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PROGRESS IN THE ANALYSES OF IRT, SOFIA LEU CORE: NEUTRONICS AND STEADY-STATE THERMAL-HYDRAULIC ANALYSES

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ABSTRACT

The initial LEU (IRT-4M fuel assemblies, 19.75% ^{235}U) core of the new IRT, Sofia research reactor of the Institute for Nuclear Research and Nuclear Energy (INRNE) of the Bulgarian Academy of Science, Sofia, Bulgaria is jointly analyzed with the RERTR Program at Argonne National Laboratory (ANL) to evaluate its characteristics important for safety analyses. The initial configuration using 16 fuel assemblies (four 8-tube and twelve 6-tube fuel assemblies) is analyzed at a critical core state preferable for BNCT tube operation. Results of detailed neutronics and steady-state thermal-hydraulic calculations for this initial core configuration are presented in this paper; these analyses will be part of the IRT, Sofia Safety Analyses Report.
These results show that for 200 and 500 kW reactor power levels one pump in the primary circuit is sufficient for safe operation but two pumps are needed for safe operation at 1000 kW.

1. Introduction

A joint study concerning IRT, Sofia research reactor (RR) between INRNE and the RERTR Program at ANL was initiated in 2002. The previous steps study [1-4] were mainly focused on neutronics properties significant for reactor application and safety analyses. Presented here are results of further analyses significant for safety assessment of the LEU core for the critical state preferable for Boron Neutron Capture Therapy (BNCT) beam tube operation. The BNCT activity is considered nowadays as one of the most important for future IRT, Sofia application. That is why reactivity coefficients and kinetic parameters calculation previously performed for another critical state of the initial configuration [3, 4] was repeated for the

critical state concerned. Additional calculation of the control rods worth dependence on degree of insertion including safety rods cumulative worth was done. Moreover calculation of the control rods followers worth was also done. These analyses' results together with detailed power distribution data provide input information for thermal-hydraulic steady-state and accident analysis. The results of the steady-state thermal-hydraulic analysis are presented here and of accident analysis – in the supplementary paper [5].
The results of neutronics calculation also presented here include peaking power dependence on unbalanced control rods positioning that has to be accounted for safety evaluation and fuel tube corner roundness (real fuel design) impact to the criticality of the fuel storage facility. The last one is important for substantiation of the Safety Analysis Report [6] conclusion about this facility safely operation. This conclusion is based on the SCALE code system [7] evaluation carried out without accounting for the fuel assembly corner's roundness.

2. Control Rods Worth and Peak Power Density

The IRT, Sofia core configuration with 16 fuel assemblies (FA) is reproduced in Figure 1 [3]. The MCNP code [8] was used for neutronics analysis. The previously developed model of the reactor [2] was modified to account for updated control rods and BNCT beam tube design and extended by additional inner part of the biological shielding including (about 30 cm thickness from its inner wall). The horizontal cross section of calculation MCNP model is presented in Figure 2.

