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State of Work on Calculation Feasibility of the IR-8 Reactor Conversion to LEU fuel

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ABSTRACT

As a part of the RERTR program NRC “KI” studies the feasibility of the IR-8 reactor conversion to LEU fuel with financial support from U.S. Department of Energy. Argonne National Laboratory provides analytical support to that work in the performance of the analysis tasks with verification purposes. At the beginning of 2012 feasibility studies for the conversion of the IR-8 reactor to LEU fuel were finished. It was made a conclusion about technical possibility of the IR-8 reactor conversion to LEU fuel. Currently it is supposed to assess safety of IR-8 reactor during conversion to LEU fuel. These studies include implementation of three tasks: (1) neutronic and thermal-hydraulic calculations, (2) radiation safety and (3) analysis of possible accidents.

1. Introduction

After finishing the calculating analysis of neutronic and thermal-hydraulic characteristics of the IR-8 reactor [1] with HEU [2] NRC “Kurchatov Institute” under the RERTR program began to study the calculated characteristics of the reactor with LEU fuel [3]. Three types of FAs with LEU fuel were examined: IRT-4M, IRT-3M and IRT-U [4]. IRT-3M FA with U-9%Mo LEU was chosen for the further investigations of the IR-8 conversion. Detailed full-scale models of the reactor with this fuel were created for neutronic calculations using MCU-PTR code [5]. The ASTRA [6] code was used for thermal-hydraulic calculations of the IR-8 reactor.
A strategy of the reactor conversion to LEU fuel in the regime of partial reloading was determined.

2. Brief description of the IR-8 reactor

The IR-8 is a pool type research reactor with power up to 8 MW. It uses light water as moderator, coolant and top shield. Recently, the reactor was operating on power up to 6 MW. After replacement of heat exchangers in 2012 the permission to operate on power of up to 8 MW was received.
The IR-8 core consists of 16 IRT-3M FAs with tubular elements of square cross section. The fuel is UO_2 90% enrichment. The core and beryllium reflector are placed in the vessel on the supporting grid near bottom of the pool at ~11 m depth. The 13 rods with boron carbide absorber are used as control rods of CPS.

There are 12 horizontal experimental channels (beam tubes) in the reactor for carrying out fundamental and applied researches in nuclear physics, physics of condensed state and materials science. The IR-8 reactor construction provides possibility of installation of many vertical experimental channels (VEC) for irradiation of fuel, of structural materials and other are fulfilled.

3. MCU-PTR code description

MCU-PTR code with the data bank MDBPT50 is developed for simulation of neutron and photon particles transport by analog and non-analog Monte Carlo methods based on evaluated nuclear data for the changes in the nuclide composition of materials in interaction with neutrons.
A constant provision of the MCU-PTR code is MDBPT50 database. All necessary characteristics of nuclides being set as burnable in initial data are located in BURN part of the data bank. BURN library contains approximately 1100 nuclides. The code was verified and validated (V&V) and currently the procedure of certification by Rostehknadzor (Russian regulation body) is carried out.

4. Calculating models of IR-8 reactor loadings

The geometry of the calculating model (Fig 1…3) reproduces the actual structure of the core and reflector of the reactor IR-8, including the location in which the experimental vertical and horizontal channels for the analysis occurring in the core and reflector neutron-physical processes.
In order to simplify the whole space outside the reactor vessel is filled with water. Reactor tank and biological shield are not counted. This model consists of ~ 16000 registration zones, each characterized by a specific material with a given initial isotopic compositions.
The core is divided to 30 layers on height, each thickness is 2 cm. The fuel in each FA is divided to 180 zones (30×6 fuel elements).

