RESULTS OF TRANSIENT /ACCIDENT ANALYSIS FOR THE HEU, FIRST MIXED HEU-LEU AND FOR THE FIRST FULL LEU CORES OF THE WWR-SM REACTOR AT INP AS RUZ

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ABSTRACT

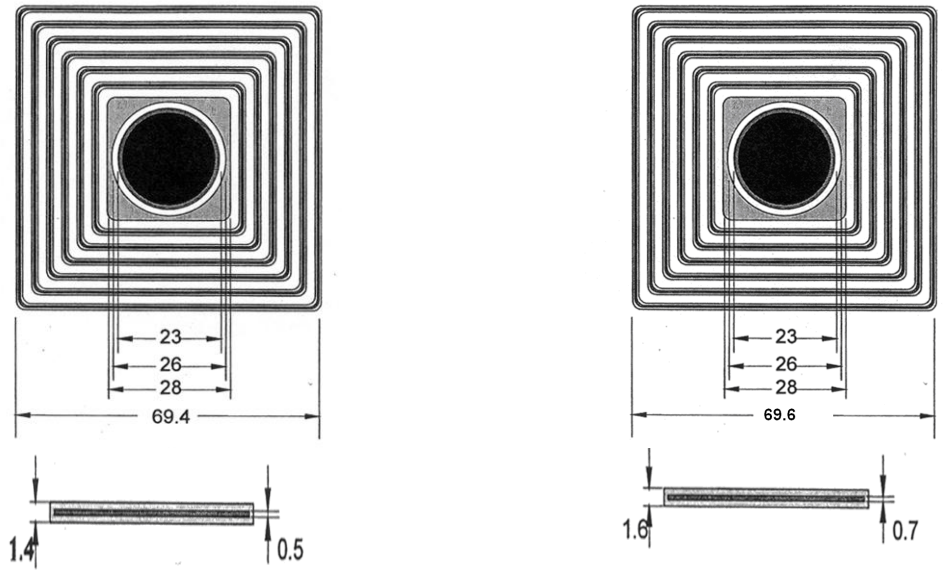
The WWR-SM reactor in Uzbekistan is preparing for the conversion from HEU (36%) fuel to LEU (19.8%) fuel. During this conversion, the HEU fuel assemblies (IRT-3M FA) being discharged at the end of each cycle will be replaced by LEU fuel assemblies (IRT-4M FA); this gradual conversion requires 9 cycles. The safety analysis report for this conversion process has been prepared. This paper presents selected results for postulated transient/accidents during this conversion process; results for transient analysis for the HEU core, the 1st mixed (HEU-LEU) core, and for the first full LEU core are presented for the following initiators: control rod motion (2 cases), loss of power, and FA blockage. These results show that safety is maintained for all transients analyzed and that the behavior of all the analyzed cores is essentially the same.

Introduction

The WWR-SM research reactor in Uzbekistan currently uses HEU (36% enrichment) IRT-3M fuel assemblies (6-tubes, UO_2-Al, 2.5 g U/cm^3, 314 g ^{235}U/FA) fabricated by the Novosibirsk Chemical Concentrates Plant in Russia. The Institute of Nuclear Physics (INP-UZ) has decided to convert the WWR-SM reactor using presently available IRT-4M oxide LEU fuel, until the IRT-3M UMo LEU fuel is available. To do that without loss of performance for the present experimental program the size of the core will increase to 20 fuel assemblies (FA) and the power of the reactor will increase to 11 MW (instead of 18 FA and 10 MW for the present core). This conversion process will occur over a period of time with partial replacement of IRT-3M HEU fuel with IRT-4M LEU fuel until a fully loaded LEU core is obtained. The neutronics parameters and steady-state thermal-hydraulics analysis of the conversion process are described in another paper [1] at this meeting. This paper presents the results for the transient/accident for the HEU, for the 1st mixed and the LEU cores.

Fuel Assemblies descriptions

Figure 1, shows a horizontal cross section of both the IRT-3M (HEU) [2] and the IRT-4M (LEU) [3] Fuel Assemblies (FA) used in the reactor, and Table 1, presents the parameters of those FA. The two FA are very similar and the major differences are the mass of ^{235}U/FA and the thickness of the fuel meat.



HEU (36 %) IRT-3M LEU (19.8 %) IRT-4M

Figure 1, Cross Section of the 6-tube IRT-3M and IRT-4M Fuel Assembly.

