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# Analysis of Beyond Design Basis Accident for Conversion of IRT MEPhI Research Reactor to LEU Fuel

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**ABSTRACT**

This paper presents the results of the analysis of Beyond Design Basis Accident − fuel channel blockage (drop of foreign object into the inlet of the core). It is assumed that one fuel assembly will melt for this accident and radiological consequences are analyzed. The core with tube-type low enriched uranium (LEU, 19.7 w/o, U-9%Mo) fuel and oxide high enriched uranium (HEU, 90 w/o) fuel is investigated. This work is the part of safety analysis for conversion of the IRT MEPhI research reactor to LEU fuel.

## 1. Introduction

The present work is connected with the conversion analysis of the IRT MEPhI research reactor at the National Research Nuclear University MEPhI. IRT MEPhI is 2.5 MW pool type research reactor. For feasibility studies of the IRT MEPhI reactor conversion tube-type fuel assembly IRT-3M with U9%Mo-Al fuel (19.7 w/o) was chosen as a LEU fuel [1].The analysis sufficient to determine that conversion from HEU to LEU fuel is technically feasible was performed. Safety analysis for the conversion was completed in July 2013.

The paper presents the results of the analysis of Beyond Design Basis Accident − fuel channel blockage (drop of foreign object into the inlet of the core). It is assumed that one fuel assembly will melt for this accident and radiological consequences are analyzed. Simple manual calculations based conservative fission product fractional release data and nomograms for site specific atmospheric dispersion data were performed.

The studies are being carried out in cooperation with Argonne National Laboratory (ANL) and at its funding support under the contract No. 0J-30402 between ANL and MEPhI.

## 2. Fission product inventory calculation

It is assumed that molten fuel assembly (FA) is 8-tube FA with average burnup of 50%. It is assumed that FA worked continuously at the power of 250 kW up to the burnup of 50%. The value of 250 kW corresponds approximately to the power of the hottest FA in the core of 16 FA at the reactor power of 2.5 MW. This assumption is conservative because only fresh FA have the peak power (the power of FA of 50% burnup is ~1.5 times less than the peak power). Reactor power, FA number in the core and maximum burnup are the same for HEU and LEU cores and the only difference in fission products inventory calculation between HEU and LEU cores was in the time of operation to reach average burnup of 50%. Fission product inventory calculation was performed using MCU-PTR code [2].

In isotope list of MCU-PTR code some short-lived isotopes with small absorption cross section are absent. Some of them however are important for source term estimation (for example Kr-88). For these radionuclides the activity was calculated by the formula:

$$Q_i(T) = p \cdot Y_i \cdot W \cdot (1 - e^{-\lambda i T}), \qquad (1)$$

in which $Q_i(T)$ –the activity of isotope $i$, Ci;

$p$=3,1·10$^{10}$/3,7·10$^{10}$ (fissions/(s·W))/(decays/(s·Ci),

$Y_i$ – cumulative yield of isotope $i$;

$T$ – the time interval during which the fuel has been at power W, s;

$W$ – FA power, W,

$\lambda_i$ – radiological decay constant of isotope $i$, s$^{-1}$.

Radionuclide activities for the burnup of 50% were calculated. The total activity is $3.45 \cdot 10^5$ Ci and $3.37 \cdot 10^5$ Ci for HEU and LEU fuel respectively. For short-lived isotopes the difference is caused by the difference in yields for $^{235}U$ and $^{239}Pu$ fissions. The activity of the long-lived isotopes is higher for LEU fuel because at the burnup of 50% the mass of burned $^{235}U$ is larger for LEU fuel.

It should be noticed that LEU will have greater plutonium bildup, however for the considered accident the release occurs under the water and plutonium contribution to the source term is negligible.

## 3. Fission product releases for fuel channel blockage accident

Considered IRT-3M HEU and LEU fuels belong to U–Al fuel type. For U–Al alloy, dispersed $UAl_x$, $U_3O_8$ and $U_xSi_y$ fuels experimental fission product releases are available [3,4]. For dispersed $UO_2$/Al and LEU UMo fuels there are no available experimental data on fission product releases. So the estimation of fission product releases for IRT-3M HEU and LEU fuels is done on the basis of the data for the other dispersed U–Al fuels.

