the additional core lifetime inserting either 36 fresh HEU FA or 36 fresh LEU FA.

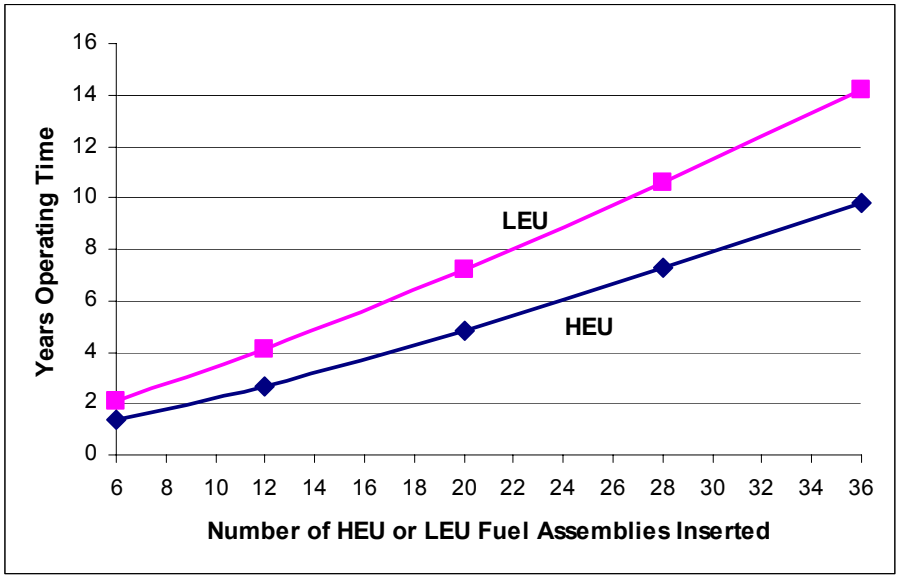
The fuel reload strategy adopted for the continued operation of the core was to discharge the highest burnup HEU FAs and reload fresh HEU (36%, 40.2 g ^{235}U/FA) or LEU (19.75%, 49.7 g ^{235}U/FA) fuel into those same core locations. The first two reload cycles were refueled with six fresh FA each and loaded into the core periphery. The next three reload cycles were refueled using eight fresh FA after discharging the highest burnup HEU FA. A total of 36 FA were utilized in all five fresh reload operation cycles.

A comparison of the operating times beginning in August 2006 for each of the five reload cycles is presented in Table 3. Figure 5 shows years of operation as a function of the number of fresh HEU or LEU fuel assemblies that are inserted into the core.

Table 3. Summary of Operating Times for Incremental Insertion of 36 Burned HEU Fuel Assemblies with 36 Fresh HEU or 36 Fresh LEU Fuel Assemblies Beginning in **August 2006**.

Cycle	HEU or LEU FA Inserted per Cycle	Total HEU or LEU FA Inserted	FPD with HEU Fuel	Cum. Years Oper. Using HEU	FPD with LEU Fuel	Cum. Years Oper. With LEU
1	6	6	70	1.4	110	2.1
2	6	12	137	2.7	210	4.1
3	8	20	247	4.8	376	7.2
4	8	28	377	7.3	552	10.6
5	8	36	507	9.8	738	14.2

Figure 5. Years Operating Time versus the Number of HEU or LEU Fuel Assemblies Inserted.



Excess reactivities are plotted in Figure 6. The base case for both HEU and LEU fuel replacement starts with the August 2006 core. The highest excess reactivity points correspond to the condition of the core just prior to startup with no Xe-135, saturated Sm-149 and all fresh fuel loaded to begin the next cycle of operation. The lowest reactivity points correspond to EOC conditions with equilibrium Xe-135 and Sm-149 concentrations in all FA. The REBUS calculated excess reactivity for all points shown in Figure 4 has been reduced by 0.55% to account for the over-prediction of the reactivity shown in Table 2. All REBUS depletion calculations were performed assuming all control rods were in their fully withdrawn position.

The results of the depletion of the current core using the 36 additional HEU FA indicate that the core could be operated an additional 507 fpd or almost 10 years from August 2006 until June 2016. If the proposed 36 LEU FA are used to refuel the core, the calculated lifetime would be 738 fpd or about 14 years from August 2006 until about October 2020. These results assumed that the core will be operated at full power for 1250 hours or 52 full-power-days per year, using the assumed strategy of loading fresh fuel assemblies without shuffling.

