PROGRESS IN CONVERSION FROM HEU TO LEU FUEL AT IRT, SOFIA

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ABSTRACT

The new 200 kW IRT-Sofia research reactor of the Institute for Nuclear Research and Nuclear Energy (INRNE) of the Bulgarian Academy of Science, Sofia, Bulgaria is jointly studied with the RERTR Program at Argonne National Laboratory (ANL) to realize its conversion from the use of fuel containing highly enriched uranium (HEU, 36% ^{235}U) to use of fuel containing low enriched uranium (LEU, 19.75% ^{235}U). After the applicability of IRT-4M LEU fuel for conversion was approved, further activities related to modification of the Safety Analyses Report were initiated. An initial core configuration was selected and its neutronic calculation was performed. The initial LEU core configuration uses 16 IRT-4M fuel assemblies (four 8-tubes and twelve 6-tubes). Results of detailed calculations of this initial core configuration and the neutronics properties that are important for preparation of the Safety Analyses Report are presented. These results provide information for thermal-hydraulic and accident analyses that are needed to demonstrate that the safety margin requirements are satisfied for the selected configuration.

1. Introduction

A joint study between INRNE and the RERTR Program at ANL was initiated in 2002 to study the possibility of using LEU fuel instead of HEU fuel in the planned new 200 kW IRT-Sofia research reactor [1]. The joint work of the study was started from common calculational models based on the best available data for reactor components and materials, and was followed with comparison of the fuel lifetime and flux performance for sequences of HEU (IRT-2M, 36%) and LEU (IRT-4M, 19.7%) cores providing appropriate flux values and acceptable burn-up levels [2]. It was concluded that the LEU core performance (both in terms of fluxes for the experiments and in fuel consumption) is similar to the HEU reference core. Thus, the IRT-4M LEU fuel assemblies were accepted as suitable for conversion of 200 kW IRT, Sofia research reactor. Presented here are the selection of the initial configuration for the LEU core loading and results of the calculations that are needed for modification of the Safety Analyses Report in connection with the fuel conversion. Selection of the initial core configuration was carried out in close collaboration with scientists from RRC "Kurchatov Institute", Russian Federation.

2. Selection of the initial core configuration

According to the preliminary analysis performed [2] to obtain an acceptable level of fuel burn-up in the IRT-Sofia core, a sequence of core loading configurations needs to be used during the first cycles. No fuel assemblies are discharged from the core until the number of FA in the core becomes 22, when the LEU FA with maximum burn-up reaches an acceptable level for discharge (about 46%). The final core configuration could be achieved as a result of different sets of core loading sequences that satisfy the safety margins requirements for every loading configuration. Selection of the initial core configuration and the final loading sequence of the core are studied here.

The IRT-Sofia research reactor is planned to be equipped with a set of beam tubes (channels) that can be utilized for different purposes. In this connection still at the stage of preliminary analysis for HEU fuel, it was decided to use an initial configuration that consists of two separated parts (configuration 1, Figure 1) to provide a capability for flexible reactor application. To provide information needed for the initial core selection, a set of configurations was analyzed (Figure 1). The MCNP code [3] and previously developed models of the reactor [2] were used for this analysis. First of all, it should be noted that the HEU configuration is not applicable for LEU fuel because the core configuration that corresponds to direct substitution of IRT-2M FA by IRT-4M FA provides initial excess reactivity (0.94% Δk/k) that is not sufficient for reactor operation. That is why nine additional configurations (Figure 1) with different number of FA, beryllium reflector blocks and their arrangement were studied. The results obtained for these configurations for initial excess reactivity and relative values of channels performance are presented in Table 1. Based on these results, configuration number 5 was selected as the initial core configuration. The use of this configuration was highly recommended by specialists from RRC "Kurchatov Institute" taking into account its advantage in channels (from No. 5 to No. 7) performance and lower initial excess reactivity (that is preferable for safety) in comparison with configuration number 4.

3. Initial core configuration

The selected initial core configuration is presented in Figure 2. The positions of the experimental channels surrounding the reactor core are shown in Figure 2 as well. This core includes 16 IRT-4M fuel assemblies: twelve 6-tube FA, and four 8-tube FA. There are totally 54 cells (7.15x7.15cm) in the frames of the reactor vessel. Figure 2shows positions of the experimental channels (horizontal and vertical), channels for ionizing chambers, and the shim (KO), safety (AZ) and auto regulating (AR) rods. The calculated value of cold clean excess reactivity of the core is equal to 5.33% Δk/k.

