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Thermal-hydraulics analysis for VVR-KN fuel lead test using PLTEMP code

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Abstract: VVR-KN is one of the low-enriched fuel types to be considered for a new research reactor (RR) of a Centre for Nuclear Energy Science and Technology (CNEST) of Vietnam. This fuel type was qualified by a lead test carried out with three fuel assemblies (FAs) in 6-MWt WWR-K research reactor at the Institute of Nuclear Physics, Kazakhstan. VVR-KN fuel was then used for conversion of the WWR-K reactor core from highly-enriched to low-enriched uranium fuel and the reactor was successfully commissioned in September 2016. PLTEMP is a thermal-hydraulic code with plate and coaxial tube models that seems to be suitable for VVR-KN fuel type. Before using PLTEMP code for thermal-hydraulics analysis of the new RR, a calculation for code validation was performed based on the data of the VVR-KN fuel lead test. First, MCNP5 code was used to calculate the power distribution of WWR-K reactor core with lead test fuel assemblies (LTAs) at the core center. Then, thermal-hydraulics parameters of the LTAs were obtained by using PLTEMP code together with calculated data of the power distribution and the lead test conditions. A comparison between the analytic results and the lead test data was made to confirm the suitability of PLTEMP code for thermal-hydraulics analysis of VVR-KN fuel under forced convection and downward flow conditions.

Keywords: *VVR-KN fuel type, MCNP5, PLTEMP, WWR-K reactor, VVR-KN fuel lead test.*

I. INTRODUCTION

The CNEST project with a 10-MWt upgradable to 15-MWt research reactor (RR) has been in the pre-feasibility study phase. Meanwhile, the national research project for calculation of neutronic characteristics, thermal-hydraulics and safety analysis of the proposed RR loading with low enriched Russian fuel types has been deployed.

One of selected fuel types intended for this RR is VVR-KN, which is composed of coaxial tubes. Additionally, PLTEMP code [1] with plate and tube models has been used at the Argonne National Laboratory for RRs using Russian coaxial tube fuel type. This code seems to be suitable for thermal-hydraulics calculation of the new RR at steady-state condition.

The PLTEMP code validation with the coaxial fuel type and the natural convection mode was achieved by comparing with the experimental data in the Dalat research reactor [5]. On the other hand, in forced convection mode, the PLTEMP code was validated based on the data of the VVR-KN fuel lead test, which was performed in the 6-MWt WWR-K reactor of Kazakhstan [4]. Successful validation of the PLTEMP code was carried out in both forced and natural convection modes that confirms the suitability for thermal-hydraulics analysis of the new RR not only in normal operation but also in cases the decay

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heat to be removed by natural convection of reactor pool water.

This work includes neutronic and thermal-hydraulics calculations. A spatial thermal power distribution of WWR-K reactor core with experimental VVR-KN FAs is obtained by using MCNP5 code [2]. Thereafter, using PLTEMP code together with the results from the previous neutron calculations and the initial conditions of the fuel lead test to obtain thermal-hydraulics parameters. Eventually, comparison of the calculated results with the experimental data confirms the suitability of the PLTEMP code for VVR-KN fuel under the experimental conditions.

II. METHOD AND CALCULATION RESULTS

A. VVR-KN fuel lead test

VVR-KN fuel assembly (FA) (see **Fig. 1**) is a low-enriched fuel type that was used for the conversion of the WWR-K reactor core from highly-enriched uranium fuel of 36% (HEU) to low-enriched fuel of 19.75%of ^{235}U (LEU) [3]. Also, this fuel is being considered for the new RR of the CNEST in Vietnam.

Table I. reveals main characteristics of the VVR-KN FAs [3].

Table I. Characteristics of VVR-KN FAs.

