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Full Core Conversion and Operational Experience with LEU Fuel of the DALAT Nuclear Research Reactor

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ABSTRACT

After successful in full core conversion from HEU to LEU fuel at the end of 2011, safe operation and utilization of the Dalat Nuclear Research Reactor (DNRR) was emphasized and improved. During carrying out commissioning program for reactor start up, reactor characteristics parameters were measured and compared with design calculated data. Good agreement between experimental data and calculated data was archived and the working core has met all safety and exploiting requirements. Fuel and in-core management of LEU core were implemented by using MCNP-REBUS linkage computers code system with visual interface to make friendly using. The neutron trap was modified to serve for producing I-131 by increasing 6 containers. Other irradiation channels inside the reactor core and horizontal beam tubes are being effectively used for neutron activation analysis, fundamental research, nuclear data measurement, neutron radiography and nuclear structure study.

1. Introduction

Physics and energy start-up of the Dalat Nuclear Research Reactor (DNRR) for full core conversion to low enriched uranium (LEU) fuel were performed from November 24th, 2011 until January 13th, 2012. The program provides specific instructions for manipulating fuel assemblies (FAs) loading in the reactor core and denotes about procedures for carrying out measurements and experiments during physics and energy start-up stages to guarantee that loaded LEU FAs in the reactor core are in accordance with calculated loading diagram and implementation necessary measurements to ensure for safety operation of DNRR. Starting loading LEU fuel to the reactor core until finishing 72 hours testing operation without loading at nominal power was carried out from November, 24th, 2011 to December, 13rd, 2011.

Many computer code systems have been investigated and applied for safety analysis as well as operation management and utilization of the DNRR. Especially, in full core conversion project, some selected computer codes have been served for design calculation and safety analysis. Obtained results from full core conversion project show that calculation tools were fully met requirements about in core and fuel management and also in researching, operating management and utilization of the DNRR. Having advantages and the possibility of personal computers nowadays, REBUS-MCNP has been chosen as main calculation tool for in core and fuel management of the DNRR using LEU fuel. With complex geometry as the DNRR together with the presence of beryllium in the core, REBUS-MCNP system can fully satisfy the computational requirements. Time consuming in calculation was overcome by using PC cluster on MPI environment.

Design calculations and experiments to improve the efficiency for irradiation of radioisotope production in the neutron trap of LEU working core have been performed. By changing the structure of basket in the neutron trap, 6 containers can be loaded in and the TeO_2 target mass is increased about 1.5 times compared with 4 containers as before. The experimental results show that total activity of the isotope I-131 in the targets increases about 23% after irradiating 130 hours continuously.

2. Reactor start-up using LEU fuel

2.1. Physics start-up

Physics startup of reactor is the first phase of carrying out experiments to confirm the accuracy of design calculated results, important physical parameters of the reactor core to meet safety requirements. Physics startup includes fuel loading gradually until to approach criticality, loading for working core and implementing experiments to measure parameters of the core at low power such as control rods worth, shutdown margin, temperature effect,

The loading of LEU FAs to the reactor core was started on November, 24th, 2011 following a predetermined order in which each step loaded one or group LEU FAs to the reactor core. After each step, the ratio of $\frac{N_0}{N_i}$ (N_0 is initial number of neutron count rate, N_i is that to be obtained after step *ith*) was evaluated to estimate critical mass. At 15h35 on November, 30th, 2011 the reactor reached critical status with core configuration including 72 LEU FAs and neutron trap in center (see **Fig. 1** and **2**).

Established critical core configuration with 72 LEU FAs having neutron trap is in good agreement with design calculated results. With 72 LEU FAs, by changing position of some fuel assemblies, all new criticality conditions were achieved with lesser inserting position of regulating rod. It is concluded that the above critical configuration (**Fig. 1**) is the minimum one among established configurations. The critical mass of Uranium is 15964.12 g in which Uranium-235 is 3156.04 g.