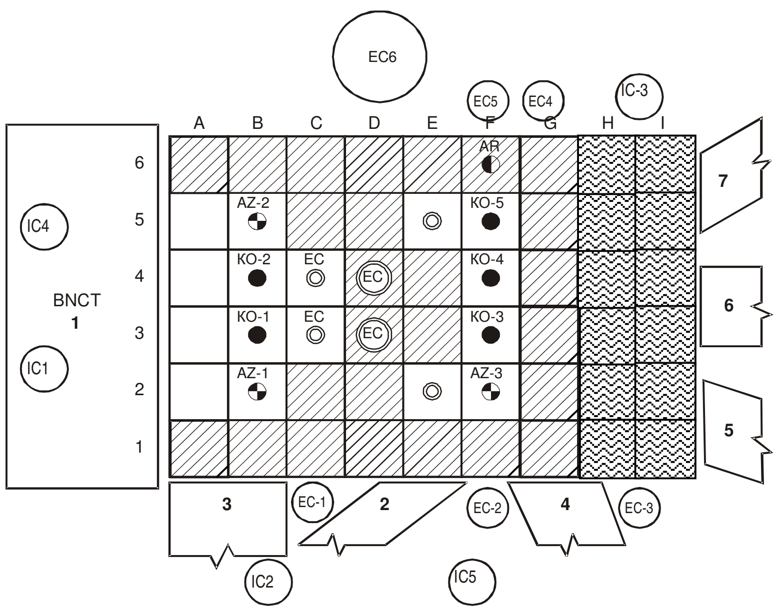

Figure 1. Initial configuration

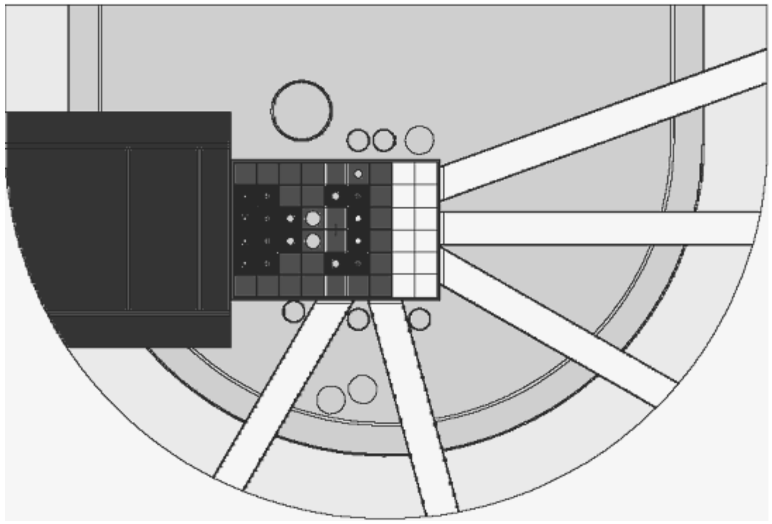

Figure 2. MCNP model for the IRT criticality and BNCT neutron source calculation

The calculated reactivity worth of every shim, safety and auto regulating rods separately (when other rods positions correspond to the concerned critical state) is shown in the Table 1. The model modification did not have significant impact on the control rods critical positions, worth and corresponding shutdown margin. The shutdown margin is equal to 1.34% Δk/k when all safety rods and the shim rod with the highest worth are fully withdrawn and other rods are fully inserted. Consequently the criterion of 1% Δk/k subcriticality is satisfied. The calculated differential cumulative worth of safety rods (AZ-1, AZ-2 and AZ-3), differential worth of automatic rod (AR), and differential worth of the most effective shim rod (KO-1) are presented in Figures 3, 4 and 5 correspondingly. The calculated differential worth of the control rods (AZ-1 and KO-1) followers used for description of accidents with the followers disengagement from the rods being in critical positions is presented in Figures 6.

Table 1. Calculated reactivity worth of the control rods

Rod	Cell	Worth, % Δk/k
AZ-1	B2	2.22
AZ-2	B5	2.25
AZ-3	F2	0.66
AZ-1, AZ-2, AZ-3	–	5.75
KO-1	B3	3.40
KO-2	B4	3.40
KO-3	F3	1.24
KO-4	F4	1.18
KO-5	F5	0.63
AR	F6	0.34
KO-1 – KO-5, AR	-	10.2