Parameter	6-Tube IRT-3M	6-Tube IRT-4M
Dispersant	UO_2-Al	UO_2-Al
Wt. % ^{235}U	36.0	19.7
$(gU/cm^3)_{meat}$	2.5	2.8
VF^D, %	28.5	31.7
^{235}U/FA, g	309	266
H_{meat}, cm	60	60
T_{meat}, mm	0.50	0.70
T_{clad}, mm	0.45	0.45
$T_{coolant}$, mm	2.05	1.85
Vol_{meat}, cm^3	342	481
Max. Clad Temp., °C	100	98

Table 1, Fuel Assembly Parameters.

Core

Two types of FA are used: IRT-3M six-tube and IRT-4M six-tube. These IRT-3M/IRT-4M are utilizing tubular coaxial square section FEs with wall thickness 1.4 in IRT-3M and 1.6 mm in IRT-4M. Cladding material is alloy *SAV-1*, thickness 0.45 mm in both FA. Channel with rod of control and protection system (RCPS) or experimental channel with external diameter 28 mm and inner diameter 24 mm is being installed in six-tube FA. Standard core actual loadings are as follows:

- HEU core: 18 six-tube IRT-3M type FA with 41 material-science channels (Figure 2,),
- 1st mixed core: 16 six-tube IRT-3M type FA and 4 six-tube IRT-4M type FA with 39 material-science channels (Figure 3,).

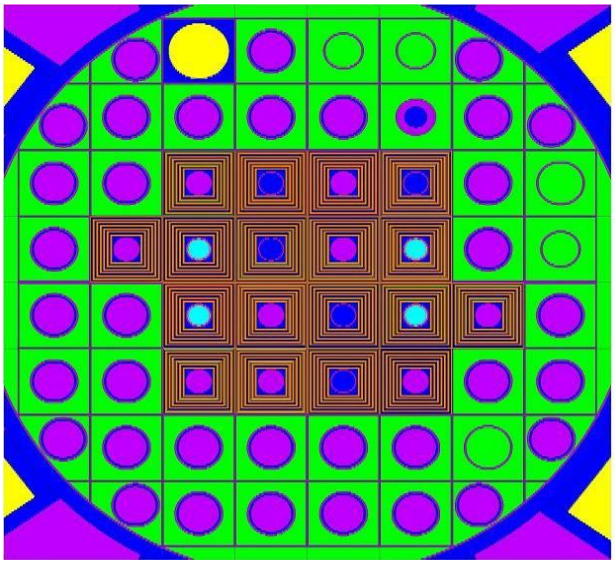


Figure 2, HEU core 18 FA of IRT-3M type with 41 vertical experimental channels.

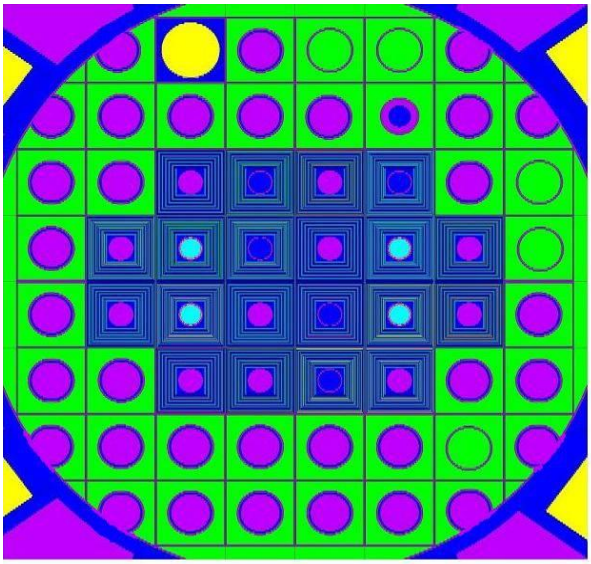


Figure 3, 1st mixed core: 16/4 six-tube IRT-3M/IRT-4M type FA with 39 material-science channels

Methods and codes used

The WIMS-ANL [4] code was used to generate the 2-group cross sections required by the burnup code IRT-2D [5]. The IRT-2D code was then used to perform cycle-by-cycle burnup analysis. The fuel compositions generated in the burnup analysis were used in a detailed model to be used in the MCNP4C [6] code. This MCNP model was used to perform all the steady state neutronics analysis which provided the detailed power densities and kinetics parameters required for the steady state thermal hydraulics analysis and the transient safety analysis.

