LVR-15 REACTOR PERFORMANCE AND TRANSFORMATION TO LOW ENRICHED FUEL.

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ABSTRACT

Experimental research reactor LVR-15 situated in Nuclear Research Institute Rez, plc. has been utilized since 1957. The present reactor nominal power is 10 MW. Standard reactor cycle is 21 days and the reactor operates 8 – 10 cycles per year. The State Office for Nuclear Safety newly licensed the reactor till 2014.

The reactor is of multi-purpose use. The basic research is carried out using horizontal neutron beams and one of them is used for development and application of the boron neutron captures therapy for brain tumors. Mostly material testing of PWR and BWR specimens is performed in high-pressure loops and irradiation rigs operated in the reactor. Several vertical channels serve for production of neutron transmutation doped silicon and isotopes production for medical purposes.

Reactor was originally designed with EK-10 fuel type and consequently reconstructed for HEU with the 80% ^{235}U enrichment in 1987-89. In period 1987-1989 the reactor undertook the second reconstruction to enhance its experimental and commercial utilizing. Later, the fuel enrichment was changed to 36 % with the use of the IRT-2M fuel type.

The neutron-physical characteristics of the reactor core and fuel cycle design and analyses are carried out using the WIMS-D4m, NODER and OMEGA programs. The codes were used for a preliminary evaluation of essential changes of main neutronic characteristics of the LVR-15 core with the prospective conversion to low enriched fuel (LEU). Three types of FA-s has been assessed: 1) currently used IRT-2M(36%), 2) IRT-3M(20%), 3) IRT-4M(20%) and results are presented in the article.

1. Introduction

The LVR-15 research reactor was designed in 1957 and is situated in the Nuclear Research Institute Rez plc. The reactor has maximal power 10 MW and is operated in 21 days irradiation cycles, with 8 - 10 cycles per year. The fuel assembly consists either of 4 concentric tubes or 3 concentric tubes with a control rod in the center of the assembly. The fuel IRT-2M contains 36 % enriched uranium.

The reactor is used as a multipurpose facility [1] and it enable to perform research and services in the following areas:

- Material irradiation and testing in reactor loops and rigs,
- Production of radiation doped silicon,
- Regular production of radioisotopes for the radio pharmaceuticals and technical radiation sources,
- Irradiation devices for special irradiation,
- Pneumatic rabbit for activation analysis,
- Development of boron neutron capture therapy at the thermal column channel,
- Neutron physics research (e.g. neutron diffraction for different purposes, SANS) at reactor horizontal channels.

Loop experiments and material testing in rigs belongs among the main research activities performed at the reactor. Material research on the loops and rigs is mainly used for material embrittlement, corrosion, and material-coolant interaction studies that are very important for NPP utilization, plant life aging, and general safety.

The irradiation facilities are complemented with well-equipped hot cells that are also used for removing of irradiated specimens from experimental devices and their post-irradiation examination. More special post-irradiation examinations of irradiated specimens are performed in either hot or semi-hot cells situated in another building near the reactor [2].

Reactor was originally designed with EK-10 fuel type and consequently reconstructed for HEU with the 80% ^{235}U enrichment. In period 1987-1989 the reactor undertook the second reconstruction to enhance its experimental and commercial utilizing. After that it could be potentially operated on the level up to 15MW. Later, in 1995, the fuel enrichment was changed to 36 % with the use of the IRT-2M fuel type. The transformation took several reactor cycles using gradually different mixed core configurations. General demand to decrease the enrichment of the used fuel require to asses neutron-physical characteristics of the core with available LEU of Russian production. A preliminary evaluation of expected essential changes of main neutronic characteristics of the LVR-15 core connected with the prospective conversion to low enriched fuel (LEU) were performed on the base of comparison of 1) currently used IRT-2M(36%), 2) IRT-3M(20%), 3) IRT-4M(20%). The results of calculation study are presented in the article.