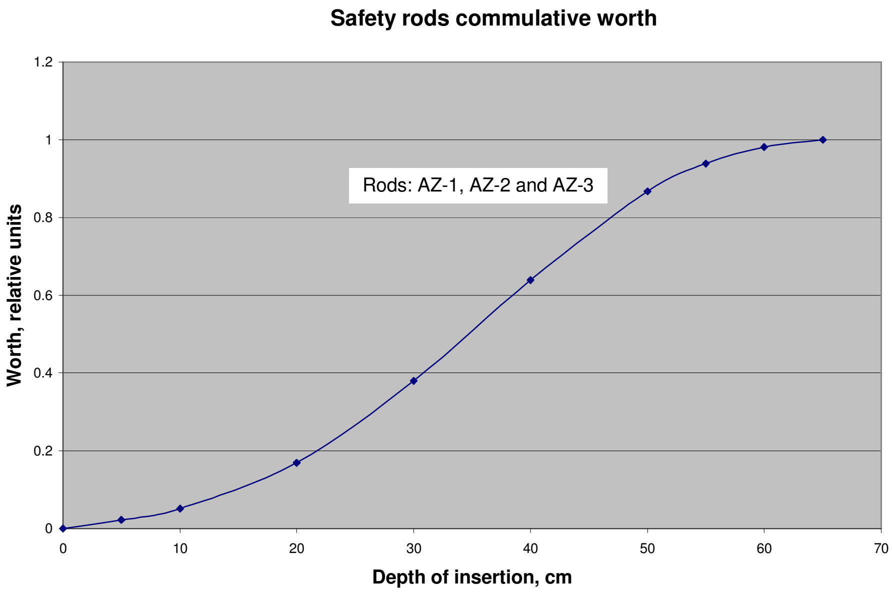

Figure 3. Safety rods (AZ-1, AZ-2 and AZ-3) cumulative worth

Core power distribution and power peaking factors for the considered critical state obtained with the modified model (as expected) are practically the same as those obtained for the previously applied model [4]. The calculated peak power density for the considered critical state is achieved in the outer tube of FA located in the cell C3 and is equal to 86.1 W/cc. The core critical state could be achieved for different sets of positions for the shim rods and in order to evaluate if corresponding unbalanced shim rods positioning could result in a higher peak power density calculations for a set of different critical states were carried out. A set of six critical states including the considered one, previously studied [3] that is preferable for operation with horizontal channels No. 2, 4 -7, and four additional ones were analyzed and the results are presented in the Table 2. The first critical state is the base case discussed before and the second case is that used previously in reference 3. The sixth critical state is used for modeling of the core loading accident when it is allowed that the positioning of the 6 tube FA in the cell C3 and the 8 tube FA in the cell A3 are erroneously exchanged during the core loading.