Fuel composition	UO_2 + Al
Number of fuel elements per FA (standard FA/ FA with control rod inside)	8 tubes / 5 tubes
Fuel cladding material	Al (SAV-1)
Uranium enrichment, %	19.75
Content of ^{235}U per FA, g	245 / 196
Content of ^{235}U per a core volume unit, g/l	104.4 / 83.5
Surface of heat removal per a core volume unit, cm^2/cm^3	5.46 / 4.33
Fuel element thickness, mm	1.6
Thickness of gaps between fuel elements, mm	2
Total weight of FA, kg	4.66 / 4.21
Fuel meat thickness, mm	0.7
Fuel meat length, mm	600
Fuel meat uranium density, g/cm^3	3.0
Volume fraction of water in FA	0.54
Infinite-medium neutron multiplication factor, k_∞	1.648

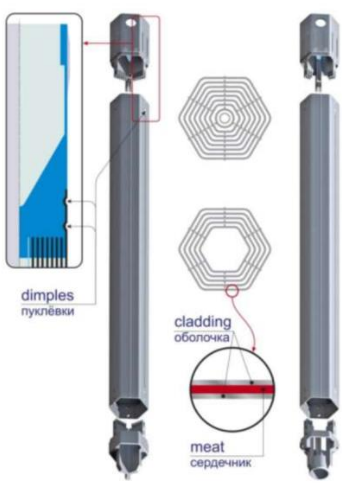

Fig. 1. Two types of VVR-KN FAs.

There are two types of VVR-KN FAs, including the standard one with 8 fuel elements (FEs) and the other with 5 fuel elements for control rod placement inside FA-2 (**Fig. 1**).

For the fuel lead test, the LEU VVR-KN FA's width across flat is 66.3 mm, which makes it possible to be installed into the existing support plate using a triangular lattice with a spacing of 68.3 mm of the HEU VVR-C FA of WWR-K reactor. An increase in the FA robustness in case of beyond design basis accidents was achieved through the longitudinal spacer ribs introduced on the FE outside which ensure the guaranteed clearance between the FEs.

A program was developed for the lead test fuel of the VVR-KN FAs in the WWR-K reactor. An experimental device of three experimental VVR-KN FAs (EFAs) with beryllium blocks around was placed in the reactor core center (**Fig. 2**). The WWR-K reactor can reach at the permitted thermal power of up to 6 MWt. Its core consisted of 38 HEU VVR-C FAs, including 32 FAs with 5 FEs (FA-1) and 6 FAs with 3 FEs (FA-2) shown in **Fig. 2.** To ensure the required reactivity margin during the test, hexagonal beryllium blocks were installed into the core's peripheral cells.

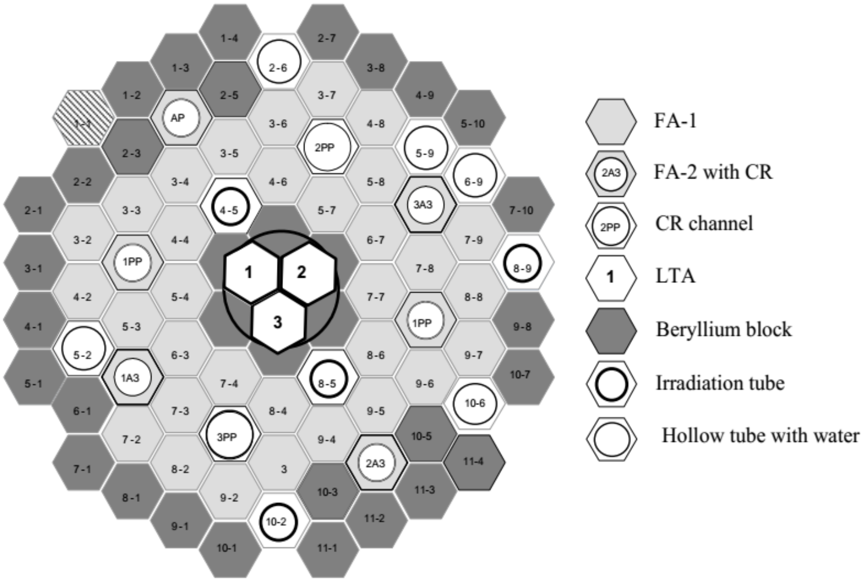

Fig. 2. WWR-K reactor core mapwith HEU VVR-C FAs and three LEU EFAs [4].

