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PRELIMINARY RESULTS OF FULL CORE CONVERSION FROM HEU TO LEU FUEL OF THE DALAT NUCLEAR RESEARCH REACTOR

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ABSTRACT

This paper presents preliminary results of design calculation for full core conversion from HEU (High Enriched Uranium) to LEU (Low Enriched Uranium) of the Dalat Nuclear Research Reactor (DNRR). From four candidate cores, some main parameters related to safety, utilization were investigated for determining better ones. Two candidate cores were chosen for detailed analysis about neutronics and thermal hydraulics. The reactor physics parameters like neutron flux, power distribution, excess reactivity, temperature coefficients, kinetics parameters, fuel burn up and fuel cladding temperature were calculated to serve about safety, operation and utilization of DNRR based on the two candidate cores. By using 92 fresh fuel assemblies for working core, the first reactor operation cycle can be predicted around 11 years with 1300 hours operation full power per year. The computer codes were used for neutronics calculation includes WIMS-ANL, REBUS-PC, MCNP5 and VARI3D. For thermal hydraulics analysis, PLTEMP code was used to estimate cladding temperature, coolant temperature and ONB margin at steady state condition.

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 preliminary 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 2). 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

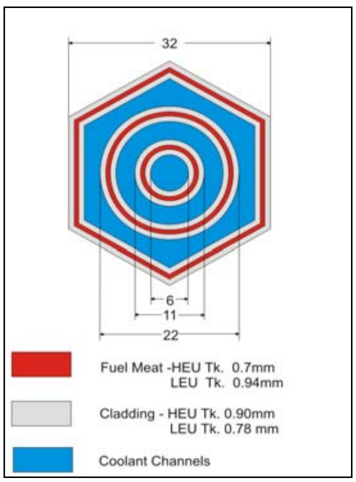

Table 1. Characteristics of WWR-M2 HEU Fuel Assembly and LEU Fuel Assembly

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.

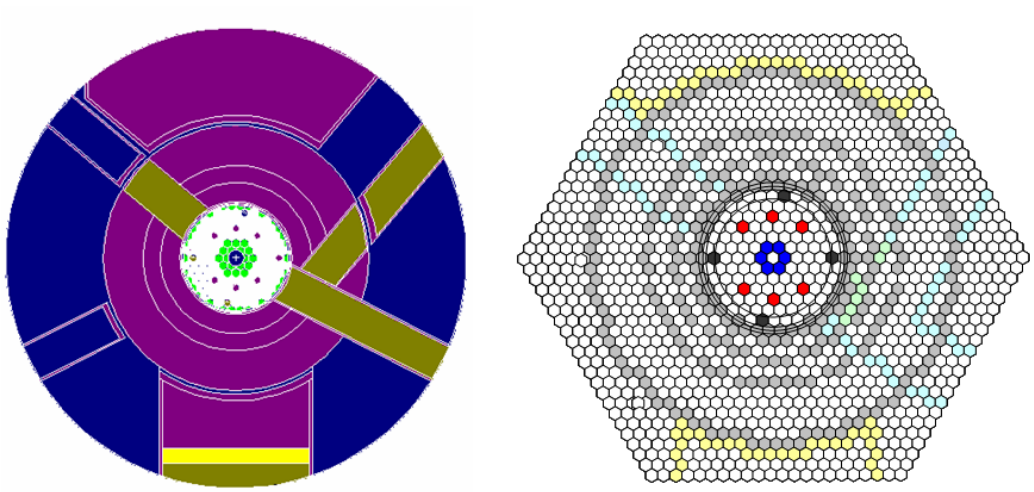

Figure 2. The DNRR was modelled by MCNP and REBUS-PC Codes

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

First of all, 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. Core number 1 and 2 still kept neutron trap but core number 3 and 4 the neutron trap was modified for irradiation purpose. In the Figure 3 shows detailed each core arrangement. White colour is fuel assembly, yellow is beryllium and blue is water channel only. Dry irradiation channel 7-1 and wet irradiation channel 1-4 and 13-2 are kept for NAA or other application.