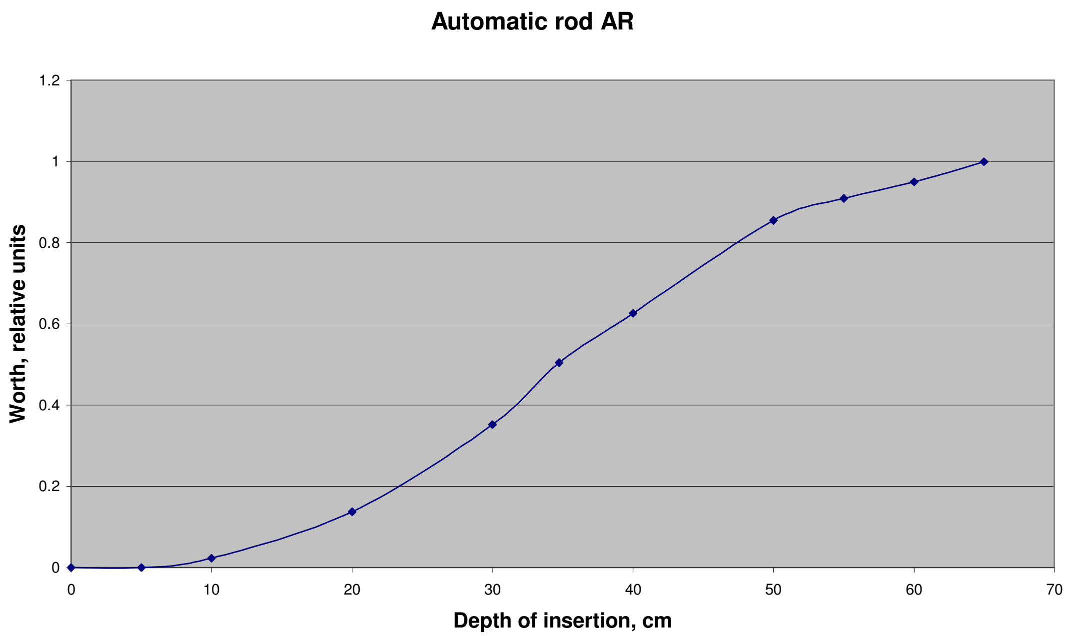

Figure 4. Automatic rod AR worth

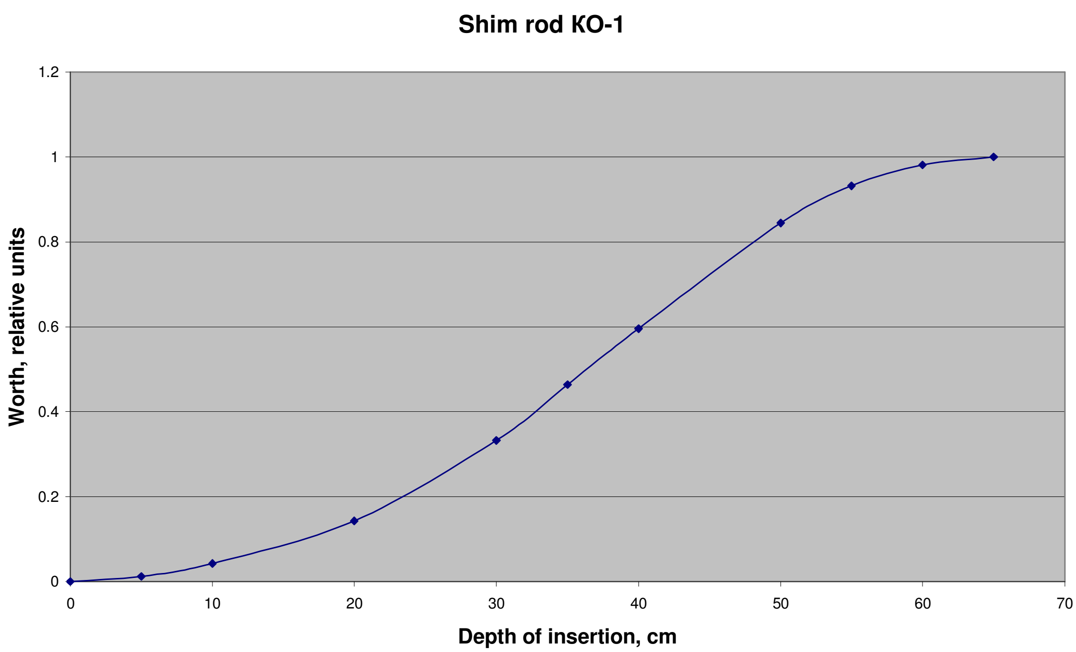

Figure 5. Shim rod KO-1 worth

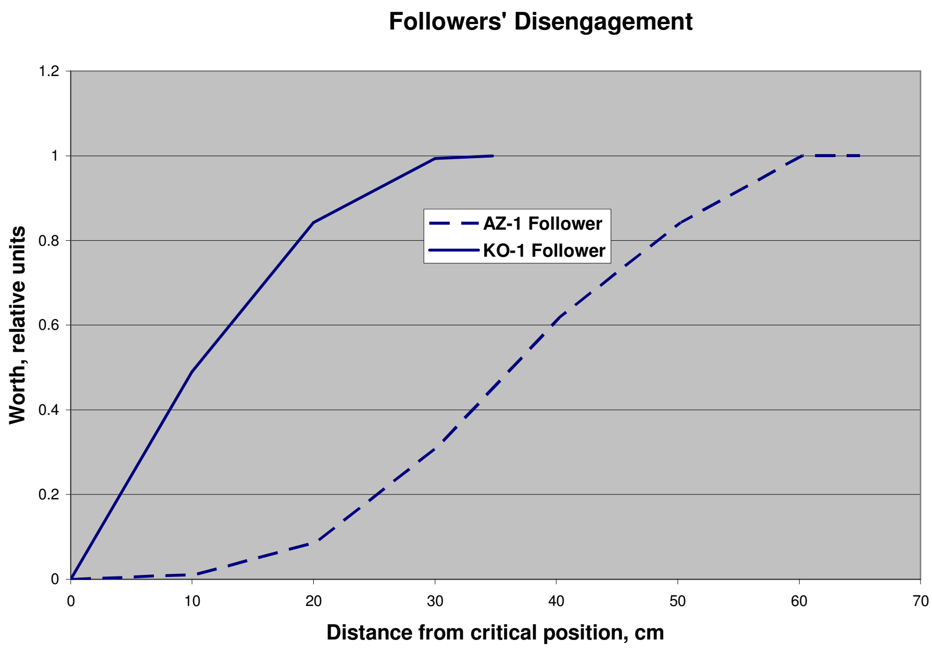

Figure 6. KO-1 and AZ-1 Followers Worth

Table 2. Peak power density (P_{max})for different critical states

No.	Shim rods (KO-1 – KO-5) depth of insertion, cm					P_{max}, W/cc
	1	2	3	4	5	
1	33.0	33.0	65.0	65.0	0	86.1
2	65.0	65.0	13.0	13.0	0	88.0
3	39.2	39.2	39.2	39.2	0	83.9
4	0	49.8	65.0	65.0	0	84.6
5	37.3	37.3	37.3	37.3	37.3	85.3
6	33.4	33.4	65.0	65.0	0	86.0

3. Reactivity Coefficients and Kinetic Parameters

Reactivity coefficients and kinetic parameters were calculated for fresh fuel assemblies and control rods positions (described above) corresponding to the critical state of the initial core that is preferable for operation with BNCT beam tube. The calculations were performed using the MCNP5 code [9].The results of calculations are presented in Table 3.

Table 3. Reactivity coefficients and kinetic parameters

Moderator Temperature Reactivity Coefficient, $\Delta\rho(\%)/^{\circ}C$: 21 ºC to 127 ºC Fuel Temperature (Doppler) Reactivity Coefficient, $\Delta\rho(\%)/^{\circ}C$: 21 ºC to 127 ºC	 – 0.00987 – 0.00203
Moderator Density (Void) reactivity Coefficient, $\Delta\rho(\%)$/(% of Void): 0 to 5%: 5% to 10%: 0 to 10%:	 – 0.279 – 0.298 – 0.288
Prompt Neutron Lifetime, μsec	84.9
Effective Delayed Neutron Fraction (β_{eff})	0.00783

4. Impact of Modeling the FA Corners on Fuel Storage Criticality

In order to evaluate if SCALE calculation for a criticality of the IRT, Sofia fuel storage facility is conservative, MCNP calculations was performed. In the SCALE calculation an approximate model in which the FA is modeled with sharp corners was used [6]. The results of MCNP fuel storage criticality calculations with and without accounting for of FA corner roundness are presented in Table 4. The fuel storage