The transient safety analysis calculations for which results are presented below were performed using the PARET code [7]. A two-channel model was utilized in PARET. One channel represents the hottest fuel plate and its associated coolant flow; the other channel represents an average fuel plate and its associated coolant flow. Since PARET allows only a single fuel type, both in terms of geometry and material, the transition cores have been analyzed as two portions: 3M and 4M, after which the powers are added. The use of PARET in this manner has been documented [8].

Transients start at nominal maximum operating power, which is 10 MW for the HEU core and 11 MW for all other cores; trip setpoint for overpower is 12 MW in all cores. The delay time between crossing any trip set point and the start of control rod motion into the core is 0.1 s. Scram during reactor operation consists of the drop of the three safety rods (SR) from fully withdrawn position and drop of six shim rods (CR) from critical position; drop of the six CR rods is (conservatively) ignored in the calculations presented below. The SR insert full length of 0.6 m in 0.5 s. The CR fully insert with constant velocity of 35 mm/s. The 3 primary pumps continue to run after scram for reactivity-induced transients. The pumps coast down to essentially zero flow over 4 s following loss of power. A small auxiliary pump operates continuously on battery power at 100 m^3/h; this pump is under procurement by INP and will be set up in the reactor primary circuit system.

Kinetics Parameters and Reactivity Coefficients

The kinetics parameters and reactivity coefficients are important for the determination of the core response after an incident/transient. Table 2, presents the results [1] for full HEU, 1st mixed and full LEU cores. The results show, that the reactivity coefficients and the kinetics parameters for the first mixed and full LEU core are similar to those for the reference IRT-3M HEU core.

	HEU core 18IRT-3M	**1st mixed core 16 IRT-3M, 4IRT-4M**	**Full LEU core 20 IRT-4M**
Coolant Temperature Reactivity Coefficient (%/ºK)			
293 to 400 K	-9.2E-3	-9.5E-3	-9.1E-03
400 to 600 K	-7.1E-3	-7.7E-3	-7.3E-03
293 to 600 K	-7.9E-3	-8.3E-3	-7.9E-03
Coolant Void Reactivity Coefficient (%/% void)			
0 to 5%	-2.8E-1	-2.6E-1	-2.7E-01
5 to 10%	-3.0E-1	-2.9E-1	-2.9E-01
0 to 10%	-2.9E-1	-2.8E-1	-2.8E-01
Doppler Reactivity Coefficient (%/ºK)			
293 to 400 K	-2.2E-3	-2.1E-3	-2.3E-03
400 to 600 K	-1.5E-3	-1.7E-3	-2.2E-03
293 to 600 K	-1.7E-3	-1.8E-3	-2.2E-03
Effective Delayed Neutron Fraction, β_{eff} [6]			
	7.6E-3	7.6E-3	7.6E-3
Prompt Neutron Generation Time, μs			
	52	53	56

Table 2, Kinetics parameters and reactivity coefficients (uncertainty less than 3 % for all parameters)

Transients/Accidents Analyzed

Four different accident situations considered in the Safety Analysis Report are presented in this paper:

- Automatic Rod Withdrawal
- Control Rod Ejection
- Loss of Power
- Blockage of FA

Each case is presented in a separate section below.

Automatic Rod Withdrawal Transient

For this transient, signal (due to insulation failure or open circuit) will be generated indicating decreasing power and auto rod AR will be (automatically) withdrawn to bring power to the operating level. Power will increase and scram will occur on high power. Transient analysis has been performed assuming AR motion starts from its normal mid-core-height location, withdrawal occurs at constant rod speed of 10 mm/s, and reactivity addition is based on rod worth versus position curve. The available AR worth to insert is 0.376 $ for full IRT-3M core and is 0.434 $ for the 1st mixed core, 0.439 $ for LEU core, and it would take 35 s for total withdrawal.