The calculation results of four candidate cores was depicted in Table 2. From these results, it can be found that core number 1 and 4 meet requirements in safety with highest values of shutdown margin and small power peaking factor, and in utilization with good enough for excess reactivity and neutron flux distribution at neutron trap and rotary specimen.

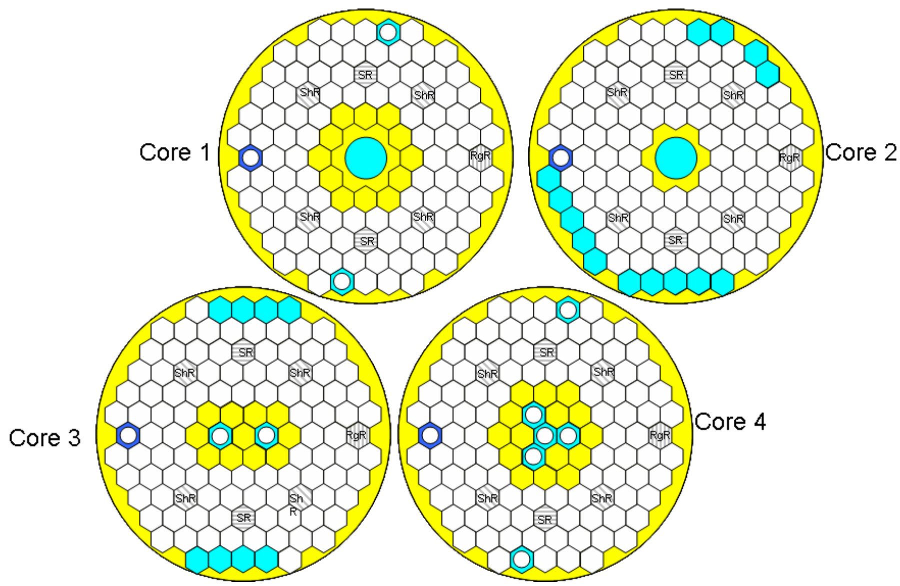

Figure 3. Fort candidate cores from 1 to 4

Table 2. Calculation results of candidate cores at first step

Parameters	Core 1	Core 2	Core 3	Core 4
Excess Reactivity (%)	7.44	7.40	8.20	7.57
Shutdown Margin (%)	-2.38	-1.52	-1.14	-2.20
Radial Power Peaking Factor				
Control Rods Out	1.405	1.451	1.490	1.424
Control Rods In	1.463	1.580	1.602	1.482
Thermal Neutron Flux at Neutron Trap Center (n/cm^2)				
Control Rods Out	2.28E+13	2.24E+13	1.92E+13	2.01E+13
Control Rods In	2.21E+13	2.27E+13	1.88E+13	1.92E+13
Fast Neutron Flux at Neutron Trap Center (n/cm^2)				
Control Rods Out	1.95E+12	3.42E+12	3.08E+12	2.41E+12
Control Rods In	1.94E+12	3.52E+12	3.12E+12	2.36E+12
Average Thermal Neutron Flux at Rotation (n/cm^2)				
Control Rods Out	3.81E+12	3.47E+12	3.63E+12	3.80E+12
Control Rods In	3.85E+12	3.45E+12	3.64E+12	3.85E+12

The first cycle length of both cores number 1 and 4 were estimated by REBUS-PC and REBUS-MCNP Linkage system codes. Burn up calculation by both codes 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% for both cores. The result of depletion in both core show 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 8.3% and maximum value of 11.7%. 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. The Figure 4 shows burn up distribution in both core with two results from different codes.