2. Reactor loops

The research in the reactor loops is mostly connected with environmental degradation processes as corrosion, mechanical, and radiation effect. The loops also enable to study cladding - coolant interaction namely effect of water chemistry components, deposition of corrosion products, and corrosion of zirconium alloys with and without radiation. Basic in-

pile water chemistry and corrosion monitoring technique are developed and implemented at the reactor loops to measure and control important parameters as conductivity, pH, redox potential, dissolved hydrogen and oxygen, water impurity content, corrosion potential, contact electrical resistance [3]. Detailed description of loops and experiments is presented in [4], research in the field of stress corrosion cracking of ferritic and austenitic steels is presented in [5].

2.1 The RVS-3 loop

The RVS-3 loop simulating the PWR/VVER environment is designed for material and radioactivity transport investigation. The loop was recently reconstructed with the aim to upgrade or replace principal loop components and especially and simplify the loop primary circuit. The loop is being prepared for experiments under the EFDA (European Fusion Development Agreement) the material development program. Operational parameters:

Pressure 16.5 MPa
Temperature 345°C
Water flow rate 10 000 kg/hr
Neutron flux ~$1x10^{18}$ n/m^2s
Electrical heating capacity 100 kW

2.2 The RVS-4

The RVS-4 loop was designed for the study of PWR environment and fuel cladding interaction with the electrically heated fuel model rods. At present the loop is under functional and start-up tests for study of water chemistry of VVER reactors. Operational parameters of the loop are identical with RVS-3 except for the flow rate – 2 000 kg/hr.

2.3 Zinc loop

Zinc loop is used for the study of activity transport in primary circuits and corrosion reduction with the zinc injection. The loop was designed as an integrated one modeling primary curcuit and stem generator. Operational parameters of the loop are identical with RVS-3 except for the flow rate – 2 000 kg/hr.

2.4 BWR-1 loop

The BWR-1 loop is usually used for testing and development of in-pile water chemistry and corrosion monitoring technique and their components [6, 7] in the BWR environment. Operational parameters:

Pressure 10 MPa
Temperature 300°C
Water flow rate 2 000 kg/hr
Neutron flux ~$1x10^{18}$ n/m^2s

2.5 BWR-2 loop

The BWR-2 loop is designed for stress corrosion cracking tests of reactor pressure vessel (RPV) and internal steels materials specimens (non irradiated and pre-irradiated in reactor rigs)

under simultaneous BWR coolant and irradiation conditions. The cycling and constant load modes can be applied at the specimens. Its test channels can be used in two modifications:

- a channel situated near the core (with the possibility to test specimens up to 2T CT),
- a channel situated in the core (with the possibility to test specimens up to 1T CT).

Operational parameters:

Pressure 12 MPa.
Temperature 300°C.
Water flow rate 8 000 kg/hr.
Force applied to the specimen 150 kN.

3. Reactor rigs

In the reactor two types of rigs can be operated – cylindrical and flat one. Specially designed heating system arranged into sections together with thermocouples ensures full temperature control of the rig. The rigs enable to irradiate four types of specimens: tensile, CT, round CT, and Charpy-v specimen.

3.1 BLANKA type rig

BLANKA type rig was developed for the research in the ADTT project. The rig enables the irradiation of samples (molten salts with additives) with temperature up to 800 °C. The rig is not equipped by the external heating sections, the irradiation temperature is generated with reactor gamma heating and controlled either with the reactor power or inert gas pressure in the insulation thimble. The temperature is monitored with thermocouples situated in the irradiation section. The gas samples can be taken for their further analysis.

3.2 CHOUCA reactor rigs

The CHOUCA reactor rigs (of French production) are used for irradiation of reactor pressure vessel materials for the research of their material properties degradation under radiation and temperature. The specimen holder consists of six sections and is inserted in the rig heating channel (6 heating sections are situated along the rig height and each section has its own thermocouples) which ensures irradiation temperature range from 200±10 °C to 350±10 °C in inert gas [8]. Ar, He or mixture Ar and He can be used as an inert gas [9]. A rig specimen holder enables to irradiate, e.g.:

Charpy V-impact specimens,
Tensile specimens of different types up to 12 mm in diameter,
Slow strain rate test specimens,
Fracture toughness 0.5T CT specimens.