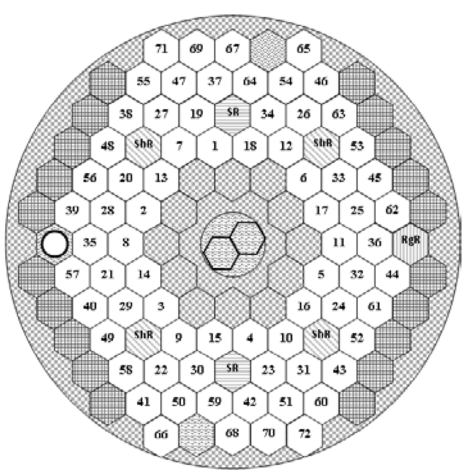

Fig. 1. Critical core configuration and order of loaded fuel assemblies

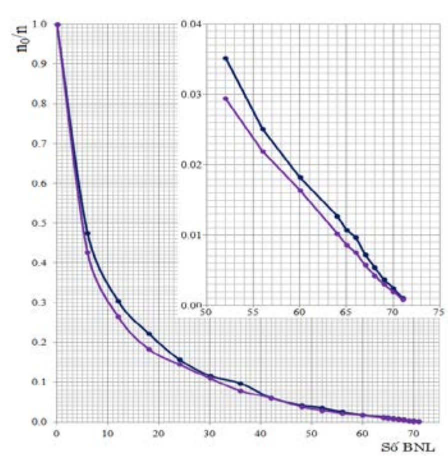

Fig. 2. N_0/N_i ratio versus number of FAs loading to the core

After completion of fuel loading to approach criticality, fuel loading for working core was carried out. During fuel loading for working core, effective worth of loaded fuel assemblies and shutdown margin were preliminarily evaluated to ensure shutdown margin limit not be violated. **Fig. 3** shows the current working core of DNRR, including 92 LEU FAs (80 fresh LEU FAs and 12 partial burnt LEU FAs, the burn up about 1.5 to 3.5 %) and neutron trap at the center. Total mass of U-235 that was loaded to the reactor core is about 4246.26 g. Shutdown margin (or sub-criticality when 2 safety rods are fully withdrawn) is 2.5 $ (about 2% $\Delta k/k$), smaller than calculated value (3.65 $) but still completely satisfy the requirement >1% for the DNRR. Excess reactivity of the core configuration is about 9.5 $, higher than calculated value (8.29 $), ensuring operation time of the reactor more than 10 years with recent exploiting condition. So, it can be said that the current working core meets not only safety requirements but reactor utilization also (ensure about shutdown margin and sufficient excess reactivity for reactor operation and utilization).

Determination of control rod worth

The calibration of control rods of DNRR were implemented two time during fuel loading for working core in configuration with 82 fresh LEU FAs and 92 LEU FAs. Control rod worths and integral characteristics in core configuration with 92 LEU FAs are presented in **Table 1**. Measured results were smaller than design calculated results about 12% in average. Calculated values of control rod worths are higher than experimental data because of evaluated Beryllium poisoning effect higher than practice.

Table 1. Effective worth of regulating rod, 4 shim rods and 2 safety rods in core configuration with 92 LEU FAs.

Control Rod	**Effective reactivity ($)**	
	Measured value	**Calculated value**
Regulating rod	0.495	0.531
Shim rod 1	2.966	3.178
Shim rod 2	3.219	3.263
Shim rod 3	2.817	2.985
Shim rod 4	2.531	2.709
Safety rod 1	2.487	2.604
Safety rod 2	2.195	2.219

Thermal neutron flux distribution measurement in the reactor core

Measurement of thermal neutron flux distribution following axial and radial in the reactor core was carried out by Lu metal foils neutron activation. A number of positions in the reactor core were chosen to measure thermal neutron flux distribution including neutron trap, irradiation channels 1-4 and 13-2, and 10 FAs at the cells: 1-1, 2-2, 2-3, 2-7, 3-3, 3-4, 4-5, 6-4, 12-2 and 12-7. From the measured results, it can be seen that the maximum peaking factor of 1.49 is achieved at outer corner of hexagonal tube of the fuel assembly in cell 6-4. Neutron distribution of working core has large deviation from North (thermal column) to South (thermalizing column). Neutron flux in southern region of the core (cell 12-1 and 12-7) is about 28 % smaller than those in Northern region (cell 2-1 and 2-7). The asymmetry of the reactor core has reason from the not identical reflector that was noted from the former HEU fuel core.