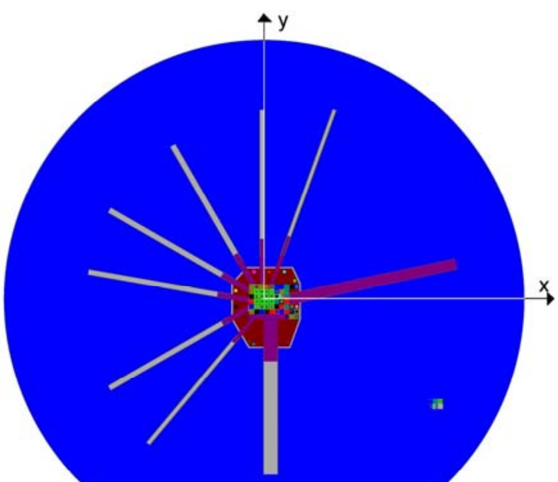


Fig. 1. Cross section of calculating model

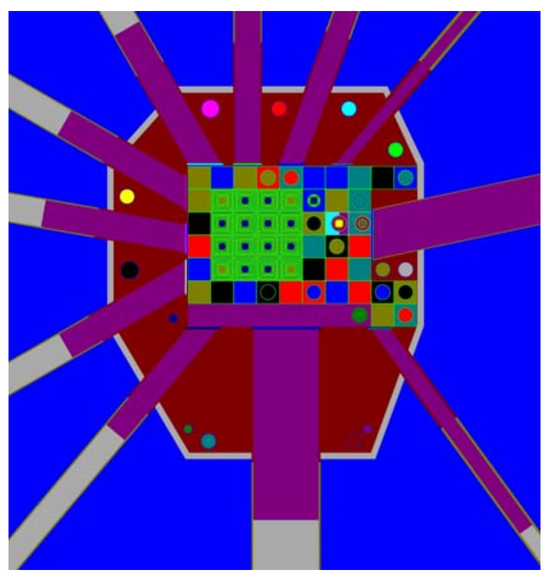


Fig. 2. Cross section of reactor with HEC's

5. Feasibility study for the IR-8 reactor conversion to LEU

5.1. The choice of FA with LEU fuel for the IR-8 reactor conversion

Three variants of IRT type FAs with LEU were analyzed: IRT-4M (UO_2), IRT-3M (U-9%Mo) from tubular fuel elements and also IRT-U (U-9%Mo) from rod fuel elements. FAs with tubular fuel elements are used at the IR-8 reactor since 1965 (FA IRT-M, -2M). IRT-3M FAs (90% enrichment) are used since 1981. A great experience of operation and calculations of tubular FA has been accumulated. These assemblies have shown their high reliability even when the fuel burn-up is over 60%. Besides that FA with tubular fuel elements were tested hydraulically and velocities in them are well known [7]. Therefore, it was decided not to consider IRT-U FA for the IR-8 conversion.

Calculated analysis showed that IRT-4M FAs does not provide the necessary duration of the reactor cycle, which was achieved by using IRT-3M FAs 90% enrichment (Fig. 4, Table. 1).

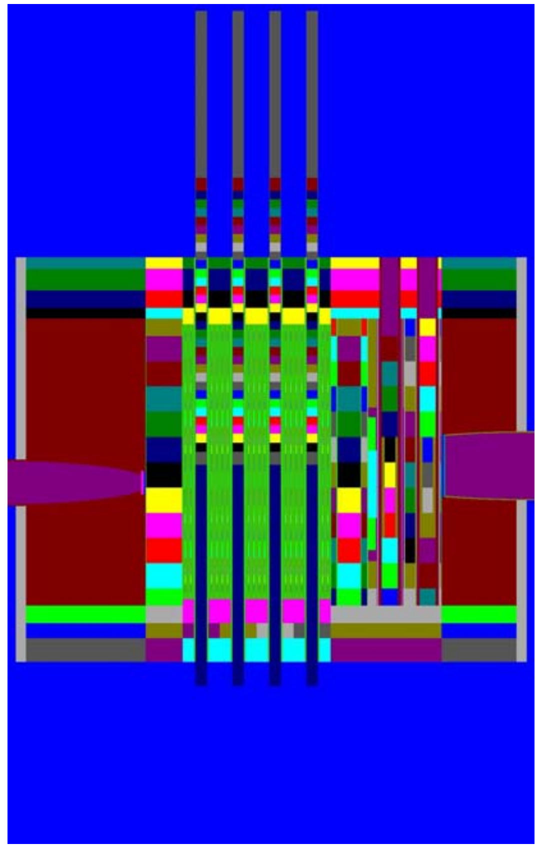
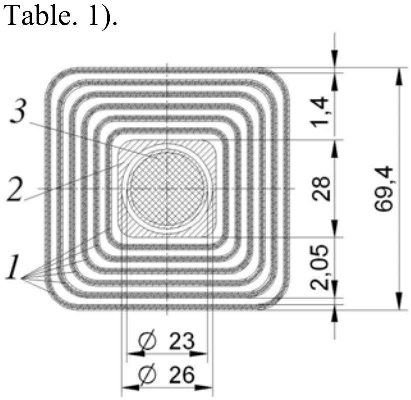