Core analysis

Results from PARET are shown in Figures 4 and 5. The initial power is10 MW for HEU core, and 11 MW for the 1st mixed and LEU cores. Scram occurs due to power exceeding 12 MW after 8.33 s for the HEU core, after 3.26 s for the 1st mixed core, and after 3.34 s for the LEU core. The maximum power and reactivity are: 12.03 MW and 0.0763 $

after 8.44 s for HEU core; 12.03 MW and 0.049 $ after 3.36 s for 1st mixed core; and 12.03 MW and 0.049 $ after 3.45 s for LEU core. The maximum clad surface temperature is: 107.9 °C at 8.44 s for the IRT-3M (HEU) core; 107.98 °C at 3.36 s for IRT-3M FA and 92.17 °C at 3.37 s for IRT-4M FA in the 1st mixed core; and 103.1 °C at 3.46 s for the IRT-4M (LEU) core. These clad temperatures are much lower than clad damage temperatures and are even lower than the temperature at which bubble formation in the coolant occurs (124 °C, see reference [1]).

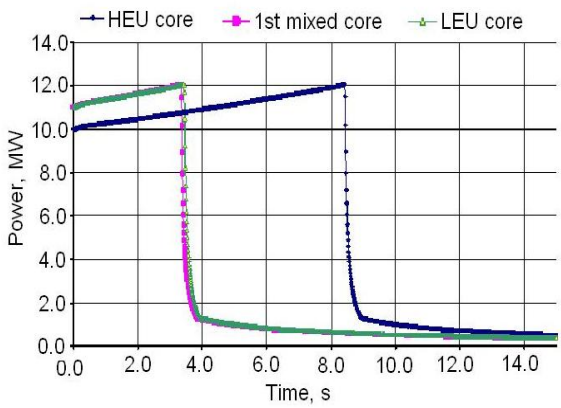


Figure 4 **Auto Rod Withdrawal - Power**

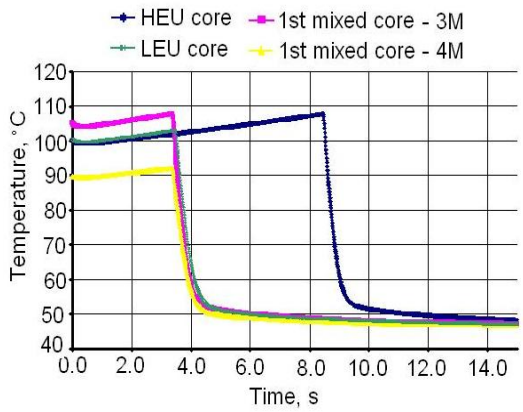


Figure 5 **Auto Rod Withdrawal - Peak Clad Surface Temperature**

Control Rod Ejection Transient

Each CR is connected by a cable to electric drive located on a special platform. The rod is being cooled by downward water flow through the gap between the rod and channel wall. Therefore rod ejection from a core in upward direction is unlikely. However, it was assumed that the rod can be ejected and the resulting transient was analyzed. Maximum reactivity worth from control rods is for the pair of rods CR-1: 5.526 $ for the HEU core, 5.263 $ for the 1st mixed core and 5.658 $ for the LEU core. This transient assumes the ejection of one of the two rods, and half of the worth of the pair (2.763 $ for the HEU core, 2.632 $ for the 1st mixed core and 2.829 $ for the LEU core) is used for the reactivity worth of the ejected rod. The rod ejection is assumed to occur over the same time as required for rod insertion during scram, which is 0.5 s.

The results from PARET are shown in Figures 6 and 7. The initial power is 10 MW for HEU core, and 11 MW for the 1st mixed and LEU cores. Scram occurs due to power exceeding 12 MW after 0.04 s for the HEU core, after 0.02 s for the 1st mixed core, and after 0.03 s for the LEU core. The maximum power and reactivity are 28.20 MW and 0.6811 $ after 0.14 s for HEU core, 25.1 MW after 0.13 s and 0.599 $ after 0.12 s for 1st mixed core, and 26.91 MW after 0.13 s and 0.633 $ after 0.12 s for LEU core. The maximum clad surface temperature is 125.15 °C at 0.17 s for HEU core, 124.1 °C at 0.16 s for IRT-3M FA and 103 °C at 0.17 s for IRT-4M FA in the 1st mixed core, and 118.7 °C at 0.17 s for LEU core. These clad temperatures are much lower than clad damage temperatures; however, for IRT-3M FA, they are near the temperature at which bubble formation in the coolant occurs (124 °C, see reference [1]).