facility (in all results discussed here) was modeled conservatively as an infinite lattice with a pitch equal to 14 cm (according to the real facility geometry) filled with 8-tube FAs. The MCNP results (with and without the FA round corners) show that use of the IRT-4M FA model without round corner (as in the SCALE model) yields conservative criticality evaluation of the IRT, Sofia fuel storage facility.

Table 4. Criticality of the fuel storage facility – MCNP Results

Fuel tube corner geometry	k_{eff}
Sharp (without round corners)- approximation	0.81191 ± 0.00046
With round corners	0.80128 ± 0.00051

4. Steady State Thermal Hydraulic Analysis

Steady-state thermal hydraulic analyses were performed using the code PLTEMP/ANL V3.4 (ANL) [10]. The fuel assembly with the peak power density (FA-C3, a 6-tube FA) was used in these calculations. All the 6 tubes and 7-coolant channels are modelled and 15 axial segments are used. In PLTEMP, the fuel tubes were modelled as parallel plates. The calculations were performed for power levels of 200, 500, and 1000 kW.

The PLTEMP analyses presented below were performed for two cases: a) with the “hot channel factors” (HCF) based on the parameters (uncertainties) presented in [11]; and b) without hot channel factors (using all nominal parameters). For the case with “hot channel factors”, the maximum possible value of the coolant inlet temperature, 47.2°C was used, instead of nominal – 45°C. For power levels of 200 and 500 kW calculations were performed with one primary circuit pump and with both one and two pumps – for 1000 kW. Detailed power distribution obtained using MCNP detailed models discussed in section 2 were used in the PLTEMP calculations. The hottest fuel assembly was modeled in PLTEMP using the power generated in the hottest segment of the FA. It was assumed that 94% of the power is generated in the fuel meat and 6% is directly deposited in the coolant.