Fig. 3. Longitudinal section of reactor calculating model



1 – fuel elements;
2 – channel of the CPS rod; 3 – CPS rod

Fig. 4. Cross section of IRT-3M FA with control rod

FA	IRT-3M	IRT-3M
Fuel enrichment, %	90	19.7
^{235}U content, g	264	352
Fuel meat thickness, mm	0.4	0.5
Active length of meat, mm	600	600
Fuel meat composition	UO_2-Al	U- 9%Mo
Uranium density in the meat, g/cm^3	1.07	5.2
Cladding material	Al alloy CAB-1	Al alloy CAB-1
Cladding thickness, mm	0.5	0.45
Number of fuel elements in FA	6	6
Heat-exchange surface, m^2	1.42	1.38

Tab. 1: The main parameters of IRT-3M type FA

5.2. Neutronic parameters of the reactor

As a reference point for determine the parameters of the reactor the BOEC with HEU and LEU was selected. On the basis of the reactor loading #35 with HEU (Fig. 5) and ampoule rigs in 6-4 and 7-3 cells detailed full-scale neutronic models of IR-8 loadings with HEU and/or LEU fuel have been created. The reactor conversion from HEU to LEU was carried out in the regime of partial reloadings in each of them two “fresh” FA’s with LEU instead of

HEU were loaded into the center of the core. The cycle length was adjusted by means of calculation so that the values of excess reactivity at the end of equilibrium cycles were the same for HEU and LEU cores. Thus, the reactor was outputted into the equilibrium cycle (Fig. 6) and the main neutronic parameters of this cycle were determined (Table 2).

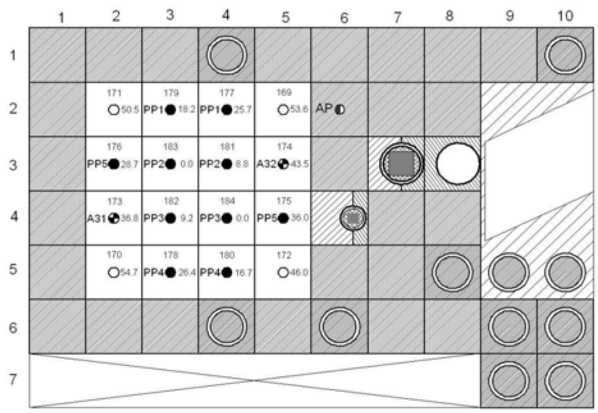


Fig.. 5 Configuration of the IR-8 loading # 35 (HEU fuel).
Excess reactivity – 8.2 %Δk/k

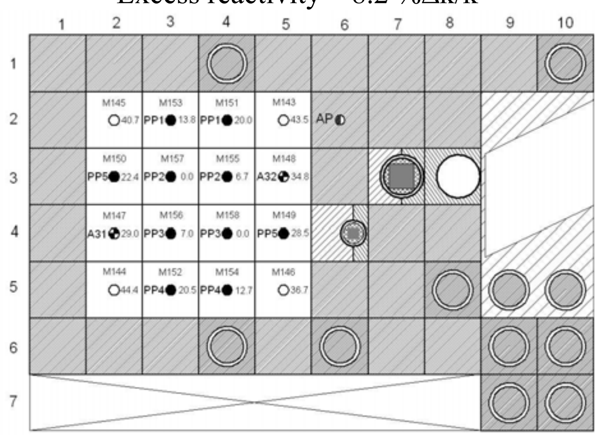
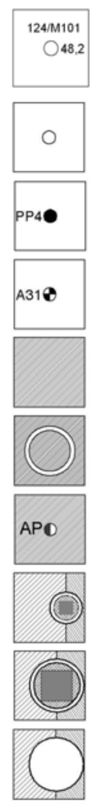