The reactivity worth of each shim, safety and auto regulating rods calculated separately is shown in the Table 2. The corresponding shutdown margin is calculated for a cold core condition where it is assumed that all safety rods and the auto regulating rod are fully-withdrawn, and all shim rods are fully-inserted. The shutdown margin in those conditions is equal to 3.65% Δk/k. This value meets the shutdown margin criterion that the subcriticality must be at least 1% Δk/k under these conditions. Moreover, the criterion of 1% Δk/k subcriticality is satisfied even if the shim rod with the highest worth is also fully withdrawn. In that case the calculated subcriticality is equal to 1.13% Δk/k. It must be noted that the sum of the rod worths in Table 2 gives just 0.79%

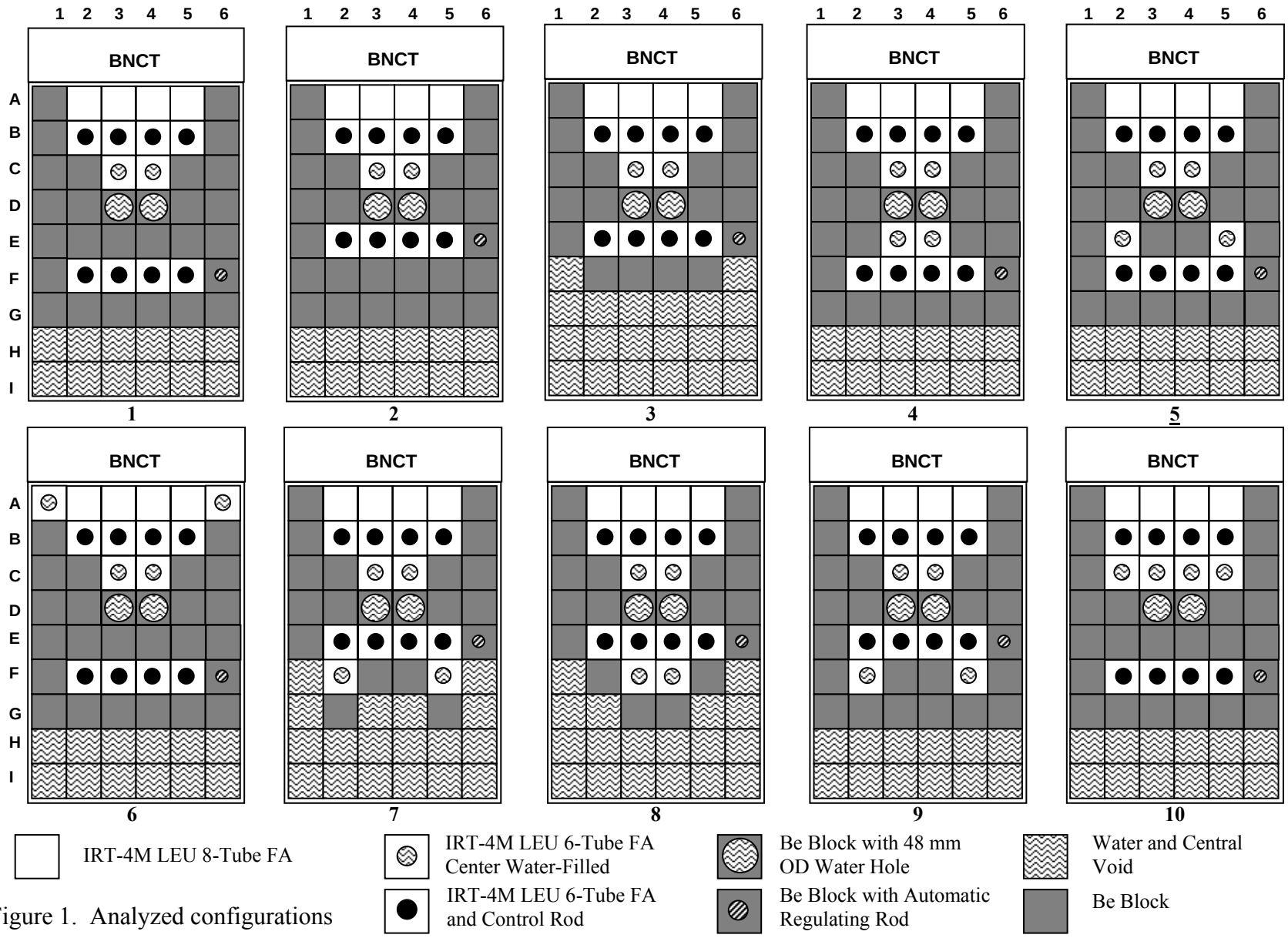

Figure 1. Analyzed configurations

Table 1. IRT, Sofia Performance Comparison: LEU/HEU Ratio of k_{eff} x Flux
Configuration No.1 for HEU**, ρ=3.56%,** for LEU **ρ=0.94%**
Configurations 1-3: 14 FA; Configurations 4-10: 16 FA