3.3 FLAT rigs

The FLAT rigs were designed to enable irradiation of larger specimens in the inert gas in NRI (Fig. 1). Separate heating sections are situated along the rig height and each section has its own thermocouples which ensures to control the irradiation temperature range from 200±10 °C to 350±10 °C in inert gas. Several modifications of the rig enable the irradiation e.g.:

Up to six 1T CT specimens,
Four 2T CT specimens,

Batch of different specimens of lower dimensions which volume lies between above-mentioned volumes.

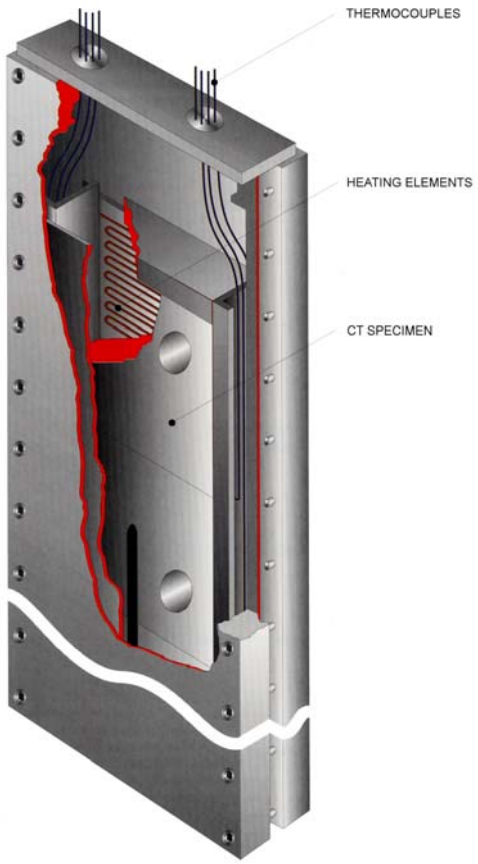

FIG. 1. 2T-CT Irradiation Rig.

4. Assessment of IRT-xM fuel types in LVR-15 conditions.

Short preliminary study has been performed to give essential information about changes of main neutronic characteristics of LVR-15 core loaded with different type of Russian tubular fuel assemblies. Three types of FA-s has been assessed in the study as follows:

- IRT-2M(36%) - currently in use,
- IRT-3M(20%) - proposed
- IRT-4M(20%) – proposed.

Basic characteristic of them (and relatives types) can be seen in next table.

Table 1. Characteristics of the investigated fuel types.

type	Fuel composition	enrichment [%]	Mass U235 [g]	Mass U [g]	tube	Thikness f.meat [mm]	Thikness w.gap [mm]	Density *) of U in meat [g/cm3]
IRT-2M-80	Umet & Al	80	171	214	2	0.4	4.5	1.317
IRT-2M-36	UO2	36.13	230	637	2	0.7	4.5	2.241
IRT-3M-36	UO2	36.13	352	974	1.4	0.5	2.05	2.416
IRT-3M-20	U9Mo	19.75	400	2025	1.4	0.5	2.05	5.022
IRT-4M-20	UO2	19.75	300	1519	1.6	0.7	1.85	2.693

***) Calculated**

4.1 Calculational Tools.

Analyses of MEU and LEU loaded cores were performed using the 3-D nodal diffusion code *NODER*. Four-group burnup dependent cross sections were generated with *WIMSD4m* (*ENDF/B-V*) code. The energy grouped Monte Carlo criticality code *OMEGA* (*Thermos, BNAB78*) was utilized to validate the calculational methodology and to spectrum calculation in fresh cores.

4.2 Reactivity excess in fresh cores

As the configuration of the core to be analyzed, the arrangement shown on the picture (Fig.2) was chosen for the study. This universal arrangement is frequently used for the LVR-15 analyses, because the real operational arrangements vary from cycle to cycle, but they are in most cases only similar variations of it.