Fig. 6 Configuration of IR-8 loading #29M (16 FA with LEU fuel).
Excess reactivity –7.1 %Δk/k

Legend



FA number (HEU/HEU) average burn-up U^{235} in FA, %

6-tube FA IRT-3M

6-tube FA IRT-3M with shim rod

6-tube FA IRT-3M with safety rod

whole beryllium block

beryllium block with hole d=48 mm and plug d=44 mm

beryllium block with automatic regulating rod

gamma-radiation shield with AR in 6-4 cell

gamma-radiation shield with AR in 7-3 cell

gamma-radiation shield for AR in 8-3 cell

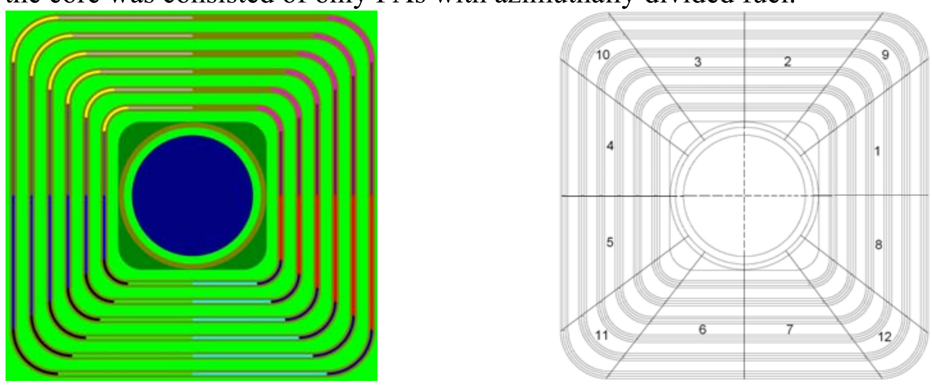
Parameter	Value		Parameter	Value	
	HEU	LEU		HEU	LEU
Uranium enrichment, %	90	19.7	Max. thermal neutron flux, cm^{-2} s^{-1}:		
^{235}U loading in FA, g	264	352	- in the core (2-2 cell)	$1.5 \cdot 10^{14}$	$9.5 \cdot 10^{13}$
Cycle length, FPD's	31.3	36.3	- in the reflector (cell 2-1)	$1.6 \cdot 10^{14}$	$1.3 \cdot 10^{14}$
Cycles per year	5.3	4.6	- on the beam tube end face	$1.3 \cdot 10^{14}$	$1.2 \cdot 10^{14}$
FA consumption per year	10.7	9.2	- in the VEC	$7.1 \cdot 10^{13}$	$6.7 \cdot 10^{13}$
Excess reactivity, %Δk/k: - the beginning of cycle - the end of cycle	 8.2 1.8	 7.1 1.8	Max. fast (E > 0.5 MeV) neutron flux in the AR (6-4 cell), cm^{-2} s^{-1}	$2.7 \cdot 10^{13}$	$2.5 \cdot 10^{13}$

Tab. 2. Main neutronic parameters of the IR-8 reactor at the 8 MW power

5.3. Thermal-hydraulic parameters of the reactor

To increase the accuracy of the calculations and for taking into account the azimuthally irregularities within the FA as well as across the core at the calculating model all fuel elements of FA were additionally divided into 12 sectors by azimuth (Fig. 7). Thus, the number of registration zones with fuel increased from 2880 to 34560.

Calculations for determining the fuel burn-up for the equilibrium cycles for the reactor operating in the regime of partial reloading of two FAs with azimuthally divided fuel elements were carried out. (loadings # 29M÷37M). At the beginning of the reactor operating cycle #36M the core was consisted of only FAs with azimuthally divided fuel.