Shutdown Margin Comparisons for the Reload Cores

According to DNRR (Ref. 3), "The shutdown margin is relative to the cold, poisoning free core condition, and should be at least 1% sub-critical with the four shim rods and the regulating rod fully-inserted, and the two safety rods fully-withdrawn." The shutdown margins for the five reload cores are shown in Table 4. The shutdown margins were calculated using the MCNP model of the reactor with burned fuel compositions calculated using the REBUS-PC code. Each shutdown margin calculation was made after all fresh FA were loaded for startup of the next cycle with the core cold and Xe-free. This is the time in the fuel management cycle when the core has the most excess reactivity. The results show that the shutdown margin is much greater than the required value of 1.0 % $\Delta k/k$ below critical for all five reload cycles.

Figure 6. Fuel Lifetime of 36 HEU Fuel Assemblies Compared to 36 LEU Fuel Assemblies Loaded into the Dalat Research Reactor Core

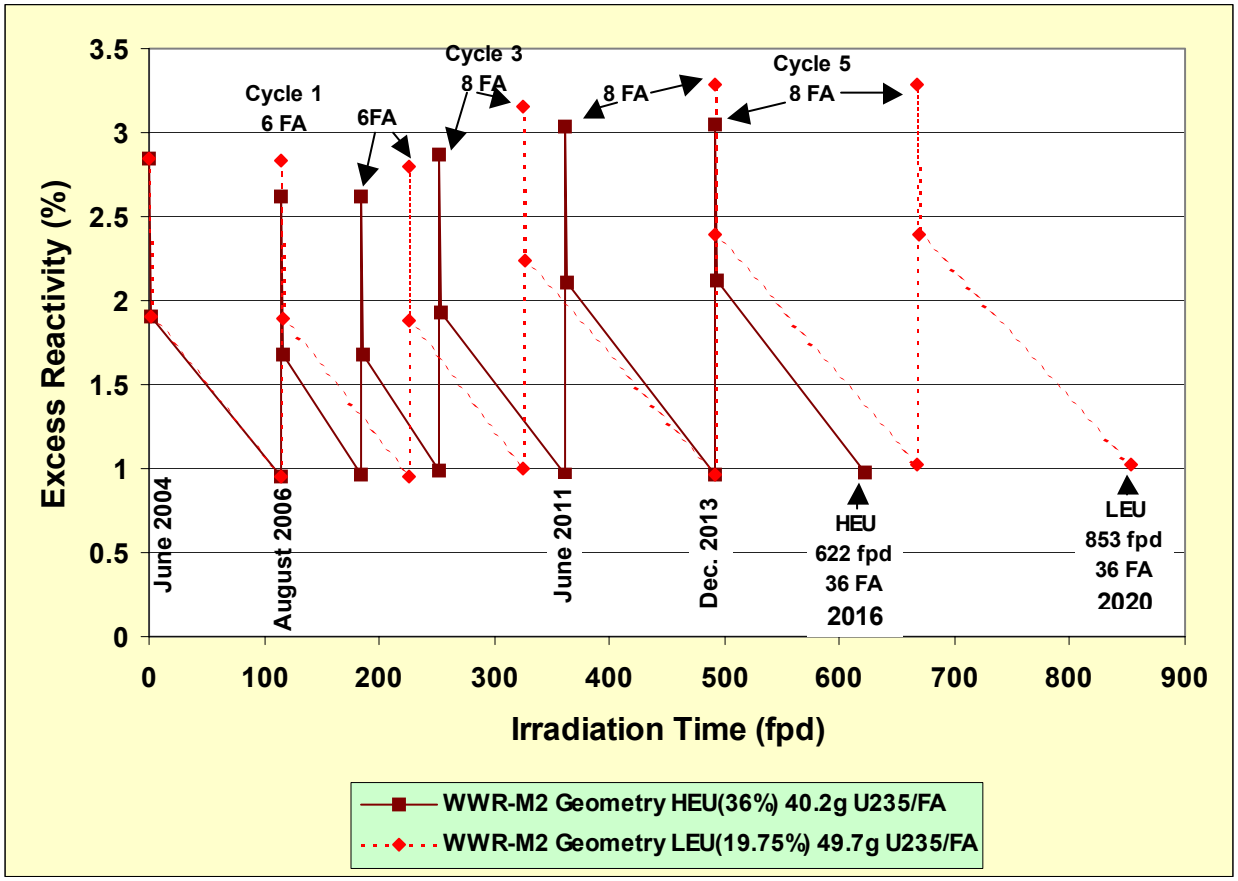


Table 4. DNRR Shutdown Margins* Calculated for Five Reload Cycles With Up To 36 HEU and LEU Fresh Fuel Assemblies

Reload Cycle Number	No. of Fresh FA Loaded	HEU FA % Δk/k	LEU FA % Δk/k
1	6	-5.78	-5.55
2	12	-5.76	-5.58
3	20	-5.48	-5.12
4	28	-5.34	-4.95
5	36	-5.31	-4.78

* The MCNP calculated reactivity worth of the four shim rods for the June 2004 HEU core was higher than the "measured" value by 1.33% Δk/k. Therefore, the absolute value of the calculated shutdown margin given in the above table was decreased by the same 1.33%.