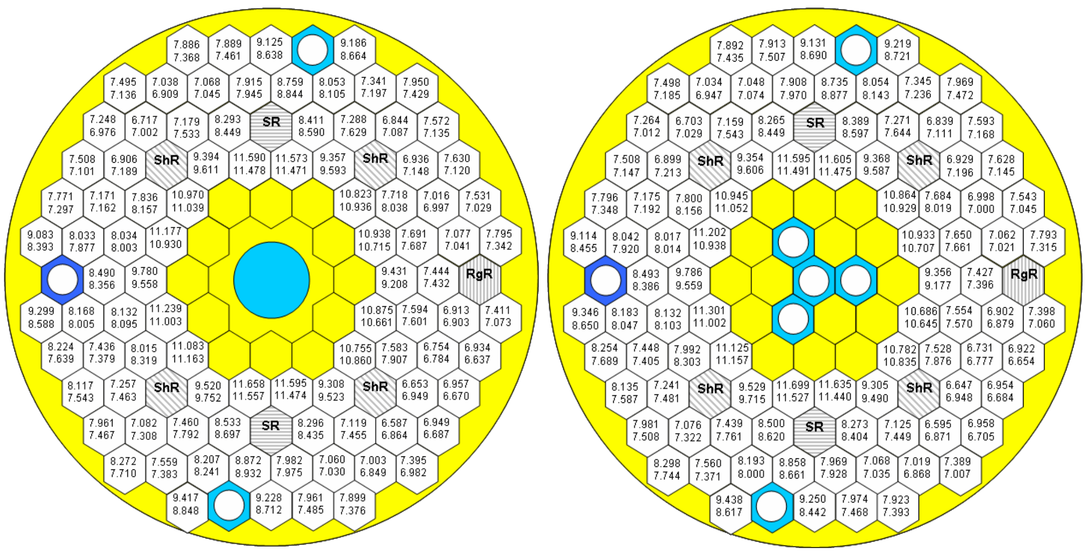

Figure 4. Burn up distribution of candidate cores number 1 and 4 (upper values from REBUS-MCNP Linkage system and under values from REBUS-PC)

Power peaking factors of both candidate cores with different position of control rods were calculated and presented in Table 3. The maximum power peaking factor of both cores is in position of control rods at 300mm.

Table 3. Power peaking factor following control rod positions

Position (mm)	Peaking Factor			
	F.A. Radial	Core Radial	Axial	Total
CORE1				
0	1.404	1.407	1.275	2.517
200	1.443	1.423	1.388	2.849
250	1.457	1.427	1.415	2.943
300	1.467	1.431	1.428	2.998
350	1.485	1.436	1.400	2.987
400	1.504	1.441	1.334	2.893
500	1.540	1.450	1.237	2.761
CORE4				
0	1.424	1.423	1.276	2.587
200	1.463	1.440	1.385	2.917
250	1.476	1.445	1.415	3.018
300	1.489	1.449	1.427	3.077
350	1.508	1.453	1.398	3.064
400	1.523	1.459	1.334	2.966
500	1.555	1.468	1.238	2.826

Reactivity feedback coefficients were calculated by both MCNP and REBUS codes with a little differences (see Table 4). The negative results of reactivity feedback coefficients show the inherent safety of the LEU core. Table 5 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 4. Feedback reactivity coefficients of Core 1 and Core 4 (fresh core)

Parameter	Core 1		Core 4	
	MCNP	REBUS	MCNP	REBUS
Moderator Temperature Reactivity Coefficient (%/^{o}C)				
296 ^{o}K to 350 ^{o}K		-0.0098		-0.0103
350 ^{o}K to 400 ^{o}K		-0.0099		-0.0104
293 ^{o}K to 400 ^{o}K	-0.0119	-0.0099	-0.0119	-0.0103
Fuel Temperature (Doppler) Reactivity Coefficient (%/^{o}C)				
293 ^{o}K to 400 ^{o}K	-0.0019		-0.0020	
400 ^{o}K to 500 ^{o}K	-0.0018		-0.0017	
500 ^{o}K to 600 ^{o}K	-0.0016		-0.0017	
300 ^{o}K to 560 ^{o}K		-0.0017		-0.0018
Moderator Density (Void) Reactivity Coefficient (%/% of void)				
0 to 5 %	-0.256	-0.213	-0.258	-0.224
5% to 10 %	-0.282	-0.233	-0.279	-0.244
10 % to 20 %	-0.322	-0.267	-0.323	-0.280

Table 5. Kinetics parameters of Core 1 and Core 4 (fresh core)