Cross section of the calculating model with the partition of FA into 12 sectors

Numbering of the sectors

Fig. 7. Calculating model of IRT-3M FA with LEU

For determining thermal-hydraulic parameters of the reactor with LEU fuel two starting loadings of the reactor equilibrium cycle were analyzed. In the first case two “fresh” FAs were loaded into 3-4 and 4-3 cells (loading #37M) and in the second case “fresh” FAs were loaded into 3-3 and 4-4 cells (loading #36M). Both loadings were examined in xenon-free and equilibrium xenon concentrations states. For these loadings power densities for all fuel zones were calculated. Basing on obtained results the most thermally stressed FAs were determined and thermal-hydraulic calculations were carried out. The most thermally stressed was the beginning of the equilibrium cycle with “fresh” FAs in 3-3 and 4-4 cells. The maximum of power density located in an external fuel element of the FA in 3-2 cell in sector #3 on the border with a beryllium reflector (Fig. 8).

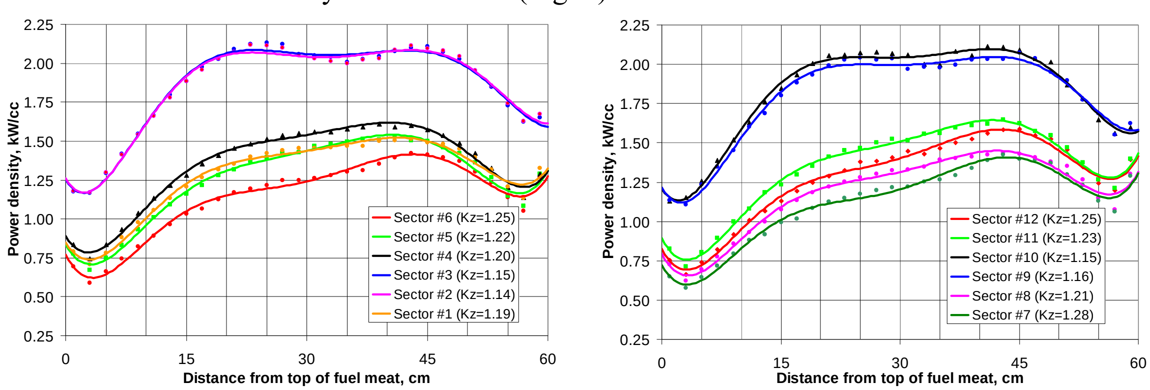


Fig. 8. Distribution of power density on the height of the fuel element

Main thermal-hydraulic parameters of the reactor equilibrium cycle are shown in Table 3.

Parameter	Value	Parameter	Value	
			HEU	LEU
Coolant flow, l/s: through the reactor/all FAs	208/105	Power of FA in the most thermally stressed cell, kW	580	620
Coolant velocity in the gaps FA (average), m / s	2.4	Maximum calculating power density in meat, MW/m^3	3440	2840
Coolant pressure at the core inlet, MPa	0.186	Maximum heat flux on the surface of fuel elements, kW/m^2	640	690
Pressure drop across the core, MPa	0.022	Maximum design temperature on the surface of fuel elements, °C	107	106
Coolant temperature at the core inlet, °C	до 50	The temperature of onset nucleate boiling (ONB),°C	130	131
Coolant temperature rise in the reactor, °C	6.9	Safety margin to onset nucleate boiling (ONB)	1,40	1,45

Tab. 3. Thermal-hydraulic parameters of the IR-8 reactor at the 8 MW power

6. Safety assess of the reactor IR-8 reactor during conversion to LEU fuel

Currently the studies to assess safety of IR-8 reactor during conversion to LEU fuel were started. These studies include implementation of three tasks: (1) neutronic and thermal-hydraulic calculations, (2) radiation safety and (3) analysis of possible accidents.

6.1. Neutronic and thermal-hydraulic calculations

This part of analysis includes: (a) calculations of the main neutronic and thermal-hydraulic parameters for the equilibrium loading of the core with HEU; (b) calculations of the main neutronic, reactivity and thermal-hydraulic parameters for different loadings of the core during conversion form HEU to LEU.

The main neutron and reactivity characteristics will include: the reactivity feedback coefficients (depending on the temperature and density) and the other parameters of the neutron kinetics (lifetime of prompt neutrons and the effective fraction of delayed neutrons) and c) the integral and differential reactivity worth of control rods.