Neutron Flux Performance

The neutron flux performance of fast and thermal fluxes in all irradiation positions is compared in Table 5. All neutron flux calculations were performed using the MCNP code. Fuel compositions were calculated using the REBUS-PC code and transferred to MCNP for these calculations. The comparison locations are the Dry and Wet irradiation positions, Central Flux Trap, at the surface of Beam Tubes #3 and #4, and in the Rotating Rack (see Figs. 1 and 2). A comparison is made at the beginning of each reload cycle after one fpd of operation without control rods inserted. The tabulated results are the presented as the ratio of the fast (> 0.821 MeV) or thermal flux (< 0.625 eV) in the LEU reloaded core to the HEU reloaded core.

The fast and thermal flux ratios decrease 1 to 2% in the Flux Trap. In all other comparison locations, the fast flux is essentially the same. In these same locations the thermal flux has been reduced 1 to 5% as more LEU fuel is reloaded.

Table 5. Neutron Flux Performance Comparisons for Five Reload Cycles: LEU/HEU Ratio

	Cycle 1		Cycle 2		Cycle 3		Cycle 4		Cycle 5	
	Thermal	Fast	Thermal	Fast	Thermal	Fast	Thermal	Fast	Thermal	Fast
Dry Irradiation Channels										
Cell 13-2	0.97	1.01	0.97	1.01	0.94	1.02	0.95	1.01	0.95	1.00
Cell 7-1	0.99	0.99	0.99	0.99	0.96	1.00	0.96	1.00	0.95	1.00
Wet Irradiation Channel										
Cell 1-4	0.99	1.02	0.98	1.02	0.96	1.02	0.97	1.03	0.96	1.03
FLUX TRAP	0.99	1.00	0.99	0.99	0.98	0.98	0.98	0.98	0.98	0.98
Beam Tubes (surface flux)										
BT # 4	0.98	1.01	0.97	1.02	0.96	1.02	0.96	1.02	0.96	1.01
BT # 3	0.99	1.01	0.98	1.01	0.97	1.02	0.97	1.00	0.97	1.01
Rotating Rack Average	1.00	1.00	1.00	1.01	1.00	1.01	0.99	1.00	0.99	1.00

LEU WWR-M2 Fuel Qualification

LEU WWR-M2 fuel assemblies with LEU UO_2-Al dispersion fuel meat and 2.5 g U/cm^3 are fully-qualified in accord with ROSATOM (Federal Agency for Atomic Energy of the Russian Federation) requirements and have been commercially available from the Novosibirsk Chemical Concentrates Plant in Russia since 2001 (Refs. 9-11).

Conclusions

The results of analyses described in this paper conclude that it is feasible to insert LEU WWR-M2 fuel assemblies instead of HEU (36%) fuel assemblies into the Dalat Nuclear Research

Reactor when additional reactivity is required for operation around August 2006. This conclusion is based on the following results:

- The MCNP models of the 10 critical configurations were in good agreement with measured data. The comparison of the REBUS-PC/MCNP models of the recent HEU burned cores was also in good agreement with measured data.
- Around August 2006, additional reactivity will need to be added to the current core which began operation with re-shuffled HEU irradiated fuel in June 2004.
- If 36 HEU fuel assemblies each containing 40.2 g ^{235}U are inserted without fuel shuffling over the next five operating cycles, the core could operate for an additional 10 years until June 2016.
- If 36 LEU fuel assemblies each containing 49.7 g ^{235}U are inserted without fuel shuffling over the next five operating cycles, the core could operate for an additional 14 years until October 2020.
- The fresh LEU WWR-M2 replacement fuel would last longer than the fresh HEU WWR-M2 replacement fuel mainly because the ^{235}U content of the LEU fuel assemblies is higher than the amount needed to simply match the operating lifetime of the HEU fuel assemblies.
- Without further shuffling of the irradiated HEU fuel, incremental insertion of 28 LEU FA would allow operation for almost one year longer than insertion of 36 HEU FA.
- DNRR shutdown margin requirements would be satisfied with incremental insertion of either 36 fresh HEU or 36 fresh LEU fuel assemblies.
- Fast neutron flux performance in the experiment positions will be essentially the same with insertion of either fresh HEU or fresh LEU fuel assemblies.
- Thermal neutron fluxes with insertion of fresh LEU fuel will be lower by 1-5%, depending on the experiment type and location, than with insertion of fresh HEU fuel,

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