RERTR 2010 — 32nd INTERNATIONAL MEETING ON
REDUCED ENRICHMENT FOR RESEARCH AND TEST REACTORS

October 10-14, 2010
SANA Lisboa Hotel
Lisbon, Portugal

NEUTRONICS AND THERMAL HYDRAULICS CALCULATION FOR FULL CORE CONVERSION FROM HEU TO LEU FUEL OF THE DALAT NUCLEAR RESEARCH REACTOR

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Abstract: The Dalat Nuclear Research Reactor (DNRR) is a pool type research reactor which was reconstructed from the 250 kW TRIGA-MARK II reactor. From March 1984, after its reconstruction and upgrading, the DNRR was operating with the nominal power of 500 kW using Russian fuel type WWR-M2 enriched to 36%. In September 2007, in the framework of the RRRFR program, the DNRR was partially converted to LEU fuel and the reactor has been operating safely with the mixed core of HEU and LEU fuels since then. The works for full core conversion of the DNRR have been performing. Contract for Nuclear fuel manufacture and supply for DNRR among JSC TVEL, Moscow, Russia and Vietnam Atomic Energy Institute and Battelle Enery Alliance, LLC, Idaho Falls, USA has been signed. The study for DNRR's full-core conversion to LEU has been undertaking in the framework of technical cooperation between the Vietnam Atomic Energy Institute (VAEI) and the Argonne National Laboratory (ANL) under Reduced Enrichment for Research and Test Reactor (RERTR) programme. This paper presents the results of neutronics and thermal hydraulics calculation for full core conversion from HEU to LEU fuel of the DNRR.

Keywords: HEU, LEU WWR-M2, WIMS-ANL, REBUS-PC, MCNP5, VARI3D, PLTEMP, ONB, DNBR, MFIPR

1. Introduction

The Dalat Nuclear Research Reactor (DNRR) is a 500-kW pool-type research reactor using light water as both moderator and coolant. It was reconstructed from a 250kW TRIGA MARK II loaded with WWR-M2 fuel enriched to 36% and put into operation in 1984[1].

After finishing partial conversion LEU fuel in September 2007, the DNRR has been operating safely with the mixed HEU-LEU fuel core since then[2]. Recently, the feasibility study for full core conversion has been jointly carrying out by Vietnam Atomic Energy Institute (VAEI) and Argonne National Laboratory (ANL). This report shows results of nuclear and thermal-hydraulics analyses for the DNRR loaded with LEU WWR-M2 fuel assemblies.

At the design stage, four core configuration candidates have been investigated for the parameters related to reactor physics and safety including power peaking factors, shutdown margins, neutron flux and utilization (for radioisotope production, beam port experiment, NAA, …) using MCNP5[3], REBUS-PC[4] computer codes. After carefully considering, two core configuration candidates was chosen for detailed neutronics and thermal-hydraulics analyses.

Besides the MCNP and REBUS-PC models were benchmarked to the experimental data and successfully used for DNRR partial core conversion[5,6], in this work, thermal-hydraulics PLTEMP[7] code was validated by comparing the analytical results with the measured data collected on the current Mixed-Core.

To design the LEU cores, the control rod positions, neutron trap, irradiation channels, beryllium reflector were kept unchanged except some rearrangement of blocks beryllium to meet the requirements of safety and reactor utilization.

2. WWR-M2 HEU and LEU Fuel Assembly

Each HEU fuel assembly contains about 40.2 g of U-235 distributed inside three coaxial fuel tubes (elements), of which the outermost one is hexagonal shaped and the two inner ones are circular (see Figure 1). Each fuel element is composed of three layers; the fuel meat has a thickness of 0.7 mm and is cladded by two aluminum alloy layers with thickness of 0.9 mm. The spaces between fuel elements are channels for coolant water. The total length of the fuel assembly is 865 mm, of which the fuelled part is 600 mm.