6.2. Radiation safety

This part of analysis includes: (a) calculations of gaseous radioactive emissions into the environment during normal operation; (b) analysis of the possible effects of the IR-8 reactor radiation to the population; (c) calculate accidental release of iodine radionuclides in the environment associated with the melting of a FA.

6.3. Analysis of postulated accidents

Analysis is to be performed for various transients, up to and including design basis and beyond design basis as required by the Regulatory Body (RTN). The list of initiating events is made on the basis of the existing SAR of IR-8 reactor for the HEU fuel, and the recommendations for research reactor safety of the IAEA (Ref: Del-6.2 Sec-2.2.4).

Analysis of possible accidents includes: (a) Analysis of the pre-emergency accidents; (b) Analysis of the design basis accidents (DBA); (c) Analysis of the beyond DBA.

Conclusion

Feasibility studies of the IR-8 reactor conversion from HEU to LEU show [8]:

1. Decrease of neutron flux
   - ➢ Maximum fast (E>0.5 MeV) neutron flux:
     - reduced up to 4% in ECs of the first row of the reflector;
     - almost the same in ECs of the core, VECs and beam tubes.
   - ➢ Maximum heat neutron flux reduced up to:
     - 5% in VECs; -10% in beam tube ends; -19% in ECs of the reflector first row;
     - 12 % in the AR samples; -37% in ECs of the core.

2. For partial compensation of neutron flux decreasing due to the conversion to LEU fuel the replacing of “poisoned” removable beryllium blocks is necessary.

3. Operation of the reactor with LEU fuel (IRT-3M, U-9%Mo) is possible at the power of up to 8 MW. All this allow to make a conclusion about technical feasibility of the IR-8 reactor conversion to LEU fuel.

Currently the studies to assess safety of IR-8 reactor during conversion to LEU fuel were started. Calculations of the main neutronic, reactivity and thermal-hydraulic parameters for the equilibrium loading of the core with HEU actually will be finished.

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References

[1] Ryazantsev E.P., Nasonov V.A., Egorenkov P.M. et al. Contemporary Status and Employment Perspectives of the IR-8 Reactor of RRC KI. International Scientific Conference "Research Reactors in the XXI Century." Moscow, June 20-22, 2006. Ed. NIKIET, M., 2006.

[2] Nasonov V.A., Erak D.Y., Taliev A.V., Pesnya Y.E. The ongoing status of the project on feasibility studies for the conversion of the IR-8 reactor to LEU fuel. Proceedings of the 2011 International Meeting on RRFM. Roma,Italy, March 20 – 24, 2011.

[3] Pesnya Y.E., Erak D.Y., Nasonov V.A., Taliev A.V. Neutronic parameters of the IR-8 reactor core consisting of IRT-3M type FAs with U-9%Mo LEU fuel being analyzed. Proceedings of the 2011 International Meeting on RERTR. Santiago, Chile, October 23-27, 2011.

[4] Nasonov V.A., Ryazantsev E.P., Taliev A.V., Egorenkov P.M. The Work under “The Program Reduced Enrichment of Fuel for Research Reactors” to 19.7% in RRC “Kurchatov Institute” – Atomic Energy, 2010, v.108, is. 3, p. 299-303.

[5] Alexeev N.I., Gomin E.A., Marin S.V., Nasonov V.A., Shkarovsky D.A. MCU-PTR Code for Precision Calculation of Pool and Tank Types Research Reactors. - Atomic Energy, vol. 109, is. 3, 2010, p. 123-129.

[6] Taliev A.V. Modernized ASTRA code for thermal calculation of research reactor FAs consisting from tubular coaxial fuel elements. Preprint IAE-6405/5, Moscow, 2006.

[7] Nasonov V.A., Ryazantsev E.P., Taliev A.V., Yashin A.F. Water velocities determination in the gaps of IRT-3M and IRT-4M FA. Atomic Energy, 2011, v.110, is. 6, p. 317-321.

[8] Erak D.Y., Nasonov V.A., Pesnya Y.E. Calculations for the IR-8 reactor conversion to LEU fuel. Report on International Meeting “Research Reactor Fuel Management”-RRFM-2012, Prague (Czech Republic) March 18-22, 2012.