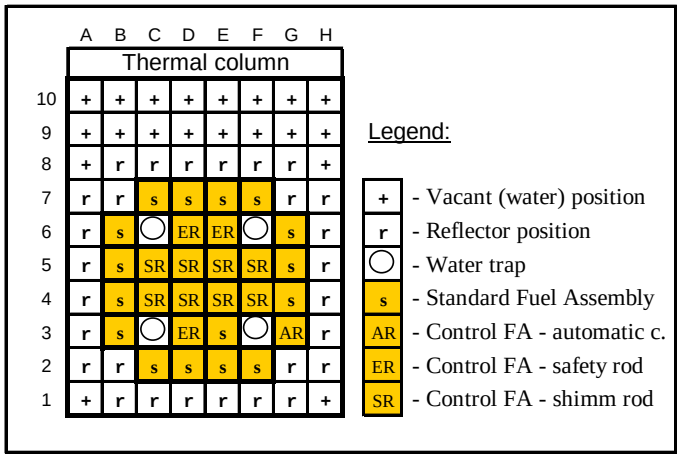

Fig. 2. Layout of the KS28 model core

The reactivity excess and summary control rods worth has been calculated in fresh, cold, unpoisoned core to demonstrate its essential properties from point of view neutron criticality safety.

Table 2. Reactivity and k_{eff} of the KS28 core model for the investigated fuel types.

fuel type	k_{eff} Rods position Up	k_{eff} Rods position Down	reactivity Excess %δk/k	reactivity Worth %δk/k	W/E -
IRT-2M(36%)	1.1526	0.9546	13.24	18.00	**1.36**
IRT-4M(20%)	1.1199	0.9372	10.71	17.41	**1.63**
IRT-3M(20%)	1.1682	0.9974	14.40	14.66	**1.02**

4.3 Fuel exploitation

Cycle length and fuel consumption has been calculated according to the appropriate model of depleted core that is presented in the Fig. 3. Several parameters were used, e.g. length of the main macrocycle (MMC), number of cycles per MMC and number of average fuel assemblies (FA) per cycle, to asses the balance of the fuel consumption and cycle length in the equilibrium core. The results are summarized in the Table 2.

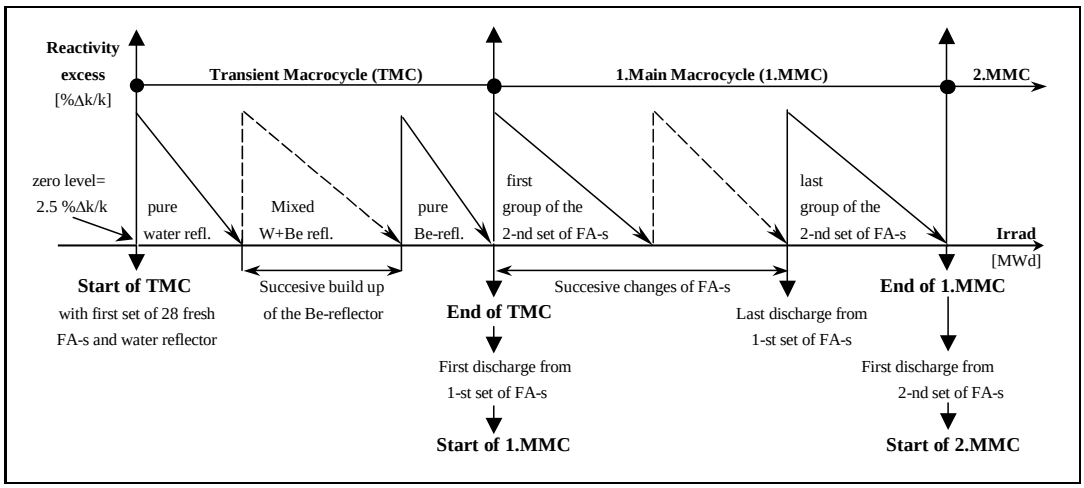

Fig.3. Model of the cycling.