The purpose of the tests was to achieve a burn-up of 60% in which defined the life of the reactor FEs. For this purpose, three 8-element EFAs were manufactured with the following ^{235}U mass: 245.3 g in LTA-1, 244.7 g in LTA-2, and 245.0 g in LTA-3.

The parameters which were monitored during the fuel lead test include the coolant temperatures at the core inlet and outlet, the coolant temperature at the outlet of the experimental device with three VVR-KN EFAs inside, the coolant flow rate through the reactor core, the coolant level in the reactor pool, the coolant pressure at the EFAs outlet, the coolant and air activity beneath the reactor head.

The VVR-KN fuel lead test was successfully performed with the total duration of 816 days, including 480 days at the reactor power rate of 6 MWt. The average ^{235}U burn-up of 59.7% (LTA-2) and 60.3% (LTA-3) was achieved.

The safe operating mode for VVR-KN EFAs (no fuel cladding surface boiling) was maintained during the experiment.

Table II presents thermal-hydraulics analysis results from the fuel lead test [4].

Table II. Thermal-hydraulics analysis results for VVR-KN EFAs.

Reactor power, MWt	5.65
Reactor pool water level, m	5.3
Total power of three EFAs, kWt	989
Power of most heat-rated EFA, kWt	333

Core inlet coolant temperature, ^{0}C	32
Maximum fuel cladding temperature, ^{0}C	88
Maximum heat flux, kW/m^2	508
Hot-spot heat flux, kW/m^2	374
Hot-spot saturation temperature, ^{0}C	107.4
Surface boiling onset temperature, ^{0}C	114.7
Surface boiling onset temperature margin	1.43
Flux instability safety margin	2.1

B. Calculation of thermal power distribution

MCNP5 code was used to calculate thermal power of three EFAs placed in the WWR-K RR, as well as their spatial thermal power distribution in the reactor core.

MCNP5 is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/ electron transport, including the capability to calculate eigenvalues for critical systems [2].

Geometrical model of WWR-K RR including three LEU VVR-KN EFAs, all HEU VVR-C FAs, beryllium reflector and core configuration was imported into MCNP5 input file.

After executing the code, results of the EFAs thermal power distribution were gained. Accordingly, thermal power of the three EFAs were of 325 kWt, 322 kWt, and 337 kWt respectively (**Fig. 3**).

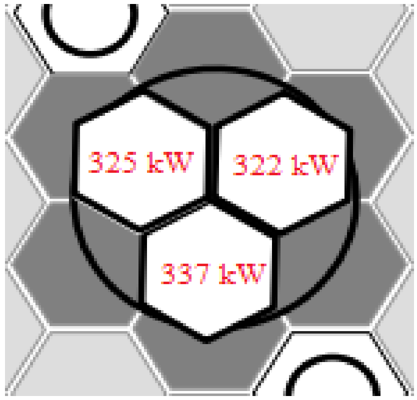

Fig. 3. Thermal power of the three EFAs.

The axial thermal power distribution was obtained as well and shown in **Fig. 4**. As it is evident from the **Fig. 4** that the power peaking factor of EFAs were of 1.287, 1.286, and 1.290 each one.

The calculation of the thermal-hydraulics parameters, as well as safe margins, required attention to thermal power at hottest positions. So that, the hot channel factors of each fuel element of EFAs were calculated by MCNP5 code and presented in **Table III**.