Channel **No.**	Energy range	Config. 2 **ρ=5.72%**	Config. 3 **ρ=4.29%**	Config. 4 **ρ=6.55%**	Config. 5 **ρ=5.33%**	Config. 6 **ρ=1.90%**	Config. 7 **ρ=5.14%**	Config. 8 **ρ=5.92%**	Config. 9 **ρ=7.29%**	Config. 10 **ρ=5.13%**
1*	E<6.25e-7 MeV	**0.77**	**0.81**	**0.67**	**0.68**	**0.79**	**0.73**	**0.70**	**0.67**	**0.76**
	6.25e-7<E<8.21e-1 MeV	**0.91**	**0.95**	**0.79**	**0.80**	**1.00**	**0.87**	**0.83**	**0.78**	**0.90**
	E>8.21e-1 MeV	**0.86**	**0.90**	**0.74**	**0.75**	**0.95**	**0.83**	**0.77**	**0.74**	**0.85**
2	E<6.25e-7 MeV	**1.15**	**0.92**	**1.17**	**1.15**	**0.99**	**0.94**	**0.95**	**1.18**	**1.09**
	6.25e-7<E<8.21e-1 MeV	**1.74**	**1.37**	**1.47**	**1.94**	**1.10**	**1.57**	**1.47**	**2.02**	**1.42**
	E>8.21e-1 MeV	**1.85**	**1.60**	**1.45**	**2.17**	**1.07**	**1.99**	**1.74**	**2.21**	**1.41**
3	E<6.25e-7 MeV	**0.94**	**0.95**	**0.82**	**0.84**	**0.82**	**0.89**	**0.86**	**0.85**	**0.91**
	6.25e-7<E<8.21e-1 MeV	**0.99**	**1.03**	**0.86**	**0.88**	**1.32**	**0.94**	**0.91**	**0.88**	**1.08**
	E>8.21e-1 MeV	**0.96**	**0.98**	**0.82**	**0.84**	**1.86**	**0.90**	**0.89**	**0.84**	**1.08**
4	E<6.25e-7 MeV	**1.19**	**1.14**	**1.41**	**1.45**	**1.05**	**1.24**	**1.39**	**1.39**	**1.13**
	6.25e-7<E<8.21e-1 MeV	**0.85**	**1.38**	**1.58**	**1.74**	**1.16**	**2.07**	**1.99**	**1.63**	**1.22**
	E>8.21e-1 MeV	**0.71**	**1.74**	**1.52**	**1.69**	**1.09**	**3.05**	**2.55**	**1.56**	**1.18**
5	E<6.25e-7 MeV	**1.10**	**1.11**	**1.47**	**1.44**	**1.06**	**1.19**	**1.28**	**1.37**	**1.11**
	6.25e-7<E<8.21e-1 MeV	**0.61**	**1.09**	**1.66**	**1.56**	**1.13**	**1.31**	**1.38**	**1.20**	**1.17**
	E>8.21e-1 MeV	**0.51**	**0.97**	**1.58**	**1.54**	**1.12**	**1.24**	**1.33**	**1.11**	**1.14**
6	E<6.25e-7 MeV	**1.09**	**1.06**	**1.46**	**1.44**	**1.04**	**1.16**	**1.19**	**1.34**	**1.10**
	6.25e-7<E<8.21e-1 MeV	**0.60**	**0.98**	**1.62**	**1.55**	**1.14**	**1.26**	**1.09**	**1.14**	**1.15**
	E>8.21e-1 MeV	**0.49**	**0.92**	**1.65**	**1.61**	**1.11**	**1.35**	**1.13**	**1.06**	**1.16**
7	E<6.25e-7 MeV	**1.10**	**1.15**	**1.47**	**1.47**	**1.06**	**1.23**	**1.36**	**1.37**	**1.12**
	6.25e-7<E<8.21e-1 MeV	**0.60**	**1.16**	**1.57**	**1.51**	**1.13**	**1.35**	**1.49**	**1.22**	**1.15**
	E>8.21e-1 MeV	**0.52**	**1.10**	**1.65**	**1.62**	**1.14**	**1.39**	**1.56**	**1.18**	**1.18**

*in front of the filters of the BNCT channel

Δk/k subcriticality. That demonstrates conservatism of an additive cumulative rod worth assessment.