A LEU (19.75% enriched) fuel assembly contains an average of 49.7 g of U-235 with UO_2-Al dispersion fuel meat. Each of the fuel elements in the HEU and LEU fuel assemblies has the same overall thickness of 2.5 mm, but the LEU fuel meat and cladding thickness are 0.94 mm and 0.78 mm, respectively. Table 1 compares the key design parameters for the WWR-M2 HEU fuel assembly and the LEU fuel assembly

Figure 1. WWR-M2 Fuel Assembly

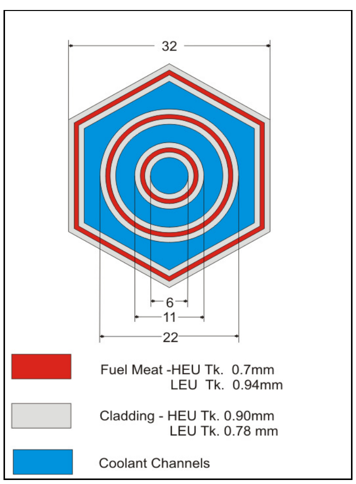

Table 1. Characteristics of WWR-M2 HEU Fuel Assembly and LEU Fuel Assembly

Fuel Assembly Parameter	VVR-M2 HEU	VVR-M2 LEU
Enrichment, %	36.0	19.75
Average mass of ^{235}U in FA, g	40.2	49.7
Fuel meat composition	U-Al Alloy	UO_2+Al
Uranium density of fuel meat, g/cm^3	1.4	2.5
Cladding material	SAV-1	SAV-1
Fuel element thickness, mm	2.5	2.5
Fuel meat thickness, mm	0.7	0.94
Fuel cladding thickness, mm	0.9	0.78

3. Calculation models

Diffusion code REBUS-PC with FD flux solution method was used to model hexagonal-Z multigroup for DNRR. Micro cross section with 7 groups for REBUS-PC was generated by WIMSD-ANL[8] with super-cell options for fuel assemblies and other components. Burn up calculation was calculated by the code to compare with obtained results from REBUS-MCNP linkage[9] two ways (MCNP is used for calculate neutron flux and cross section in 1 group neutron energy and burn up calculation is implemented by REBUS-PC). Prompt neutron life time and effective delayed neutron fraction were estimated by VARI3D and MCNP5 Codes.

MCNP code was used to calculate detail neutron flux distribution, peaking factor following radial of hottest fuel assembly, temperature coefficients and reactor kinetics parameters. True geometry of fuel and other components (control rods, neutron trap, beam tubes, reflectors …) inside reactor core are modelled except top and bottom fuel assemblies because of complicated geometry and the parts were modelled with material homogenized between light water and aluminum.

For burn up calculation, each fuel assembly was divided to 5 depletion nodes with 12 cm length each node and fuel depletion chains included production of six Pu isotopes, Am-241, Np-237 and lumped fission product. Both method calculation about fuel burn up by diffusion code REBUS-PC code and transport code REBUS-MCNP linkage system code were done together and the different of both system codes was acceptable.

Nuclear cross sections to be serve for calculation were based on ENDF-B/VI cross-section library in diffusion code (REBUS-PC) and transport code (MCNP5) also.

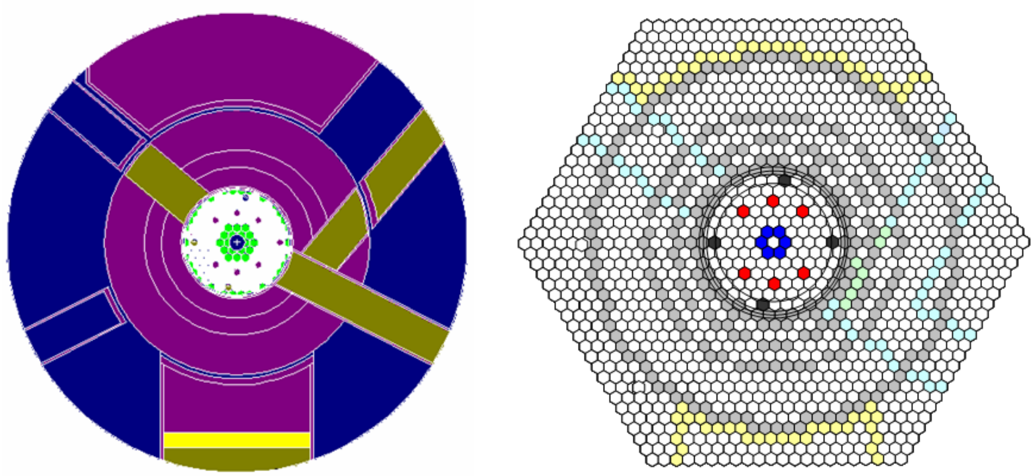

Figure 2. The DNRR was modelled by MCNP and REBUS-PC Codes

In thermal-hydraulics analyses, the fuel assembly was modelled for PLTEMP code as three coaxial tubes. The Collier heat transfer correlation was chosen for DNRR natural convection regime.