Family, i	Core 1			Core 4		
	λ_i	a_i	β_i	λ_i	a_i	β_i
1	1.334E-02	3.507E-02	2.648E-04	1.33E-02	3.51E-02	2.66E-04
2	3.273E-02	1.804E-01	1.363E-03	3.27E-02	1.80E-01	1.37E-03
3	1.208E-01	1.742E-01	1.315E-03	1.21E-01	1.74E-01	1.32E-03
4	3.030E-01	3.843E-01	2.902E-03	3.03E-01	3.84E-01	2.91E-03
5	8.503E-01	1.594E-01	1.204E-03	8.50E-01	1.59E-01	1.21E-03
6	2.856E+00	6.666E-02	5.033E-04	2.86E+00	6.67E-02	5.05E-04
Total delay neutron fraction, **β**			7.551E-03			7.580E-03
MCNP5			7.780E-03			7.880E-03
Prompt neutron life time, **ℓ**			8.795E-05			8.795E-05
β / ℓ			8.586E+01			8.619E+01

4.2. Thermal-Hydraulics Analyses

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. A good agreement between calculated data using Collier heat transfer correlation and measured data was shown in Figure 5.

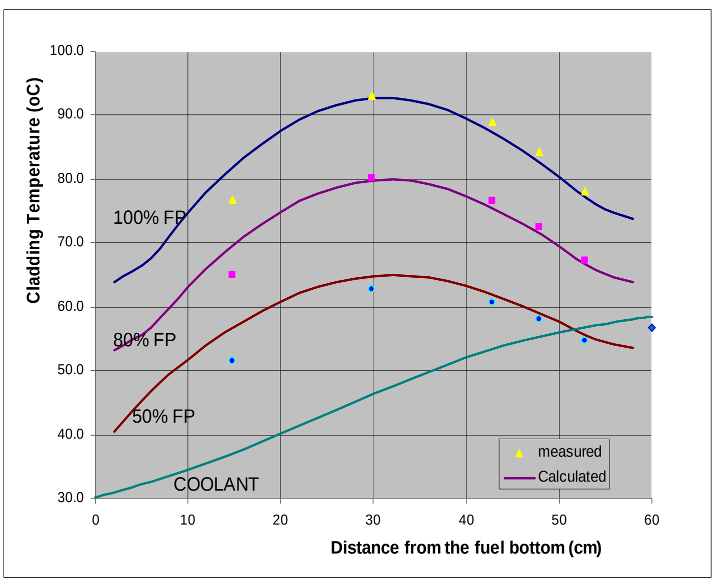

Figure 5. Comparison of cladding and coolant temperatures of current reactor core (98 HEU+6LEU) by PLTEMP code and experimental data

The PLTEMP code was then used for calculating cladding temperature, coolant temperature and ONB margin for the candidate cores. The calculated results of the core1 and core4 are presented in Tables 6, 7 and Figures 6, 7. At nominal power without uncertainties (best estimate), the maximum cladding temperature of core1 is 94.65^{o}C and that of core4 is 95.87^{o}C. Calculation was carried out for nominal power with systematic errors (estimated about 70kW) and the maximum cladding temperature for core1 and core4 are 100.20°C and 101.52°C respectively. In this case, by using Shah correlation, the obtained minimum DNBR is 8.9 for core1 and 8.7 for core4. The minimum flow instability power ratio (MFIPR) are 1.95 and 1.92 for core1 and core2 respectively. From above-mentioned calculated results, it is concluded that the 2 candidate cores meet the requirement of thermal hydraulics safety. At the power of 500kW with systematic errors, maximum cladding temperatures are below the permissible value of 103°C[10] and far below the ONB temperature (estimated about 116°C using Forster-Greif correlation). The maximum outlet coolant temperature is calculated about 60°C, much lower than saturated temperature (108°C).