Determination of effective worth of FAs, beryllium rods and void effect

The measurements of effective worth of FAs, beryllium rods and void effect (by inserting an empty aluminum tube with diameter of 30 mm) were also performed. These are important parameters related to safety of the reactor. Positions for measurement of effective reactivity of FAs, Be rods and void effect were chosen to examine the distribution, symmetry of the core and the interference effects at some special positions. Effective reactivity of FAs, beryllium rods and void effect were determined by comparing position change of control rods before and after withdrawing FA or beryllium rod or before and after inserting watertight aluminum tube. Reactivity worth values were obtained using integral characteristics curves of control rods. **Figs 4÷6** show the measured results of effective worth of 14 FAs in the reactor core at different positions; effective worth of beryllium rods around neutron trap and a new beryllium rod at irradiation channel 1-4; void effect at neutron trap, irradiation channel 1-4 and cell 6-3, which surrounded by other FAs.

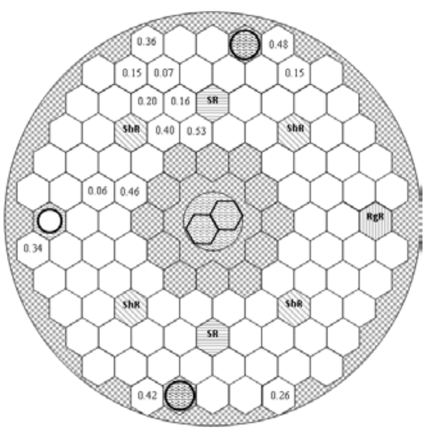

Fig. 4. Effective worth of FAs in the reactor core

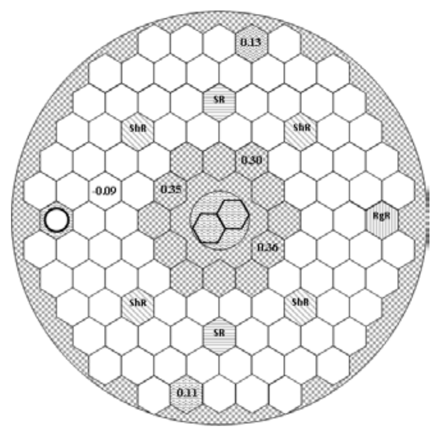

Fig. 5. Effective worth of Be rods in the reactor core

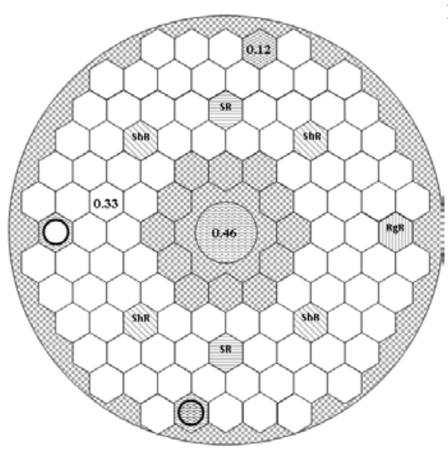

Fig. 6. Measured results of void effect at some positions in the reactor core

The most effective worth of fuel assembly measured at cell 4-5 is 0.53 $. Measured results of effective reactivity of fuel assemblies and Be rods show a quite large tilting of reactor power from North to South direction. Void effect has negative value in the reactor core (cell 1-4 and 6-3) while positive in the neutron trap. Void effect in neutron trap has positive value because almost neutrons coming in neutron trap are thermalized, that is absorption effect of water in neutron trap is dominant compared to moderation effect. The replacement of water by air or decreasing of water density when increasing steadily of temperature introduces a positive reactivity. With the core using HEU fuel also has positive reactivity of void in neutron trap.