4. Results and discussions

4.1. Nuclear Analyses

The working LEU core design commenced by considering four candidate cores with different arrangement of fuel assemblies in the reactor core were established basing on constraint about safety, utilization. All candidate cores are loaded by 92 LEU fuel assemblies for first cycle. Figure 3 shows detailed each core arrangement. In these candidate cores, there are some rearrangement of beryllium blocks to get a new neutron trap except core 3 with original neutron trap. Dry irradiation channel 7-1 and wet irradiation channel 1-4 and 13-2 are kept for neutron activation analysis (NAA) or other application.

From the calculation results of shutdown margins, excess reactivities, power peaking factors, and neutron performance at the irradiation positions, core number 1 and number 2 have the better features from the safety and utilization point of view and were chosen for detailed analysis.

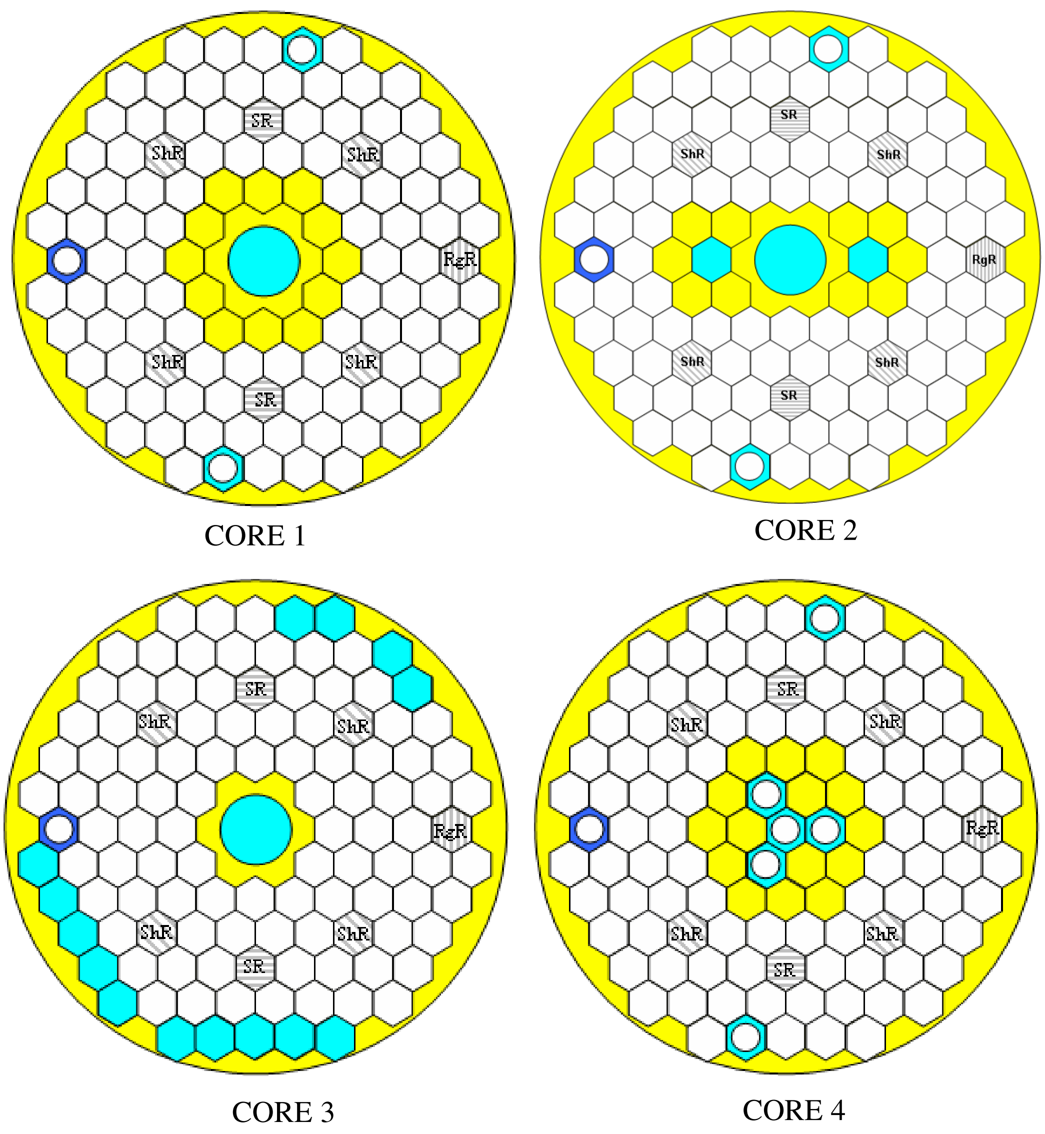

Figure 3. Fort candidate cores from 1 to 4

The main calculated characteristics of LEU core 1 and core 2 are showed in the Table 2. In the safety point of view, core 1 is better than core 2 for larger shutdown margin and smaller power peaking factors. The shutdown margins of the two cores are met the safety requirement of -1.0%. Calculated neutron flux at the neutron trap of the two LEU core are nearly the same with that of current mixed core. Core 2 is better core 1 in the utilization point of view because there are more irradiation positions. Table 3 show the control rod worths. Detailed neutron flux performance at the main irradiation positions are presented in Table 4.