Experimental data for all types of aluminum-based plate-type fuel show a consistent pattern, namely that major release of fission gases begins when the aluminum cladding blisters or begins to melt [4]. At 700°C 100% of fission gas is released for all fuel types independently of the differences in the properties of fuel particles used in fuel composition. This release is assumed to be related to the reaction of the fuel composition with the aluminum matrix and to the $UA1_4$-A1 eutectic. $UO_2$ and UMo fuel also undergo chemical reaction between fuel particle and aluminum at the temperatures >600°C. The rate of chemical reaction may be different for different fuel compositions but once aluminum of matrix melts the result is the same for all fuel types. So, for the dispersed $UO_2$/Al and LEU UMo/Al fuels fission product releases can be estimated as similar to other known dispersed fuels.

As a result of the analysis of available experimental data presented in IAEA reports and other corresponding references for the estimation of source term for IRT MEPhI reactor with HEU and

LEU fuel the release fractions of fission products from fuel to pool water and from pool water to air were selected (Table 1). These release fractions were used in further calculations.

**Table 1. Release fractions of fission products for IRT-3M HEU and LEU fuel**

| Radionuclide | Release fraction from fuel $f_1$ | Release fraction from pool water $f_2$ | Release fraction from fuel to reactor room $f_1 \cdot f_2$ |
|---|---|---|---|
| Noble gases | 1 | 0.02 | 0.02 |
| I, Cs, Te | 0.27 | $1 \cdot 10^{-4}$ | $2.7 \cdot 10^{-5}$ |
| Ba, Sr, Ru | 0.03 | $<1 \cdot 10^{-6}$ | $3 \cdot 10^{-8}$ |
| other | 0.001 | $<1 \cdot 10^{-6}$ | $1 \cdot 10^{-9}$ |

## 4. Radiological consequences analysis

### 4.1. Regulatory radiation safety standards

The dose limits to the staff and to member of the public are determined in radiation safety standards RSS-99/2009 [5]. During normal operation of radiation-dangerous object the annual maximum permissible effective doses are: 20 mSv for operating staff and 1 mSv for critical group of population. After the accident radiation safety standards allow planned higher exposure of operating staff in some cases. Planned higher exposure of operating staff with effective dose of 100 mSv and with double maximum equivalent doses is allowed by institutions of local governing. Planned higher exposure of operating staff with effective dose of 200 mSv and with fourfold maximum equivalent dose is allowed by federal executive authorities.

### 4.2. Scenarios of radionuclide release into environment

Radioactivity release into atmosphere depends not only on the release fractions of fission products from fuel to pool water and from pool water to air of the reactor room, but on the reactor design features such as ventilation system. The ventilation system of IRT MEPhI reactor consists of common ventilation and special ventilation. Common ventilation draws air from reactor room and other technical rooms through the common ventilating shaft with the exhaust stack height of 21.5 m. Special ventilation is intended for the air intake from the places of radioactivity release (area above the pool water and under the pool upper flooring and some other rooms). The air of special ventilation system cleaned by HEPA filters is drawn through the exhaust stack at the height of 40 m. There are no containment and confinement at IRT MEPhI reactor.

Operating staff actions during considered accident is determined by emergency plan. The emergency actions are as follows:

- primary and secondary coolant circuits shutdown;
- evacuation of the staff from the reactor room;
- shutdown of common ventilation;
- accessing of coal filters in special ventilation system;
- radiation monitoring;
- warning of external organizations.

That is, it is not assumed that special ventilation system will be turned off. Therefore in the first accident scenario special ventilation continues to operate (common ventilation off). For this scenario the doses to members of the public due to short-term radioactivity release through the exhaust stack at the height of 40 m are estimated. It is difficult to make a realistic estimation of the doses to staff for this scenario. Widespread variant of the consideration of the accident is the scenario with all ventilation off. Therefore the second scenario was considered with common and special ventilation system off. For this scenario the doses to the operating staff and members of the public were calculated. The doses to the operating staff in this case will be larger than for the first accident scenario and they are considered as a conservative estimation.
So, to calculate specific activity of the radionuclide in the air released from the reactor building the following assumptions were made:

**Scenario 1**. Special ventilation system continues to operate (common ventilation off). The ventilation rate is 4000 $m^3$/h. Exhaust stack height is 40 m. Special ventilation is assumed to serve an area above the pool water only. The volume of this area is ~9 $m^3$. Instantaneous release of radionuclide from pool water to this area is assumed. The release rate through the stack at the moment of the accident is estimated as:

$$A_R^i = K_{up}^i \; w_{vent}, \qquad (2)$$

where $A_R^i$ – release rate of isotope $i$, Bq/s; $K_{up}^i$ – specific activity of the radionuclide in the air of area above the pool water, Bq/$m^3$, $w_{vent}$ – ventilation rate, $m^3$/s (1.1 $m^3$/s).

**Scenario 2.** Special ventilation system is turned off. Instantaneous release of radionuclide from pool water to the air of the reactor room is assumed. The reactor has no containment and the reactor room has the openings. The main way for the air leakage is common ventilating shaft. The dimension of the shaft is ~0.7 m x 0.7 m. The height of the shaft is ~20 m. There will be a different air draft depending on the wind speed. The volume of the reactor room is 6000 $m^3$. Building leakage rate is assumed to be 20% per day (50 $m^3$/h, air velocity in the ventilating shaft is 0.028 m/s).The release height is assumed to be 20 m (reactor building height). The release rate at the moment of the accident is:

$$A_R^i = 0.2\, K_{hall}^i \;\; V_{hall} /3600\cdot 24 \;, \qquad (3)$$

where $A_R^i$ – release rate of isotope $i$, Bq/s; $K_{hall}^i$ – specific activity of the radionuclide in the air of the reactor room, Bq/$m^3$, $V_{hall}$ – the reactor room volume.

Specific activities of the radionuclides in the air are calculated as follows:

$$K_{hall}^i\,(t=0) = f_1\, f_2\, Q_i \,/V_{hall} \qquad (4)$$

$$K_{up}^i\,(t=0) = f_1\, f_2\, Q_i \,/V_{up} \;, \qquad (5)$$

where $Q_i$ – specific activity of the radionuclide in the melted part of the core, Bq; $f_1$, $f_2$ – the release fractions of fission products from fuel to pool water and from pool water to air; $V_{hall}$, $V_{up}$ – the reactor room volume and the volume of the area above the pool water.

### 4.3. Calculation of the doses to operating staff (special ventilation system off)

Radiological consequences inside the reactor building might arise from:
- External irradiation from airborne radioactive material (photon dose to whole body, beta radiation to skin);
- Internal exposure from airborne radioactive material (inhalation).

For the estimation of the doses to operating staff the exposure time is assumed to be 10 minutes.

***External irradiation from airborne radioactive material (gamma immersion)***

The external radiation due to immersion in a radioactive cloud of limited volume (the reactor room) is considered. The cloud volume is taken as semi-infinite. The effective γ-dose rate in the reactor room for any isotope is calculated by:

$$\dot{E}_i = \delta^i K^i_{hall}(t) \tag{6}$$

where $K_{hall}{}^i$ – activity of the radionuclide $i$ in the air of the reactor room, Bq/m$^3$, $\delta^i$ – dose conversion factor for γ-immersion, (Sv/s)/(Bq/m$^3$) [6]. $K_{hall}$ *(t=0)* is given by (4). Activity concentration $K_{hall}$ *(t)* for noble gases and I, Cs, Te isotopes is calculated taking into account radiological decay constants, leak rate parameter and deposition rate (plate-out constant for I, Cs, Te isotopes).

Total external effective γ-dose to operating staff in the reactor room over 10 minutes after the accident for HEU and LEU fuel is presented in Table 2.