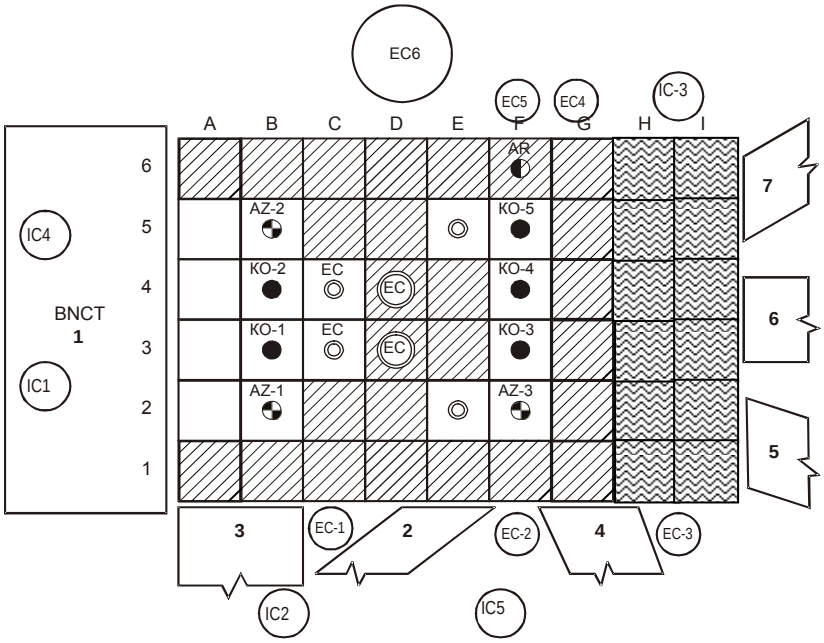

Figure 2. Initial configuration

Table 2. Calculated reactivity worth of the control rods

Rod	Cell	Worth, % Δk/k
AZ-1 AZ-2 AZ-3	B2 B5 F2	1.66 1,66 0.97
AZ-1, AZ-2, AZ-3	–	4.29
KO-1 KO-2 KO-3 KO-4 KO-5	B3 B4 F3 F4 F5	2.47 2.47 1.34 1.34 0.97
AR	F6	0.39
KO-1 – KO-5, AR	-	8.98

A critical state of the selected configuration that is preferable for operation with horizontal channels No.5 -7 is realized according to calculations (excess reactivity equal to 0.07% Δk/k) when the shim rods are inserted accordingly: KO-1=KO-2=65.0cm; KO-3=KO-4=18.8cm; AR=41.5cm, and all other rods are fully-withdrawn.

4. Burn-up calculation

Burn-up calculations were run using the REBUS code [4] for the selected initial core configuration to estimate the duration of the first cycle (without charging of additional fuel

assemblies). The reactivity profile at 200 kW power level is shown in Figure 3. The initial excess reactivity of the core obtained using diffusion theory calculation with the REBUS code is in good agreement (it is 0.5% Δk/k higher) with the value obtained using the MCNP code, taking into account approximations applied in geometry, cross sections [5] and energy dependence description in the diffusion calculation.

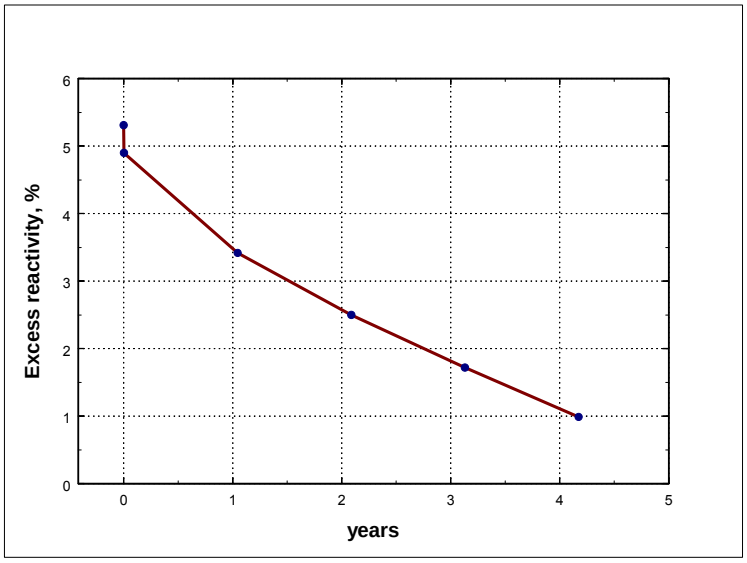

Figure 3. Burn-up reactivity profile

Reactivity losses due to equilibrium Xe and Sm are included in the curve. Based on this calculation, the lifetime of this core will be about four years. This lifetime evaluation corresponds to continuous operation. The core will operate for a much longer time if a realistic duty factor would be taken into account. The maximum planed duty factor value is about 0.3 that corresponds to operation during twelve hours every day, five days a week and forty one weeks per year.