Table 2. Calculated characteristics of LEU core 1 and core 2

Parameters	Core 1	Core 2	Current Mixed Core
Excess Reactivity (%) - Fresh	6.63	6.78	
Excess Reactivity (%) – After 600FPDs	3.79	3.89	
Shutdown Margin (%) – Fresh	-2.92	-2.58	-4.56
Shutdown Margin (%)- After 600 FPDs	-6.62	-6.29	
Radial Power Peaking Factor			
Control Rods Out	1.398	1.474	1.431
Control Rods In	1.434	1.561	
Thermal Neutron Flux at Neutron Trap Center (n/cm^2)			
Control Rods Out	2.22E+13	2.21E+13	2.22E+13
Control Rods In	2.14E+13	2.15E+13	
Fast Neutron Flux at Neutron Trap Center (n/cm^2)			
Control Rods Out	1.95E+12	2.48E+12	3.15E+12
Control Rods In	1.92E+12	2.47E+12	

Table 3. Control Rods worths (%) of Core1 and Core2

Control Rods	Core1 Fresh	MCNP error	Core1 Burnt	MCNP error	Core2 Fresh	MCNP error	Core2 Burnt	MCNP error
Sh1	2.5896	0.000091	2.3539	0.000091	2.5647	0.000098	2.3156	0.000098
Sh2	2.6100	0.000111	2.4033	0.000124	2.5873	0.000094	2.3326	0.000128
Sh3	2.7784	0.000118	2.5381	0.000122	2.7344	0.000097	2.4765	0.000117
Sh4	2.4687	0.000122	2.2604	0.000117	2.4626	0.000128	2.2865	0.000132
ReR	0.4363	0.000126	0.3629	0.000119	0.3621	0.000124	0.3606	0.000130
Sa1	2.1955	0.000106	2.3084	0.000115	2.2589	0.000099	2.2535	0.000125
Sa2	2.2356	0.000119	2.3579	0.000105	2.2897	0.000102	2.3127	0.000122