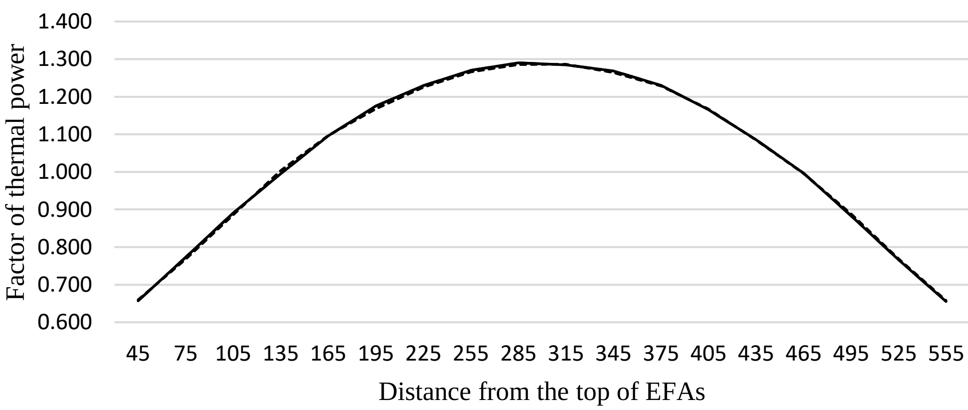

Fig. 4. Axial thermal power distribution of EFAs.

Table III. Factor of thermal power at the hottest position of EFAs.

	1st FE	**2nd FE**	**3rd FE**	**4th FE**	**5th FE**	**6th FE**	**7th FE**	**8th FE**
1st EFA	1.455	1.417	1.404	1.415	1.445	1.478	1.534	1.613
2nd EFA	1.458	1.420	1.408	1.419	1.448	1.482	1.538	1.617
3rd EFA	1.363	1.327	1.316	1.326	1.353	1.385	1.437	1.511

C. Thermal-hydraulics analysis

The calculation of cladding surface temperature, coolant temperature and safety margins of EFAs was performed by using PLTEMP V3.8 code.

PLTEMP code is a thermal-hydraulics code for plate and concentric-tube geometry with capability of calculating natural circulation flow. It was also used to analyze thermal-hydraulics parameters for core conversion of the Dalat RR from HEU to LEU fuel [5].

The initial conditions of the VVR-KN fuel lead test were as follows:

- The coolant temperature at the core inlet is 32^0C;
- The coolant flow rate through an EFA is 13 m^3/h;
- Total thermal power of the EFAs is 983 kWt;

All these conditions together with geometry of EFAs, coolant flow and previous calculated results of the EFAs thermal power distribution were imported into input file of PLTEMP code. Additionally, the system error, such as uncertainty of power and flow measurement, heat transfer coefficient, must be considered to obtain the desired results. The obtained thermal-hydraulics parameters are presented in **Table IV** and **Fig. 5**.

Eventually, the computing results were compared with experimental ones obtained from VVR-KN fuel lead test (**Table V**).

Table IV. Cladding temperature and coolant temperature of the EFAs with system error.

Distance from the top (mm)	**1st EFA**		**2nd EFA**		**3rd EFA**	
	Cladding temp. (^{0}C)	**Coolant temp.** (^{0}C)	**Cladding temp.** (^{0}C)	**Coolant temp.** (^{0}C)	**Cladding temp.** (^{0}C)	**Coolant temp.** (^{0}C)
45	56.072	33.269	56.379	33.293	55.158	33.223
75	60.471	34.208	60.833	34.250	59.403	34.128
105	64.942	35.301	65.361	35.365	63.721	35.181
135	68.992	36.540	69.463	36.628	67.632	36.374
165	73.012	37.914	73.535	38.027	71.515	37.697
195	76.444	39.405	77.013	39.547	74.829	39.134
225	79.210	40.983	79.817	41.154	77.502	40.653
255	81.649	42.624	82.291	42.825	79.858	42.232
285	83.535	44.305	84.206	44.537	81.682	43.850
315	84.721	45.994	85.414	46.257	82.830	45.476
345	85.662	47.668	86.374	47.963	83.741	47.088
375	85.938	49.307	<u>86.662</u>	49.632	84.009	48.666
405	85.496	50.877	86.225	51.232	83.584	50.179

435	84.554	52.353	85.282	52.735	82.675	51.600
465	83.253	53.718	83.975	54.124	81.417	52.914
495	81.232	54.949	81.941	55.379	79.466	54.100
525	78.900	56.030	79.591	56.480	77.213	55.141
555	76.608	56.960	77.282	57.428	74.999	56.037
585	77.175	57.812	77.861	58.296	75.547	56.858