Table 2. Balance of the fuel consumption and cycle length in equilibrium core

Fuel type	**Length of Main Macrocycle (MMC)** /MWd/	Relative length of MMC -	**Number of cycles (190MWd) per MMC**	**Number of average FA's per cycle**
IRT-2M(36%)	2570	**1.00**	13	**1.90**
IRT-4M(20%)	3420	**1.43**	18	**1.55**
IRT-3M(20%)	5060	**1.97**	26	**1.05**

Results of calculations show reasonable differences between the studied FA performances. The LEU with the higher amount of ^{235}U in FA gives better results in defiance of the composition (UO_2 or U_9Mo). The MMC is longer and consumption of the FA's per cycle is lower.

4.4 Neutron group distribution changes

Next field of comparison is the neutron flux in equilibrium core with beryllium reflector at 10MW nominal power (Fig.4). For this purpose, set of cells important from point of view of reactor applications was selected. These 'so called' Check Cells are fully filled with water in one case and with aluminum in second one, respectively. Ranges of neutron energies used for comparison of fluxes are as follows

- Thermal: 0<En<0.625eV,
- Fast: 0.821<En<10MeV,
- Total: 0<En<10MeV.

Table 5. Neutron fluxes in the Check Cells

Average neutron flux density [1/cm2/s/10MW x1.e-13] in cell with dimensions (7.15 x 7.15 x 60.0)cm												
Check position :	**B7**											
Material of CC :	**Water**						**Alu-block**					
Fuel type	**thermal**	dev[%]	**fast**	dev[%]	**total**	dev[%]	**thermal**	dev[%]	**fast**	dev[%]	**total**	dev[%]
IRT-2M(36%-230)	7.756	-	2.483	-	16.387	-	6.079	-	2.740	-	16.053	-
IRT-3M(20%-400)	6.857	-11.6	2.696	8.6	16.421	0.2	4.917	-19.1	2.986	9.0	16.041	-0.1
IRT-4M(20%-300)	7.561	-2.5	2.785	12.2	17.559	7.2	5.561	-8.5	3.087	12.7	17.248	7.4

Average neutron flux density [1/cm2/s/10MW x1.e-13] in cell with dimensions (7.15 x 7.15 x 60.0)cm												
Check position :	**C3**											
Material of CC :	**Water**						**Alu-block**					
Fuel type	**thermal**	dev[%]	**fast**	dev[%]	**total**	dev[%]	**thermal**	dev[%]	**fast**	dev[%]	**total**	dev[%]
IRT-2M(36%-230)	11.373	-	5.692	-	29.091	-	7.831	-	6.135	-	27.111	-
IRT-3M(20%-400)	8.306	-27.0	5.957	4.6	27.220	-6.4	4.693	-40.1	6.281	2.4	24.932	-8.0
IRT-4M(20%-300)	9.411	-17.3	6.226	9.4	29.395	1.0	5.555	-29.1	6.589	7.4	27.080	-0.1

Average neutron flux density [1/cm2/s/10MW x1.e-13] in cell with dimensions (7.15 x 7.15 x 60.0)cm												
Check position :	**E9**											
Material of CC :	**Water**						**Alu-block**					
Fuel type	**thermal**	dev[%]	**fast**	dev[%]	**total**	dev[%]	**thermal**	dev[%]	**fast**	dev[%]	**total**	dev[%]
IRT-2M(36%-230)	5.302	-	0.557	-	7.672	-	5.283	-	0.626	-	7.989	-
IRT-3M(20%-400)	5.139	-3.1	0.581	4.1	7.688	0.2	5.153	-2.5	0.658	5.2	8.100	1.4
IRT-4M(20%-300)	5.469	3.1	0.599	7.5	8.116	5.8	5.524	4.6	0.680	8.6	8.594	7.6