Table 5 – Hottest channel PLTEMP results without accounting of HCF (one pump)

Parameter	Reactor Power, kW		
	200	500	1000
Maximum power density in the fuel tube, MW/m^3	86.1	215.	430.
Maximum temperature of fuel meat, ^0C	52.1	62.1	77.7
Maximum thermal flux (outer/inner), kW/m^2	31.5/28.7	78.9/71.8	158./143.
Maximum clad surface temperature (outer/inner), ^0C	51.9/52.0	61.8/61.8	77.1/77.2
Onset of nucleate boiling temperature at minimum ONBR (Bergles-Rosenhaw), ^0C	115.1	116.	117.
Minimum onset of nucleate boiling ratio (Bergles-Rosenhaw)	10.6	4.41	2.31
Safety margin to flow instability	15.1	6.06	3.04
Critical heat flux (Mirshak), kW/m^2	3440	3330	3150
Maximum coolant temperature, ^0C	48.7	54.2	63.4

The results of the thermal-hydraulics calculations are presented in Tables 5, 6 and 7, and Figure 7 for both cases (without hot channel factors and with hot channel factors) for the three power levels considered (200, 500, and 1000 kW). The dashed lines (Figure 7) describe the limiting power and flow rate values that include the scram level and power/flow rate uncertainty [11]. The vertical dashed lines are shifted by 11% to the left against the nominal flow rates corresponding to one or two pumps operation. This shift includes 8% coming from scram activation and 3% from flow rate uncertainties. The horizontal dashed lines are shifted upward from the nominal power levels of 500 kW and 1000kW by 30%; i.e. 20% from scram activation and 10% from power uncertainty [11]. The limiting solid lines in the Figure 6 corresponds to: 1) the maximum clad surface temperature for the FA operation equal to 98°C, using HCF [12], and 2) the minimum Onset of Nuclear Boiling Ratio (ONBR) equal to 1.4 [6] using the Bergles-Rosenhaw correlation.

Table 6 – Hottest channel PLTEMP results with accounting of HCF (one pump)

Parameter	Reactor Power, kW		
	200	500	1000
Maximum power density in the fuel tube, MW/m^3	94.7	237.	473.
Maximum temperature of fuel meat, ^0C	58.2	73.7	97.7
Maximum thermal flux (outer/inner), kW/m^2	34.5/ 31.7	86.5/79.2	173./158.
Maximum clad surface temperature (outer/inner), ^0C	58.0/ 58.1	73.2/73.3	96.7/97.0
Onset of nucleate boiling temperature at minimum ONBR (Bergles-Rosenhaw), ^0C	115.	116.	117.
Minimum onset of nucleate boiling ratio (Bergles-Rosenhaw)	6.57	2.74	1.85
Safety margin to flow instability	12.9	5.16	2.59
Critical heat flux (Mirshak), kW/m^2	3380	3250	3048
Maximum coolant temperature, ^0C	52.7	61.0	74.8

Table 7 – Hottest channel PLTEMP results (two pumps, 1000 kW)

Parameter	Hot Channel Factor	
	not included	included
Maximum power density in the fuel tube, MW/m^3	430.	473.
Maximum temperature of fuel meat, ^0C	66.6	80.6
Maximum thermal flux (outer/inner), kW/m^2	157./144.	172./159.
Maximum clad surface temperature (outer/inner), ^0C	66.0/66.1	79.7/79.9
Onset of nucleate boiling temperature at minimum ONBR (Bergles-Rosenhaw), ^0C	116.	117.
Minimum onset of nucleate boiling ratio (Bergles-Rosenhaw)	3.56	1.85
Safety margin to flow instability	5.01	4.26
Critical heat flux (Mirshak), kW/m^2	3456	3365
Maximum coolant temperature, ^0C	55.9	63.6