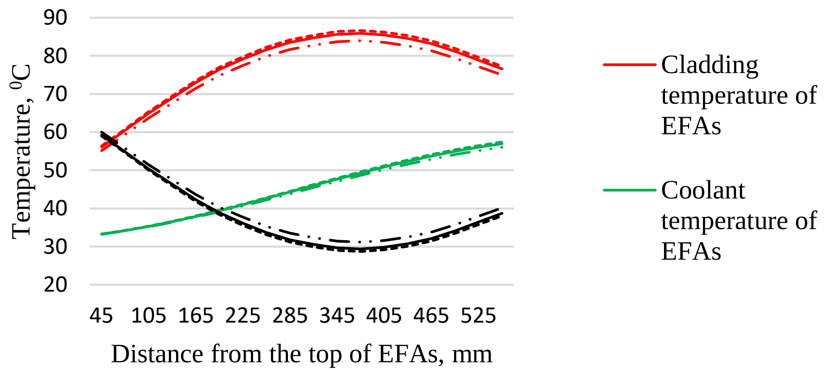

Fig. 5. Cladding and coolant temperature of the EFAs.

Table V. Comparison of the experimental with calculated results.

Parameters	Lead Test	PLTEMP
Reactor power, MWt	5.65	5.65
Power of three EFAs, kWt	989	983
Power of the most heat-rated EFA, kWt	333	337
Core inlet coolant temperature, ^{0}C	32	32
Maximum fuel cladding temperature, ^{0}C	88	86.7
Hot-spot heat flux, kW/m^2	374	387
Hot-spot saturation temperature, ^{0}C	107.4	107.4
Surface boiling onset temperature, ^{0}C	114.7	115.4
Surface boiling onset temperature margin	1.43	1.5
Flux instability safety margin	2.1	2.2

III. CONCLUSIONS

According to the calculated results shown in **Table IV**, the highest cladding temperature of VVR-KN EFAs is 86.662^{0}C, the highest coolant temperature is 58.296^{0}C, and in **Table V**, the minimum flux instability safety margin (FIR) is 2.2 and the minimum surface boiling onset temperature margin (ONBR) is 1.5.

Because the highest temperature at the outlet of the EFAs is far from saturated temperature (107.4^{0}C) so much, belong with obtained FIR and ONBR, it can be concluded that the EFAs operated safely during the test.

As it can be also seen in **Table V**, that the thermal-hydraulics parameters published from the VVR-KN fuel lead test and those computed by PLTEMP code are similar. Therefore, the PLTEMP code can be used for the thermal-hydraulics design of the new nuclear research reactor loaded with LEU VVR-KN FAs of the CNEST project.

REFERENCES

[1] Arne P. Olson, M. Kalimullah, "A users guide to the PLTEMP/ANL V3.8 Code", ANL/RERTR, Argonne National Laboratory, June 2009.

[2] J. F. Briesmeister, Ed., "MCNP - A General Monte Carlo N-Particle Transport Code, Version 5", LA-13709-M, April 2000.

[3] Yu. S. Cherepnin, S. A. Sokolov, S. Yu. Bulkin, V. A. Lukichev, O. V. Kravtsova, A. I. Radaev, "Conversion of the WWR-K research reactor to low-enriched fuel as the basis for the development and introduction of the VVR-KN fuel assemblies in existing and advanced pool-type research reactors", NIKIET, Moscow, Russia, 2014.

[4] F. M. Arinkin, L. V. Chekushina, P. V. Chakrov, Sh. Kh. Gizatulin, S. N. Koltochnik, D, Nakipov, A. A. Shaimerdenov, N. Hanan, P. Garner and J. Roglands-Ribas, "Results of the trial of lead test assemblies in the WWR-K reactor". RRFM 2014, Slovenia, 2014.

[5] "Report on neutronics and steady-state thermal-hydraulics analyses for DNRR full LEU core configurations", Reactor Center, Dalat Nuclear Research Institute, VINATOM, Dalat, 2012.