Table 4. Neutron flux performance of Core1 and Core2

CORE 1							
		Thermal (n/cm^2.s) <0.625eV		**Epithermal (n/cm^2.s) <0.821MeV**		**Fast (n/cm^2.s) <10MeV**	
		Fresh	**Burnt**	**Fresh**	**Burnt**	**Fresh**	**Burnt**
Neutron Trap	**Maximum**	2.07E+13	2.20E+13	6.79E+12	7.12E+12	1.83E+12	1.92E+12
	Average	1.45E+13	1.49E+13	6.00E+12	6.04E+12	1.62E+12	1.63E+12
Channel 13-2	**Maximum**	9.45E+12	9.86E+12	8.19E+12	8.42E+12	2.98E+12	3.02E+12
	Average	7.00E+12	7.12E+12	6.53E+12	6.51E+12	2.46E+12	2.44E+12
Channel 7-1	**Maximum**	5.41E+12	5.66E+12	9.63E+12	9.76E+12	4.22E+12	4.26E+12
	Average	4.11E+12	4.18E+12	7.23E+12	7.15E+12	3.19E+12	3.15E+12
Channel 1-4	**Maximum**	9.24E+12	9.71E+12	8.02E+12	8.22E+12	2.92E+12	2.99E+12
	Average	6.85E+12	7.01E+12	6.41E+12	6.40E+12	2.42E+12	2.40E+12

Rotary Specimen	**Average**	3.55E+12	3.56E+12	7.58E+11	7.56E+11	1.93E+11	1.93E+11
CORE 2							
		Thermal (n/cm^2.s) <0.625eV		**Epithermal (n/cm^2.s) <0.821MeV**		**Fast (n/cm^2.s) <10MeV**	
		Fresh	**Burnt**	**Fresh**	**Burnt**	**Fresh**	**Burnt**
Neutron Trap	**Maximum**	2.06E+13	2.18E+13	7.75E+12	8.07E+12	2.31E+12	2.41E+12
	Average	1.41E+13	1.44E+13	6.74E+12	6.75E+12	2.09E+12	2.09E+12
Channel 13-2	**Maximum**	9.37E+12	9.87E+12	8.24E+12	8.39E+12	2.96E+12	3.02E+12
	Average	6.98E+12	7.12E+12	6.53E+12	6.51E+12	2.47E+12	2.44E+12
Channel 7-1	**Maximum**	5.32E+12	5.60E+12	9.30E+12	9.44E+12	4.05E+12	4.08E+12
	Average	4.08E+12	4.15E+12	7.03E+12	6.91E+12	3.09E+12	3.01E+12
Channel 1-4	**Maximum**	9.16E+12	9.60E+12	8.01E+12	8.25E+12	2.91E+12	2.98E+12
	Average	6.83E+12	6.98E+12	6.40E+12	6.40E+12	2.42E+12	2.41E+12
Hole – Right (7-8)	**Average**	1.10E+13	1.14E+13	7.44E+12	7.52E+12	2.15E+12	2.17E+12
Hole – Left (7-4)	**Average**	1.13E+13	1.17E+13	7.70E+12	7.74E+12	2.23E+12	2.23E+12
Rotary Specimen	**Average**	3.52E+12	3.51E+12	7.52E+11	7.43E+11	1.92E+11	1.90E+11

Power peaking factors of both candidate cores with different position of control rods were calculated and presented in Table 5. The maximum power peaking factor of core 1 and core 2 is in position of control rods at 250 mm and 300mm respectively.

Table 5. Power peaking factor depend on control rod positions

Position (mm)	**Peaking Factor**			
	F.A. Radial	**Core Radial**	**Axial**	**Total**
CORE1				
0	1.378	1.398	1.296	2.498
150	1.378	1.399	1.343	2.589
200	1.375	1.403	1.356	2.615
250	1.377	1.409	1.365	2.648
300	1.376	1.411	1.363	2.646
350	1.378	1.415	1.336	2.605
600	1.378	1.434	1.284	2.537
CORE2				
0	1.492	1.474	1.281	2.817
150	1.492	1.489	1.335	2.966
200	1.493	1.495	1.341	2.992
250	1.491	1.503	1.356	3.039
300	1.492	1.509	1.364	3.071
350	1.497	1.520	1.330	3.024
600	1.496	1.561	1.283	2.994

Reactivity feedback coefficients calculated with the MCNP5 code show only small differences between fresh and burnt core and between Core 1 and Core 2 (see Tables 6). The negative results of reactivity feedback coefficients show the inherent safety of the LEU core. Table 7 shows the kinetics parameters of the LEU cores calculated using the VARI3D and MCNP5 codes. The results obtained from the two computer code are in good agreement. These data will be used in transient calculation for safety analysis of fully LEU core of DNRR.