**Table 2. External effective γ-dose to operating staff over 10 minutes**

| Group of radionuclides | HEU | | LEU | |
|---|---|---|---|---|
| | Dose rate, Sv/s | Dose over 10 min, mSv | Dose rate, Sv/s | Dose over 10 min, mSv |
| Noble gases | $1{,}52\cdot10^{-4}$ | 91,1 | $1{,}44\cdot10^{-4}$ | 86,5 |
| I, Cs, Te | $1{,}68\cdot10^{-6}$ | 1,01 | $1{,}67\cdot10^{-6}$ | 1,00· |
| Ba, Sr, Ru | $2{,}35\cdot10^{-10}$ | $1{,}41\cdot10^{-4}$ | $2{,}27\cdot10^{-10}$ | $1{,}36\cdot10^{-4}$ |
| other | $4{,}35\cdot10^{-11}$ | $2{,}61\cdot10^{-5}$ | $4{,}12\cdot10^{-11}$ | $2{,}47\cdot10^{-5}$ |
| **Σ** | | 92,1 | | 87,5 |

Total external effective γ-dose to operating staff in the reactor room over 10 minutes is ~90 mSv. Noble gases make the main contribution to the external effective dose.

***Internal doses arising from the inhalation of radionuclides***

Internal doses arising from the inhalation of radionuclides are a function of the time integrated activity concentration at the location of interest.

The activity of radionuclides (in Bq) entered the organism from the inhalation is calculated by:

$$Q_{inh,} = K_{hall} \cdot w_{br} \cdot T \tag{7}$$

where $w_{br}$ – breathing rate = $3.3\cdot10^{-4}$m$^3$/s (for working - day, [7]), $T$ – time of exposure.

The inhalation effective dose of any nuclide is calculated by:

$$E_{inh,} = Q_{inh,} \cdot e^{air}{}_{staff} \tag{8}$$

where $e^{airi}{}_{staff}$ – inhalation dose conversion factor (in Sv/Bq) for staff [5].

Total inhalation effective dose is 7.95 mSv for HEU and LEU fuel. The main contributors to inhalation effective dose are iodines and $^{132}Te$.

## 4.4. Calculation of the doses to the members of public

***Atmospheric dispersion***

Pasquill model based on the Gaussian plume equation [3,6] was used for atmospheric concentration calculation. Activity concentration near to ground level was calculated for Pasquill stability classes A and F. Dilution factors for effective release height of 47 m, for the surface roughness $z_0$=100 cm and for the distances from the stack of x=200 m, 500 m and 2000 m were determined using nomograms from ref. [6].

### 4.4.1 Scenario # 1 (special ventilation system on)

Activity concentration (in Bq/m$^3$) near to ground level is calculated by:

$$K_{air}(x) = A_R \cdot G(x) e^{-\lambda \cdot x / v} \tag{9}$$

where $A_R$ - release rate (in Bq/s) of the specific nuclide calculated by (2);
$G(x)$ - dilution factor, s/m$^3$; $\lambda$ —radioactive decay constant, s$^{-1}$; $v$— wind velocity m/s.

The distances from the stack of 200 m, 500 m and 2000 m are considered. Dilution factors were determined by nomograms for the wind velocity of 1.5 m/s. The dilution factors are assumed to be:

| Weather category A | Weather category F |
|---|---|
| $G(200\ m)=6\cdot10^{-5}$ s/m$^3$, | $G(200\ m)=1\cdot10^{-7}$ s/m$^3$, |
| $G(500\ m)=2\cdot10^{-5}$ s/m$^3$, | $G(500\ m)=1\cdot10^{-7}$ s/m$^3$, |
| $G(2000\ m)=1.5\cdot10^{-6}$ s/m$^3$; | $G(2000\ m)=2\cdot10^{-5}$ s/m$^3$. |

As the instantaneous release is considered (10 s) the dimension of the cloud is assumed to be ~100 m. With the wind velocity of 1.5 m/s the time of radioactive cloud passage was estimated as 1 minute. Effective dose to population over time of radioactive cloud passage (1 minute) was estimated. The dose rate at the moment of the release was calculated by the formula:

$$\dot{E} = \delta \cdot K_{air}(x) \tag{10}$$

where $K_{air}$– activity of the specific radionuclide in the air near to ground level calculated by (9), $\delta$ – dose conversion factor [6].

The effective dose over 1 minute was estimated by:

$$E = \dot{E} / \lambda_{vent} \cdot (1 - \exp(-\lambda_{vent} \cdot T)) \tag{11}$$

where $\lambda_{vent} = w_{vent}/V_{up}$ =0,12 s$^{-1}$; $w_{vent}$ – ventilation rate, m$^3$/s, $T$=1 min.