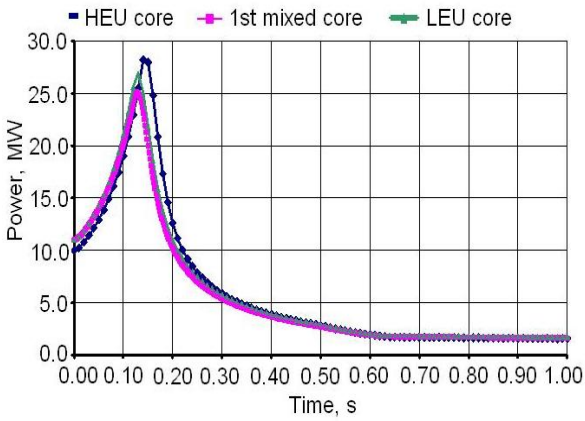


Figure 6 Transient: Control Rod Ejection - Power

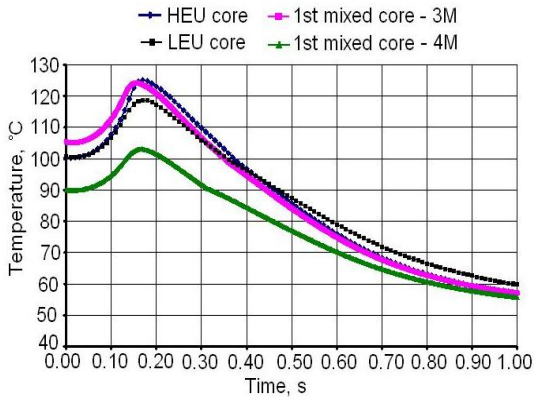


Figure 7 Transient: Control Rod Ejection - Peak Clad Surface Temperature

Loss of Power Transient

Loss of electrical power causes loss of all main primary (3 normally operating) and all secondary (4) pumps. The emergency primary pump powered by the "reliable power supply" (battery) continues operating (100 m^3/h). The pump coast down occurs over 4 s, as shown in Figure 8. Loss of power causes reactor trip – conservatively the trip is assumed to occur on low flow (80% of nominal) rather than immediately upon loss of power.

This is a loss of flow transient with scram on loss of power right at the beginning of the transient. In case of power supply loss for primary loop pumps the accident protection could also be activated by the following independent channels:

- decreasing pressure in primary loop supply pipe,
- decreasing primary loop flow at 80% of nominal level,
- decreasing core pressure drop.

When power loss occurs only one small pump (~7.5 kW power and water flow is 100 m^3/h) is left in operation with reliable power supply source – the battery. This pump is under procurement by INP and will be set up in the reactor 1st contour (primary) system.

The function of flow versus time is presented on Figure 8. This function is obtained on the basis of experiment with the addition of the small pump discussed above.