Table 6. Feedback reactivity coefficients of Core 1 and Core 2 by MCNP5 (fresh core)

Parameter	Core 1		Core 2	
	DATA	±σ	DATA	±σ
Moderator Temperature Reactivity Coefficient (%/°C)				
293 °K to 400 °K	-0.01317	0.00005	-0.01367	0.00004
Fuel Temperature (Doppler) Reactivity Coefficient (%/°C)				
293 °K to 400 °K	-0.00192	0.00005	-0.00190	0.00004
400 °K to 500 °K	-0.00182	0.00003	-0.00168	0.00004
500 °K to 600 °K	-0.00154	0.00002	-0.00173	0.00004
Moderator Density (Void) Reactivity Coefficient (%/% of void)				
0 to 5 %	-0.2514	0.0011	-0.2453	0.0008
5% to 10 %	-0.2784	0.0012	-0.2702	0.0004
10 % to 20 %	-0.3255	0.0006	-0.3122	0.0002

Table 7. Kinetics parameters of Core 1 and Core 2

Group Delayed Neutron Data						
Family, i	Core 1			Core 2		
	Decay Const. λ_i (s^{-1})	Relative Yield a_i	Fraction β_i	Decay Const. λ_i (s^{-1})	Relative Yield a_i	Fraction β_i
1	1.334E-02	3.507E-02	2.648E-04	1.334E-02	3.507E-02	2.661E-04
2	3.273E-02	1.804E-01	1.363E-03	3.273E-02	1.804E-01	1.372E-03
3	1.208E-01	1.742E-01	1.315E-03	1.208E-01	1.742E-01	1.317E-03
4	3.030E-01	3.843E-01	2.902E-03	3.030E-01	3.843E-01	2.906E-03
5	8.503E-01	1.594E-01	1.204E-03	8.503E-01	1.594E-01	1.205E-03
6	2.856E+00	6.666E-02	5.033E-04	2.856E+00	6.666E-02	5.048E-04
Total delayed neutron fraction, β VARI3D			7.551E-03			7.580E-03
MCNP5 – Fresh MCNP5 - Burnt			7.761E-03 7.762E-03			7.778E-03 7.773E-03
Prompt neutron life time, ℓ			8.925E-05			8.649E-05

The first cycle length of both cores number 1 and 2 were estimated by REBUS-MCNP Linkage system codes. Burn up calculations were performed assuming shim rods and regulating rod were in critical position following each burn up step. The value of reactivity for Xe-135 poisoning was estimate about 1.2% Δk/k for both cores. The result of depletion in both core shows that operating time will be extended about 11 years (calculated with 1300 hours per year) or 600 full power days (fpds). The burn up of U-235 in both core reached average value of 8.2% and maximum value of 11.4%. The discrepancy of calculated results from both codes has average about 3% and maximum about 8% of some fuel assemblies located at periphery of the reactor core. In the next cycle, number fuel assemblies will be inserted about 8 so the reactor core will operate with 100 fuel assemblies.

4.2. Thermal-Hydraulics Analyses

The PLTEMP/ANL3.8 thermal-hydraulics code for plate and concentric-tube geometries with capability of calculating natural circulation flow was used for thermal-hydraulics analyses. A chimney model as well as Collier heat transfer correlation and CHF Shah's correlation were newly implemented make the code suitable DNRR thermal-hydraulics calculation.

Before using PLTEMP code to calculation for DNRR with fully LEU fuel assemblies, the code was validated by comparing analytical results with experimental results of current mixed-core.