Determination of temperature coefficient of moderator

Temperature coefficient of moderator is the most important parameter, demonstrating inherent safety of reactor. To carry out experiment, the temperature inside reactor pool was raised about 10^0C by operating primary cooling pump without secondary cooling pump. To measure temperature coefficient of moderator, criticality of the reactor was established after each increased step of pool water temperature about 2.5^0C. Basing on the change of regulating rod position (due to change of temperature in the reactor core) the temperature coefficient of moderator was determined.

Water heating in reactor pool by operating primary cooling pump took long time so water in neutron trap also heated up and inserted positive reactivity (as explanation in measurement of void effect), as opposed to temperature effect in the reactor core. So, a hollow stainless steel tube 60 mm diameter was inserted in neutron trap to eliminate positive temperature effect of neutron trap.

Basing on measured results, the temperature coefficient of moderator is determined about -9.1×10^{-3} \$/^{o}C. Measured result without steel pipe containing air at neutron trap was about -5.2×10^{-3} \$/^{o}C. Thus, temperature coefficient of moderator including neutron trap still has negative value. Temperature coefficient of moderator of the core loaded with 88 HEU FAs measured in 1984 was about -8.0×10^{-3} \$/^{o}C.

2.2. Energy start-up

On January 6th, 2012 reactor power has been increased at levels of 0.5% nominal power, 10% nominal power and 20% nominal power. At each power level, thermal neutron flux in neutron trap, irradiation channels 1-4, 13-2 and rotary specimen was measured by using Au foil activation method. Also, on January 17th, 2012 thermal neutron flux of positions mentioned above was measured at power level 100%. Measured results of thermal neutron flux at several irradiation positions in the reactor core with different power levels are presented in **Table 2**.

Table 2. Measured results of thermal neutron flux in irradiation positions at different reactor power levels

Irradiation positions	**Power (% Nominal power)**			
	0,5	**10**	**20**	**100**
Neutron trap	1.143E+11	2.063E+12	4.174E+12	2.122E+13
Channel 1-4	5.288E+10	9.719E+11	1.965E+12	8.967E+12
Channel 13-2	4.749E+10	8.542E+11	1.682E+12	N/A
Rotary Specimen	N/A	N/A	N/A	4.225E+12

The experiment to determine the curve built up of Xenon poisoning and then calculating its equilibrium poisoning was conducted from January 9th, 2012 to January 12th, 2012 when the reactor was in 100% nominal power (indicating of control system without adjusting power) .

Next, Iodine hole was also determined from 12 to January 13th, 2012 after reducing power of the reactor from 100% to 0.5% nominal power by monitoring the shift position of regulating rod.

Fig. 7 presents measured results of Xenon poisoning curve and Iodine pit of the above experiment. Xenon equilibrium poisoning and other effects is totally about -1.1 β_{eff} and the maximum depth of Iodine pit determined about -0.15 β_{eff} after 3.5 hours since the reactor was down to 0.5% nominal power. After adjusting thermal power up to 500 kW, during the long operation from March, 12-16, 2012, after the reactor was operated 68 hours at nominal power, total value of poisoning and temperature effects is about -1.32 β_{eff}.

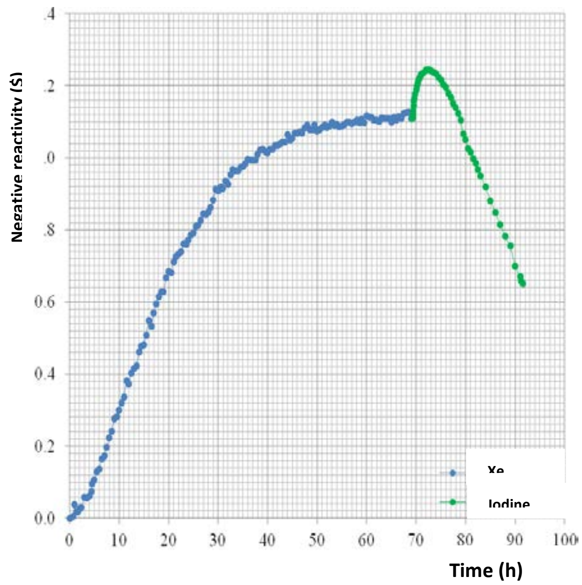

Fig. 7. Negative reactivity insertion by Xenon poisoning with operation time and Iodine pit