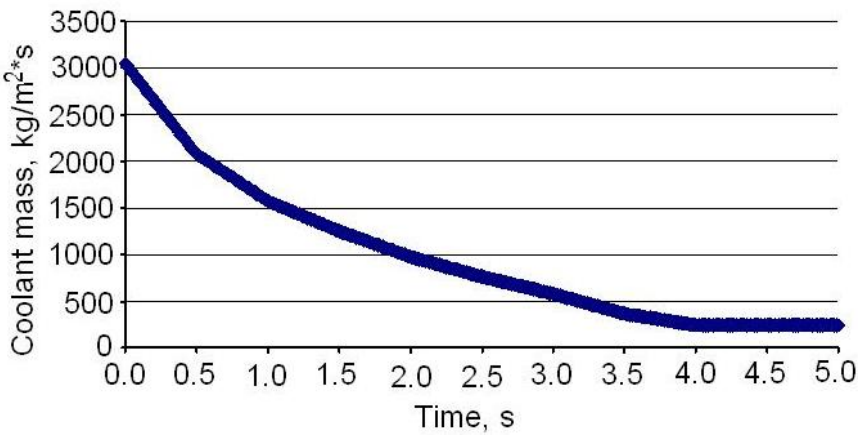


Figure 8. Loss of Power: Pumps Coastdown

Results from PARET are shown in Figures 9 and 10. As coolant flow decreases temperatures increase; this heating results in a negative reactivity feedback and the power decreases. Scram signal occurs due to coolant flow less than 80 % at 0.32 s for the HEU and 1st mixed cores and at 0.41 s for the LEU core, because (conservatively) no credit was given for the scram signal at the time of the loss of power, i.e. at the beginning of the transient. The power decrease from scram terminates the initial temperature rise, where maximum clad temperature was 105.98 °C for the HEU core, 111.45 °C for IRT-3M FA and 94.37 °C for IRT-4M FA in the 1st mixed core, and 105.66 °C at 0.43 s for the LEU core. These clad temperatures are much lower than clad damage temperatures and are even lower than the temperature at which bubble formation in the coolant occurs (124 °C, [1]). Following scram, power settles into slower decrease than the decrease in flow; therefore temperatures begin a second increase. Figure 10 shows second peak beginning from minimum temperature of 59.5 °C at about 2.0 s for the HEU core, 59.2 °C for IRT-3M FA and 56.63 °C for IRT-4M FA at about 1.97 s in the 1st mixed core and 59.42 °C at about 2.31 s for the LEU core. The coolant flow reaches battery-provided minimum at 4 s for all cores. After this time the temperatures reach second (and final) maximum of 68.99 °C at 7.3 s for the HEU core, 71.13 °C for IRT-3M FA and 65.25 °C for IRT-4M FA at 7.17 s in the 1st mixed core, 69.96 °C at 7.53 s for the LEU core, and then the temperature decreases.

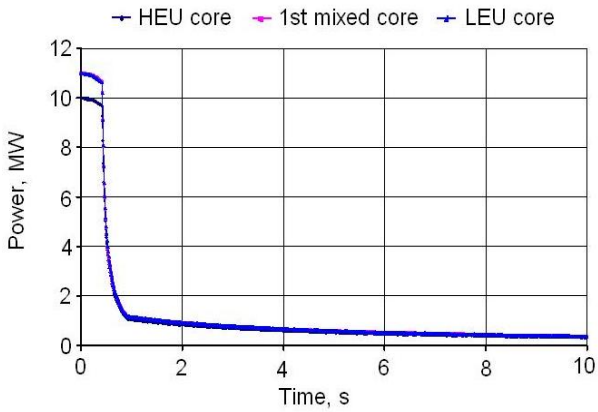


Figure 9. Transient: Loss of Power - Power

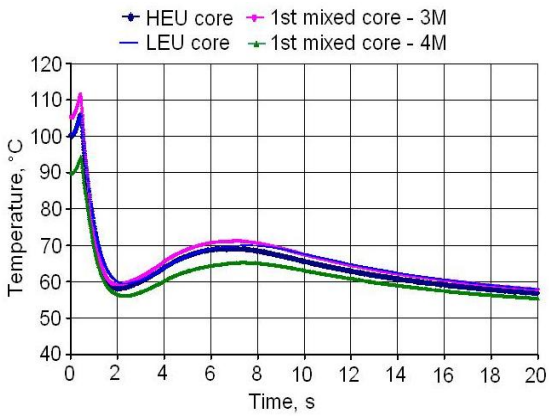


Figure 10. Transient: Loss of Power - Peak Clad Surface Temperature

Blockage of Fuel Assembly

For WWR-SM reactor accident situation one of the most serious postulated transients is the situation with coolant blockage through one FA as a result of foreign subject getting into coolant. When coolant circulation through FA is stopped, it is assumed that the FA will melt and the sharp increase in radioactivity of primary loop water will be taking place as a result of cladding melting and exposure of fuel composition. As level of gamma radiation on primary loop pipe reaches 3,600 μRad/s, the accident protection system will be actuated and reactor will be shutdown. If for any reason the accident protection system is not actuated, the reactor will be shutdown by operator following with the warning signals:

- two times increasing radioactivity level of the air discharging into exhaust pipe;
- two times increasing beta activity in the air discharging from above-reactor space.

Special ventilation system B-2 is being switched to a standby (emergency) group of gas purification filters.

Work is being performed to determine the FA with melted claddings and then FAs are reloaded in the storage cell for tightness-loss FAs. Primary loop water is being treated by going through ion exchange resin.

This transient is postulated to be caused by blockage of flow to one fuel assembly. It is further assumed that one fuel assembly will melt for this transient, and the consequences will be analyzed. For the HEU core it is assumed that the fuel assembly with the highest average burnup (60 %) melts. For the first mixed core (IRT-3M/IRT-4M FA), the maximum average burnup of the LEU FA is at most about 6.6 %, and for the HEU FA the maximum average burnup is about 60 %. Therefore, it is assumed that for the 1st mixed core the HEU FA with maximum burnup melts; this is the FA with higher inventory of fission products. The consequences of this assumed melting of one IRT-3M FA are the same for both the HEU and the first mixed core. For the LEU core it is assumed that the highest average burnup FA (40%) melts., The resulting doses are discussed below.