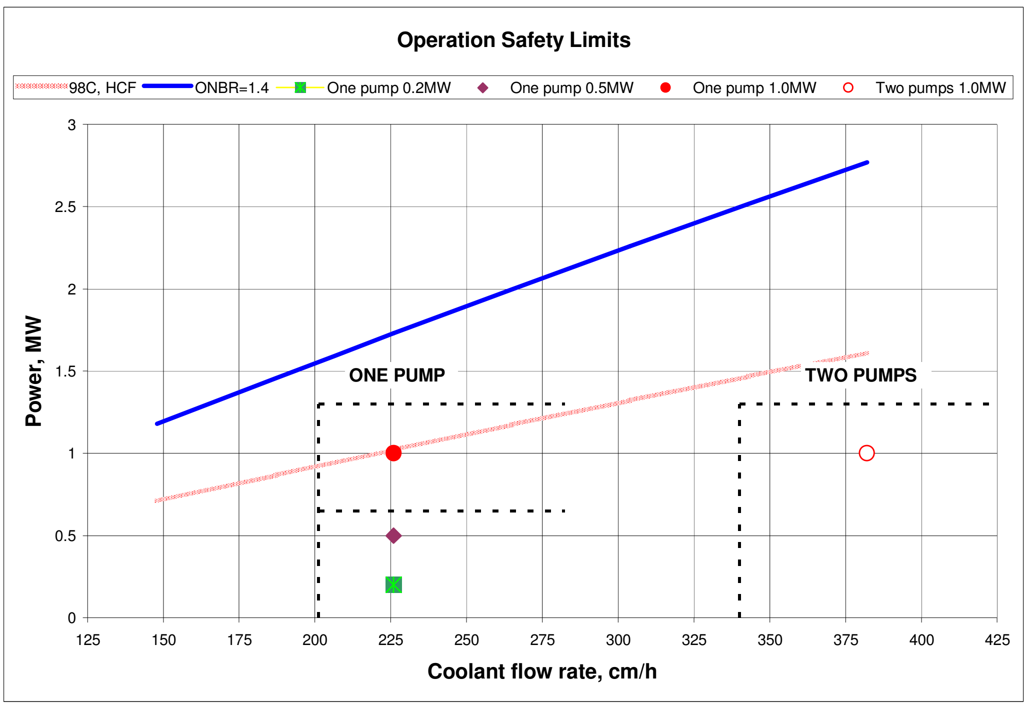

Figure 7. Reactor Power Limits for achievement of ONBR=1.4 (without accounting for HCF) and of 98°C (accounting for HCF).

The results show the following:

A) One pump operation: 1) all safety margins are met for the 200 kW and 500 kW even with the hot channel factors; 2) for 1000 kW and with the hot channel factors the maximum clad surface temperature is exactly the same as the maximum allowed in FA passport [12]. If the criterion for the maximum fuel temperature includes the HCF (according to FA passport [12]), the reactor cannot operate at 1000 kW with one pump.

B) Two pumps operation: All safety margins are met for the 1000 kW even with the hot channel factors.

5. Conclusions

The results of this study provide the following important information required for safety analyses report for the IRT, Sofia:
a) Differential worth for the control rods and for the control rod's follower ;
b) Determination of the peak power density dependence on the control rods positioning in the base case critical state, in different cofigurations with unbalanced rod positioning, and in the case of erroneous loading of a FA which could occur during core loading;
c)The use of an approximate model for the FA (sharp corners) with the SCALE code was shown to provided conservative results for the criticality of the fuel storage;
d)The steady-state thermal-hydraulic calculations demonstrated that for 200 and 500 KW reactor power levels one pump in the primary coolant circuit is sufficient for the safely operation but for operation at 1000 kW two pumps should be used.

We intend to continue the fruitful joint study between INRNE and the RERTR Program at ANL for successful completion of the safety analysis for the selected LEU core and Safety Analyses Report preparation for IRT, Sofia in accordance with licensing requirements.

6. Acknowledgement

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7. References

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