In the process of gradually raising power in energy start-up, although power indication on control system was 100% but calculated thermal power of the reactor through flow rate of primary cooling system and difference between inlet and outlet temperatures of the heat exchanger was only 460 kW, smaller than nominal power about 10%. The reason was mainly due to power density of the core using 92 LEU FAs were higher than the mixed core using 104 FAs before. The adjustment to increase thermal power of the reactor was performed by changing the coefficients on the control panel. After adjusting, the reactor was operated to determine thermal power at power setting 100%. The results of thermal power obtained from the next long operation were about 510.5 kW. This value includes 500 kW thermal power of the reactor and about 10 kW generated by primary cooling pump.

After carrying out reactor power adjustment, thermal neutron flux at some irradiation positions in the reactor core and neutron spectrum in neutron trap were measured again by neutron activation foils. Measured maximum neutron flux at neutron trap was 2.23×10^{13} n/cm^2.s (compared with calculated result was $2.14 \div 2.22 \times 10^{13}$ n/cm^2, depending on shim rods position). Those in channel 1-4 and 13-2 were 1.07×10^{13} n/cm^2.s and 8.61×10^{12} n/cm^2.s, respectively.

3. Core and fuel management for DNRR

To validate the REBUS-MCNP system with ENDF/B 7.0 and 7.1 data library, the system should be evaluated by performing calculation almost characteristics of the reactor and compared with the experimental data that were mainly gotten in during reactor startup with LEU fuel. The implemented calculations include: Neutron flux distribution and neutron spectrum; Control rod worths; Effective reactivity of fuel assemblies and beryllium rods in the core; Void effect and temperature reactivity feedback coefficient of moderator; Xenon poisoning effect; Kinetics parameters.

Calculated and experimental relative thermal neutron flux, the results were normalized to unit, at highest neutron flux in neutron trap show that

- In radial direction of the reactor core, discrepancy between calculated results and experimental data is less than 3%, except cell 6-4 and 12-2 with higher difference about 8%.
- In axial direction of fuel centerline, the difference between calculated and experimental results in range from 15 cm to 65 cm is about 4%, top and bottom of fuel have higher differences than 10%.
- In axial direction at neutron trap, the difference between calculated and experimental results is about 4%.

The calculation of the neutron spectrum and absolute neutron flux in the experimental irradiation positions including neutron trap, channel and two-channel wet dry 1-4 7-1 and 13-2 were carried out. The results showed that the difference between the calculated and experimental within 5%.

The effect of the control rod worths of working configuration with 92 LEU fuel assemblies were calculated and compared with experimental data. Experimental results are lower than the calculated results in approximately 7%. The effective reactivity of fuel assemblies and beryllium rods at different positions in the reactor core has been investigated in order to determine the effective of reactivity according to their positions in the reactor core. Calculated and experimental results showed that the difference is just within 0.05$ (or 5 cent).

Xenon poisoning was considered by calculating the present of Xenon in composition of fuel after operating 130 h of the reactor and without Xenon after cooling. The reactivity value has been calculated about 1.30 $ compared with experimental value: 1.23 $, the difference between two values is 5.8%.

DRRBurn code has been designed for using easily through the interaction between the user and display window on the screen. Calculation model and input files for all the codes in the system has been built and evaluated. The code will automatically compose input file with core configuration and composition of fuel material as well as beryllium materials that are updated basing on information about changing of core configuration and historical reactor operation supplied by the user to achieve the consistency in the whole process of core and fuel management, less experienced user can also use the system and avoid possible errors.