In fuel elements of WWR-SM reactor dispersion nuclear fuel is used: grains of dioxide of uranium are dispersed in an aluminum matrix, with the size of grains 100 μm, and 92 % of the generated fission products in a fuel composition will be in the grains of UO_2 and 8 % in the aluminum matrix [9].

The largest part of decay products of uranium is contained in grains. The temperature of melting of these grains of UO_2~2700 °C cannot be achieved only at the expense of residual energy-release (it is supposed, that the reactor is scrammed at the moment of the beginning of failure).

The residual energy is capable to melt only the aluminum matrix (temperature of melting ~600 °C), which will results in liberation of fission products contained in it. According to the work in reference [9] the aluminum matrix contains ~10 % of full activity of fission products, which well corresponds to the performed estimation of a share of the fragments, which have left a grain of fission at the expense of the kinetic energy received by them at fission of uranium (the estimation assumed the diameter of a grain of 100 μm, length of free run of a fragment in uranium – 10 μm).

After the accident, the ventilation system is stopped and the fission products can only leave the reactor building through leakage. The reactor building contains no openings and there is no leak-rate measurement. Therefore, two leakage rates were used in this study:1 %/day and 5 %/day; the results presented below are for the 1%/day leakage rate. It is also assumed that the release occurs at ground level (conservative assumption) and the diffusion factors were obtained from reference [10].

Using the assumptions described above, the doses at the boundary of the reactor (1000 m) and at 200 m were calculated, and the results for the thyroid dose and whole body doses are presented in Table 3, and Table 4,

Calculation results show that during the accident with single FA melting, the dose exposure at the boundary of the reactor (1000 m) due to external radiation from radioactive cloud, dose exposure of thyroid gland to radiation, and dose exposure to external radiation from polluted earth surface are significantly lower than the accident doses allowed: 25 Rem at distance 1 km for the whole body dose, and the limit of 300 Rem for thyroid dose [11].

For mitigation of consequences of failure the special commission which will work on the basis of instructions on mitigation of failures will be activated.

Transient: Blockage of FA – HEU Core Doses

Thyroid Dose			
Distance	2 Hour Dose	30 Day Dose	1 Year Dose
[m]	Exposure [rem]	Exposure [rem]	Exposure [rem]
200	5.34E+00	3.28E+01	3.62E+01
1000	3.45E-01	2.05E+00	2.24E+00
Whole Body (Internal)			
200	7.61E-02	7.32E-01	7.45E-01
1000	4.91E-03	4.43E-02	4.58E-02
Whole Body (External)			
200	3.88E-02	1.40E-01	1.70E-01
1000	2.50E-03	8.80E-03	2.24E-02
Whole Body (Total)			
200	1.15E-01	8.72E-01	9.15E-01
1000	7.41E-03	5.31E-02	6.82E-02

Table 3, Doses for one IRT-3M FA (60 % burn up) melting and 1 %/day leakage

Transient: Blockage of FA –LEU Core Doses

Thyroid Dose			
Distance	2 Hour Dose	30 Day Dose	1 Year Dose
[m]	Exposure [rem]	Exposure [rem]	Exposure [rem]
200	5.30E+00	3.26E+01	3.60E+01
1000	3.42E-01	2.03E+00	2.22E+00
Whole Body (Internal)			
200	7.43E-02	7.14E-01	7.27E-01
1000	4.79E-03	4.32E-02	4.47E-02
Whole Body (External)			
200	3.81E-02	1.36E-01	1.65E-01
1000	2.46E-03	8.54E-03	2.16E-02
Whole Body (Total)			
200	1.12E-01	8.50E-01	8.92E-01
1000	7.25E-03	5.18E-02	6.63E-02

Table 4, Doses for one IRT-4M FA (40 % burn up) melting and 1 %/day leakage

Conclusions

Results of calculations for the HEU core (18 FA at 10 MW), for the first mixed HEU-LEU core (20 FA at 11 MW) and LEU core (20 FA at 11 MW) show that operation of the reactor WWR-SM presents no safety problem. For all configurations the consequences of transients/accidents are essentially the same.

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