The PLTEMP code was then used for calculating cladding temperature, coolant temperature and safety margins for the candidate cores. The calculated results of the core1 and core2 are presented in Tables 8 and 9. At nominal power without uncertainties and maximum permissible inlet temperature (32^{o}C), the maximum cladding temperature of core1 is 90.50^{o}C and that of core2 is 96.41^{o}C. Calculation was carried out for nominal power with systematic errors (equivalent to 70kW power) and the maximum cladding temperature for core1 and core2 are 95.69^{o}C and 102.06^{o}C respectively. In this case, by using Shah's correlation, the obtained minimum DNBR is 9.9 for core1 and 8.7 for core2. The minimum flow instability power ratio (MFIPR) are 2.04 and 1.85 for core1 and core2 respectively. From above-mentioned calculated results, it is concluded that the 2 candidate cores meet the requirements of thermal hydraulics safety. At the power of 500kW with systematic errors, maximum cladding temperatures are below the permissible value of 103^{o}C and far below the ONB temperature (estimated about 116^{o}C using Forster-Greif correlation). The maximum outlet coolant temperature is calculated about 60^{o}C, much lower than saturated temperature (108^{o}C).

Table 8. Cladding temperature and ONB margin of Core 1 by PLTEMP Code

Distance (cm)	**500kW**				**550kW**		**600kW**	
	without sys. error		**with sys. error**		**with sys. error**		**with sys. error**	
	Tc(^{o}C)	**ΔT-ONB(^{o}C)**	**Tc(^{o}C)**	**ΔT-ONB(^{o}C)**	**Tc(^{o}C)**	**ΔT-ONB(^{o}C)**	**Tc(^{o}C)**	**ΔT-ONB(^{o}C)**
2.5	63.91	51.89	66.89	49.24	68.95	47.39	70.96	45.59
7.5	70.56	45.59	74.13	42.36	76.58	40.14	78.97	37.97
12.5	78.46	38.07	82.71	34.18	85.63	31.51	88.46	28.91
17.5	84.83	31.90	89.61	27.50	92.89	24.48	96.05	21.57
22.5	88.77	27.95	93.85	23.26	97.33	20.06	100.68	16.95
27.5	90.50	26.05	95.69	21.25	99.23	17.97	102.65	14.80
32.5	89.86	26.34	94.95	21.63	98.43	18.41	100.76	16.40
37.5	87.10	28.58	91.91	24.13	94.41	21.94	96.22	20.48
42.5	83.98	31.14	88.24	27.24	89.94	25.90	91.57	24.60
47.5	79.67	34.76	82.92	31.91	84.43	30.74	85.89	29.59
52.5	74.91	38.73	77.42	36.64	78.79	35.57	80.13	34.52
57.5	71.21	41.70	73.32	40.02	74.64	38.98	75.94	37.93

Table 9. Cladding temperature and ONB margin of Core 2 by PLTEMP Code

Distance	500kW				550kW		600kW	
	without sys. error		with sys. error		with sys. error		with sys. error	
(cm)	Tc(°C)	ΔT-ONB(°C)	Tc(°C)	ΔT-ONB(°C)	Tc(°C)	ΔT-ONB(°C)	Tc(°C)	ΔT-ONB(°C)
2.5	67.69	48.48	71.02	45.49	73.32	43.42	75.55	41.41
7.5	75.75	40.84	79.78	37.18	82.55	34.65	85.24	32.20
12.5	84.99	32.03	89.81	27.61	93.11	24.58	96.30	21.64
17.5	91.33	25.86	96.64	20.96	100.27	17.61	103.78	14.36
22.5	95.43	21.74	101.04	16.55	104.87	13.00	108.58	9.56
27.5	96.41	20.53	102.06	15.29	105.92	11.71	108.34	9.64
32.5	95.31	21.24	100.83	16.12	103.27	14.05	105.49	12.18
37.5	93.33	22.77	97.66	18.90	99.78	17.16	101.83	15.45
42.5	89.87	25.65	93.01	23.04	94.91	21.50	96.75	19.99
47.5	85.16	29.72	87.66	27.75	89.37	26.38	91.41	24.62
52.5	79.52	34.60	81.78	32.84	83.34	31.59	85.51	29.63
57.5	74.80	38.56	76.94	36.87	78.52	35.54	80.61	33.64

Figure 4. shows the comparison of cladding temperature of 92FA LEU cores and 89FA fresh HEU core and current 104FA mixed core. Compare to the 89FA fresh HEU core established in 1984, cladding temperature of core1 is lower about 2°C and core2 is higher about 4°C. The cladding temperature of current mixed core is lower than 89FA HEU Core and 92FA LEU Cores because number of Fuel Assemblies and burn up effective.