Average neutron flux density [1/cm2/s/10MW x1.e-13] in cell with dimensions (7.15 x 7.15 x 60.0)cm												
Check position :	**H5**											
Material of CC :	**Water**						**Alu-block**					
Fuel type	**thermal**	dev[%]	**fast**	dev[%]	**total**	dev[%]	**thermal**	dev[%]	**fast**	dev[%]	**total**	dev[%]
IRT-2M(36%-230)	7.853	-	2.163	-	15.327	-	6.471	-	2.399	-	14.986	-
IRT-3M(20%-400)	7.153	-8.9	2.326	7.5	15.387	0.4	5.566	-14.0	2.623	9.3	15.098	0.7
IRT-4M(20%-300)	7.758	-1.2	2.396	10.8	16.336	6.6	6.147	-5.0	2.701	12.6	16.107	7.5

Average neutron flux density [1/cm2/s/10MW x1.e-13] in cell with dimensions (7.15 x 7.15 x 60.0)cm												
Check position :	**D6 (fuel assembly)**											
Material of CC :	**Water in other positions**						**Alu in other positions**					
Fuel type	**thermal**	dev[%]	**fast**	dev[%]	**total**	dev[%]	**thermal**	dev[%]	**fast**	dev[%]	**total**	dev[%]
IRT-2M(36%-230)	9.432	-	6.610	-	28.917	-	9.538	-	6.691	-	29.268	-
IRT-3M(20%-400)	5.563	-41.0	7.038	6.5	26.414	-8.7	5.395	-43.4	7.170	7.2	26.912	-8.0
IRT-4M(20%-300)	6.297	-33.2	7.301	10.5	28.473	-1.5	6.417	-32.7	7.447	11.3	29.044	-0.8

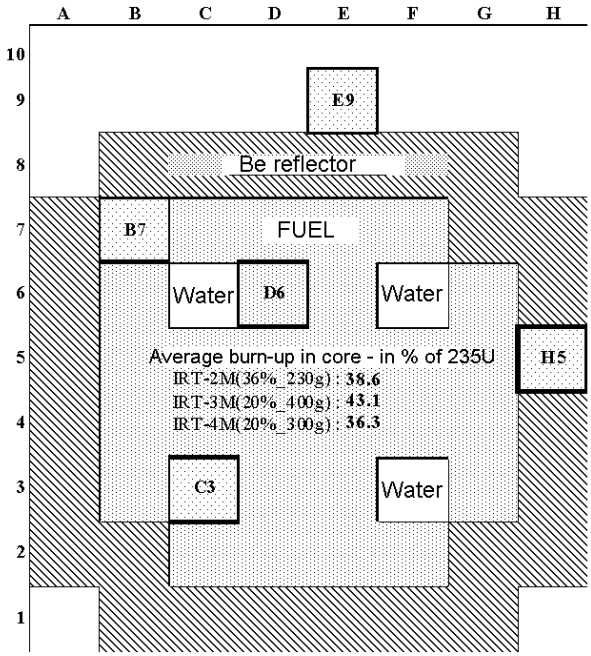

Fig. 4. Layout of Check Cells in KS28b configuration

The results of the calculations for irradiation position in the core (positions B7, C3, D6) as well as outside (H5 and E6) are presented in the Table 5. There can be detected significant decrease of the thermal neutron flux in the in-core positions with change to LEU, which is on the contrary to the users demands. The change of the thermal neutron flux in the channels out of the core is not too big. The fast flux and ratio 'fast to thermal' is generally higher for the LEU in all the studied channels that is one of few positive facts in 'cause of conversion' because most present material irradiation projects need just this.

5. Conclusion

The research reactor LVR-15 represents a significant tool for the basic and applied neutron physics, production of radioisotopes for medical and industrial purposes as well as research of structural and prospective nuclear installation.

In case of necessity of conversion to LEU, the conclusion we can make is that the most suitable FA from point of view of current offer would be IRT-3M (UMo) or IRT-4M (UO_2) with 'optimalised' content of ^{235}U. Resulting from the presented study, the optimal content of ^{235}U lies between 325 and 350g in the standard (complete) FA.

6. References

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