Table 6. Cladding temperature and ONB margin of Core 1 by PLTEMP Code

Distance	500kW		500Kw+Sys. Error		500kW+S.E.+50kW	
(cm)	Tc(°C)	D-ONB(°C)	Tc(°C)	D-ONB(°C)	Tc(°C)	D-ONB(°C)
2.5	69.74	46.59	73.27	43.41	75.69	40.99
7.5	76.37	40.24	80.46	36.52	83.27	33.78
12.5	84.96	32.03	89.78	27.61	93.08	24.44
17.5	91.22	25.94	96.52	21.04	100.15	17.61
22.5	94.65	22.44	100.20	17.31	103.99	13.79
27.5	94.47	22.30	99.96	17.22	103.71	13.82
32.5	90.16	25.98	95.25	21.27	97.73	19.19
37.5	87.21	28.40	91.48	24.54	93.34	23.05
42.5	84.11	30.95	87.37	28.15	89.06	26.80
47.5	80.16	34.24	82.61	32.28	84.15	31.11
52.5	75.32	38.32	77.37	36.75	78.79	35.80
57.5	71.63	41.34	73.60	39.81	74.99	39.04

Table 7. Cladding temperature and ONB margin of Core 4 by PLTEMP Code

Distance	500kW		500Kw+Sys. Error		500kW+SE+50kW	
(cm)	Tc(°C)	D-ONB(°C)	Tc(°C)	D-ONB(°C)	Tc(°C)	D-ONB(°C)
2.5	70.31	45.88	73.89	42.85	76.34	40.39
7.5	77.24	39.31	81.41	35.65	84.27	32.85
12.5	86.03	30.95	90.94	26.53	94.29	23.30
17.5	92.38	24.83	97.77	19.88	101.45	16.38
22.5	95.87	21.35	101.52	16.08	105.37	12.49
27.5	95.58	21.39	101.17	16.09	104.90	12.70
32.5	91.32	25.14	96.50	20.10	98.71	18.25
37.5	88.45	27.59	92.51	23.62	94.40	22.05
42.5	85.29	30.29	88.35	27.28	90.07	25.85
47.5	81.22	33.83	83.53	31.48	85.10	30.22
52.5	76.05	38.34	78.15	36.07	79.60	35.04
57.5	72.21	41.61	74.23	39.26	75.64	38.42

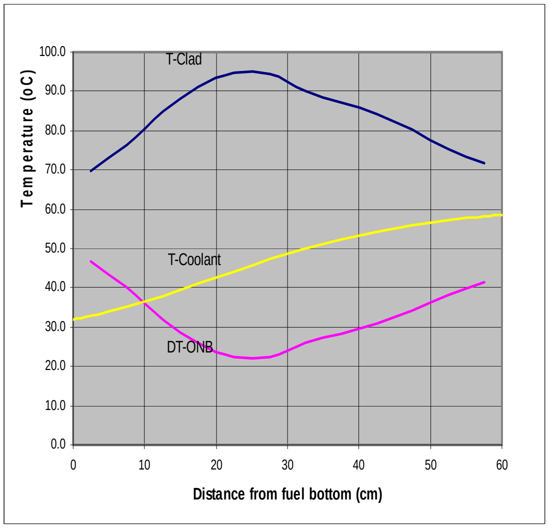

Figure 6. T/H parameters of core1 at 500kW without uncertainties

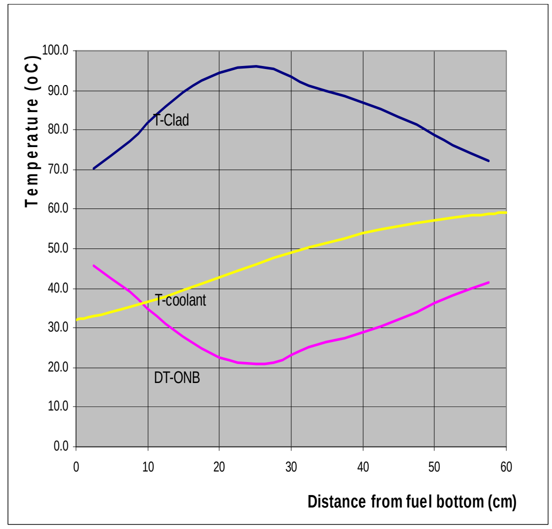

Figure 7. T/H parameters of core4 at 500kW without uncertainties

5. Conclusions

Based on the neutronics analyses, it can be concluded that the DNRR will be safely operated with the LEU cores loaded with 92 fuel assemblies while the utilization conditions are nearly the same compared to the current mixed fuel core. 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 flux at the irradiation positions are not much different with those of the current mixed fuel core.

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.

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