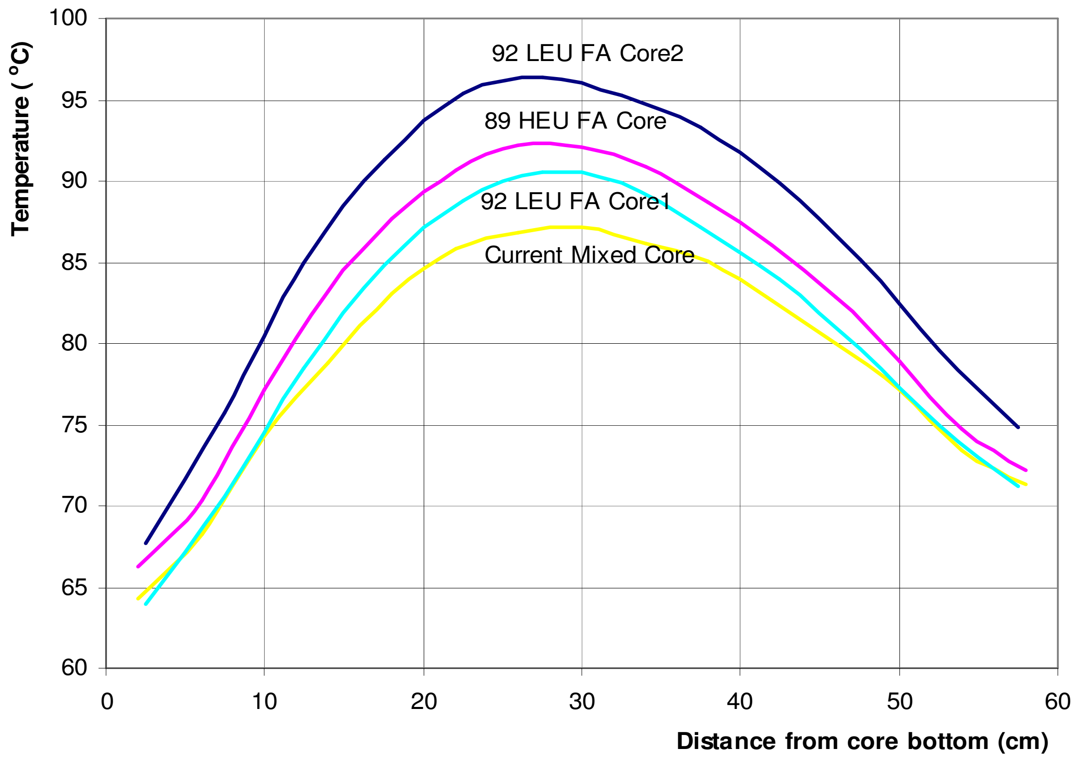

Figure 4. Comparison of calculated cladding temperature between 92FA LEU cores and HEU cores

5. Conclusions

Neutronics and steady-state thermal-hydraulic analyses for Dalat Nuclear Research Reactor show that with a little change in arrangement of Be rods, the main features of 92 LEU WWR-M2 FA cores are equivalent to those of HEU and current mixed fuel cores.

The negative values of reactivity feedback coefficients show the inherent safety feature and shutdown margin of both candidate cores meets the safety required value of -1% $\Delta k/k$. The first working core with 92 fresh LEU fuel assemblies can be operated for 600FPd or about 11 years based on the current operating schedule without shuffling. The neutron fluxes at the irradiation positions are not much different with those of the current mixed fuel core. Safety characteristics of Core1 is better than Core2 but Core2 has higher potential utilization because of more irradiation positions at the core center.

In thermal hydraulics aspect, the requirement of thermal-hydraulic safety margin for two candidate cores in normal operational condition is satisfied. The calculated maximum cladding temperature in operational condition is below the permissible value of 103^{o}C.

Acknowledgement*: The authors would like to thank Dr. Roglans-Ribas J., Dr. Hannan N. A., Dr. Garner P. L., Dr. Matos J and RERTR staff of ANL for kindness helps and valuable discussions of the work done.*

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