The diagram of the DRRBurn computer code system is presented in **Fig. 8** with functions including display, computation, management and storage.

The system has been used for core and fuel management of the DNRR from January, 2012 to March, 2014. Difference calculated and experimental excess reactivity following operation time at maximum nominal power is about 0.08 $ and it shows that the calculated results were quite suitable.

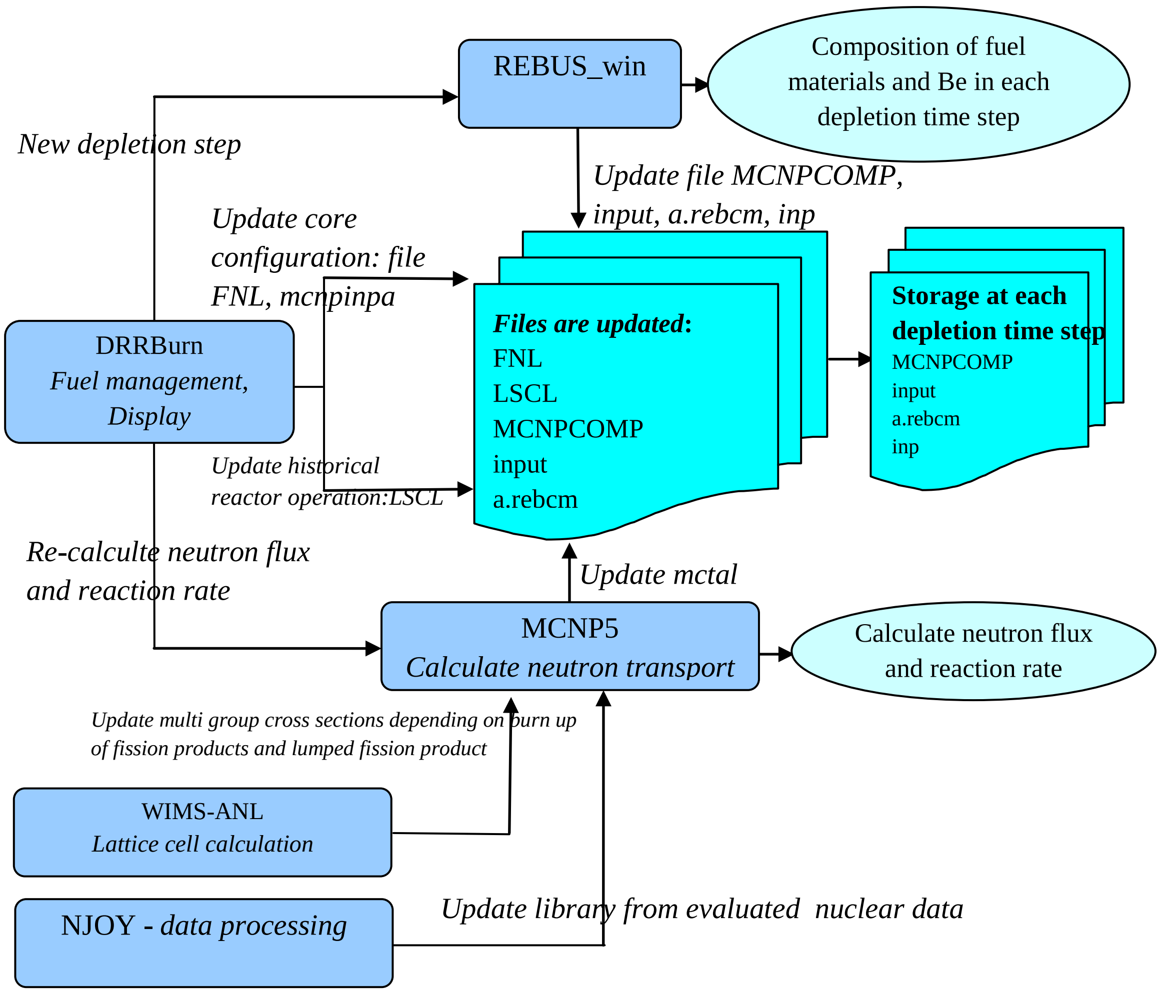

Fig.8. Structure of DRRBurn computer code system for core and fuel management of the DNRR