5. Power distribution and power peaking factors

Power distribution and nuclear power peaking factors were calculated using the diffusion theory model and the REBUS code as it had been done previously for the HEU core. The control rods were not represented in the calculational model. The results are shown in Figure 4.

The power peaking factors presented in Figure 4 are the point-wise values computed at the calculation mesh intervals. Additionally the relative power distribution over twelve equal axial intervals for every FA was calculated, as well as tube-by-tube distribution for every fuel assembly. These data are required for thermal-hydraulic calculations and could be used for comparison of thermal-hydraulic conditions for LEU and HEU fuel. However we intend to obtain more detailed and reliable data about relative power distribution by using MCNP calculation for a model with shim rods position providing critical state of the core.

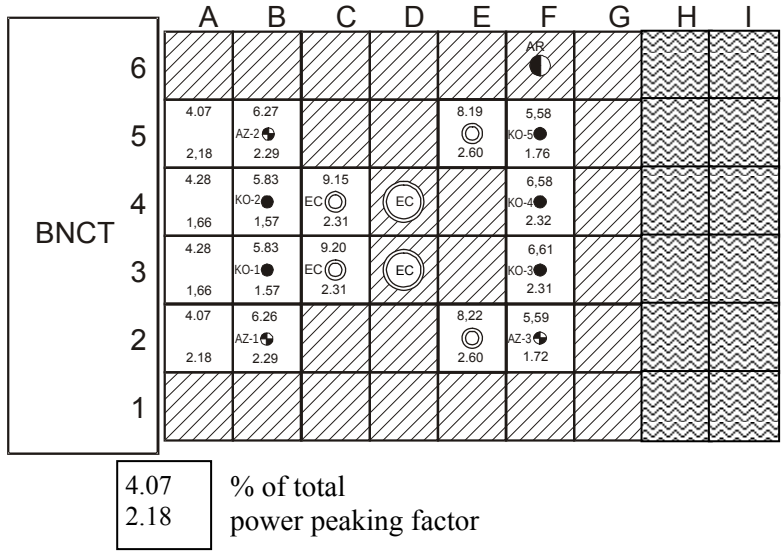

Figure 4. Power distribution and power peaking factors.

6. 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 horizontal channels No.5 -7. The calculations were performed using the MCNP5 code [6].The results of calculations are presented in Table 3.

Table 3. Reactivity coefficients and kinetic parameters

Moderator Temperature Reactivity Coefficient, $\Delta\rho(\%)/^{o}C$: 21 ºC to 127 ºC	– 0.0104
Fuel Temperature (Doppler) Reactivity Coefficient, $\Delta\rho(\%)/^{o}C$: 21 ºC to 127 ºC	– 0.00206
Moderator Density (Void) reactivity Coefficient, $\Delta\rho(\%)/(\%$ of Void$)$: 0 to 5%: 5% to 10%: 0 to 10%:	 – 0.235 – 0.253 – 0.244
Prompt Neutron Lifetime, µsec	94.2
Effective Delayed Neutron Fraction (β_{eff})	0.00783

7. Conclusion

The results of this study demonstrate significant progress in the joint study between INRNE and the RERTR Program at ANL for conversion from HEU to LEU fuel at IRT, Sofia.
Presented here new results were used for selection of the initial configuration for the LEU core loading. Selection of the initial core configuration was carried out in close collaboration with scientists from RRC "Kurchatov Institute", Russian Federation. The initial configuration for the LEU core was selected on the base of calculation results comparison for neutron flux value at the experimental positions and initial core excess reactivity value important for safety ensuring.
Calculation results for the control rods worth demonstrates that shutdown margins safety requirements are satisfied for the selected initial core configuration.
The results of burn-up calculation indicated that the selected core would last about four years for continuous operation. That means that the core would operate about three times longer time in conditions corresponding to a realistic value (about 0.3) of a duty factor.
The results obtained for relative power distribution provide useful information for comparison of thermal-hydraulic conditions between HEU and LEU cores.
We intend to continue the fruitful joint study between INRNE and the RERTR Program at ANL for conversion from HEU to LEU fuel at IRT, Sofia especially for completion of the safety analysis for the selected LEU core and Safety Analyses Report preparation in accordance with licensing requirements.

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8. References

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