5. Enhancement for radioisotope production at neutron trap of DNRR

The neutron trap in the center of the reactor core was modified and installed new basket that can be loaded 6 containers. The negative reactivity of irradiation target of 6 containers has been evaluated about 60 to 80 cents. This mean it does not effect to the operation safety but the reactor needs to have enough excess reactivity for maintaining normal operation. The calculated results showed that when increasing target mass 1.5 times the activity of I-131 will increase 1.25 times. If irradiating time (or operating time of the reactor) is about 130 hours comparing with 100 hours, the activity of I-131 target will increase to more than 17.5% and each 10 hours cooling time the activity will be decreased about 3.3%.

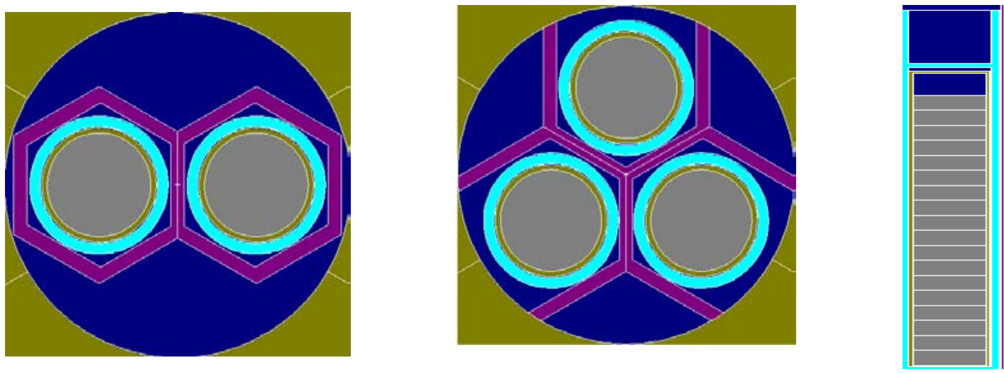

Fig.9. Calculation model before (4 loaded containers) and after (6 loaded containers) changing neutron trap structure for loading irradiation containers

After 2 operating cycles with 110 hours continuously using new neutron trap structure (6 loaded containers), total I-131 activity increased 23% to be compared with before just only 4 loaded containers.

However, other factors effect to the I-131 radioisotopes production such as dimension and material composition of container, target mass, optimize of irradiation positions, ect should be considered more detail to archive effectively in operation and utilization of the DNRR.

6. Conclusion

Start up of DNRR with entire LEU FAs core was implemented following a detailed plan. As a result, physics and energy start up were carried out successfully. The working core with 92 LEU FAs has been operating 72 hours for testing at nominal power. Experimental results of physical and thermal hydraulics parameters of the reactor during start up stages and long operation cycles at nominal power showed very good agreement with calculated results. On the other hand, experimental results of parameters related to safety such as peaking factor, axial and radial neutron flux distribution of reactor core, negative temperature coefficient, temperature of the reactor tank, temperature at inlet/outlet of primary cooling system and secondary cooling system, etc also have been investigated. It could be confirmed that current core configuration with 92 LEU FAs meets the safety and exploiting requirements. Measured neutron flux in irradiation positions of the reactor after full core conversion also showed that the reactor core using LEU fuel is not much different than previous core using HEU fuel.

REBUS-MCNP linkage system has been used for core and fuel management of DNRR after validating by comparing calculated results with experimental data of LEU cores. The DRRBurn computer code was programmed to manage calculations, set up computational model and input files as well as storage data to ensure compliance with the requirements of the IAEA in core and fuel management of DNRR.

The calculated and experimental results of I-131 radioisotope production showed that the efficiency in application of the reactor after full core conversion from HEU to LEU fuel. The activity of I-131output increased significantly after the redesign of the basket in the neutron trap and loading more target mass.

Acknowledgments

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7. References

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