***External irradiation from airborne radioactive material***

Activity concentrations of noble gases and I, Cs, Te in the radioactive cloud, effective dose rate from immersion in a radioactive cloud at the moment of the release and effective doses over 1 minute at the distances of 200 m, 500 m and 2000 m for weather category A and F were calculated for HEU and LEU fuel.

For noble gases maximum external effective dose to the population of ~0.06 mSv over 1 minute is at the distance of 200 m from the release stack for the weather category A. Maximum external effective dose for I, Cs, Te is ~$6\cdot10^{-4}$ mSv over 1 minute.

***Internal doses arising from the inhalation of radionuclides***

The activity of radionuclides (in Bq) entered the organism from the inhalation is calculated by:

$$Q_{inh,} = K_{air}\cdot w_{br}\cdot T \quad (12)$$

where $K_{air}$ – near to ground level concentration of the specific nuclide calculated by (9), Bq/m$^3$; $w_{br}$ – breathing rate = $3.3\cdot10^{-4}$m$^3$/s (for working-day) [7], $T$ – time of exposure.
The inhalation effective dose of any nuclide is calculated by:

$$E_{inh,} = Q_{inh,}\cdot e^{air}{}_{pub} \quad (13)$$

where $e^{air}{}_{pub}$ – inhalation dose conversion factor (in Sv/Bq) of the specific nuclide for population [5].
Total inhalation effective dose is 0.08 mSv for HEU and LEU fuel. The main contributors to inhalation effective dose are iodines and $^{132}$Te.

**4.4.2 Scenario # 2 (special ventilation system on)**

Activity concentration (in Bq/m$^3$) near to ground level is calculated by (9) with release rate $A_R$ calculated by (3). Dilution factors for effective release height of 20 m, for the surface roughness $z_0$=100 cm and for the distances from the stack of x=200 m, 500 m and 2000 m were defined by nomograms from [6] taking into account the wind velocity of 1.5 m/s. In calculations the dilution factors are assumed to be:

$G(200\ m)=1\cdot10^{-3}$ s/m$^3$,
$G(500\ m)=6.7\cdot10^{-4}$ s/m$^3$,
$G(2000\ m)=1\cdot10^{-4}$s/m$^3$.

Maximum values of dilution factors were selected independently of weather category.
For the building leakage rate of 20%, 76% of activity will be released after 5 days, 99% −after 15 days and 100% − after 25 days. Effective dose to population over 30 days was estimated.
The dose rate at the start of the release was calculated by (10) with $K_{air}$ calculated by (9) with release rate $A_R$ calculated by (3).
The effective dose over 30 days was calculated by:

$$E = \dot{E}/(\lambda+\lambda_l)\cdot(1-\exp(-(\lambda+\lambda_l)\cdot T)) \quad (14)$$

where λ – radioactive decay constant, s$^{-1}$, $\lambda_l=2{,}31\cdot10^{-6}$ c$^{-1}$ – leak parameter, s$^{-1}$, $T$=30 days.

***External irradiation from airborne radioactive material***

Maximum external effective dose rate of ~$7.5\cdot10^{-3}$ mSv/h is at the distance of 200 m from the release stack. External effective dose to the population for noble gases is less than 0.05 mSv over 30 days. The contribution of I, Cs, Te to the external effective dose to population is less than $6\cdot10^{-4}$ mSv over 30 days. The contribution of the other radionuclides to the external effective dose to population is less than $6\cdot10^{-8}$ mSv over 30 days.

***Internal doses arising from the inhalation of radionuclides***

The activity of radionuclides (in Bq) entered the organism from the inhalation over 30 days is calculated by:

$$Q_{inh} = w_{br} \cdot K_{air} / (\lambda + \lambda_l) \cdot (1 - \exp(-(\lambda + \lambda_l) \cdot T)) \quad (15)$$

where $K_{air}$ – near to ground level concentration of the specific nuclide calculated by (9); $w_{br}$ – breathing rate = $2.3 \cdot 10^{-4}$ m$^3$/s (for average 24 hr-day) [7]; $T$ – time of exposure =30 days; $\lambda$ – radioactive decay constant, s$^{-1}$, $\lambda_l = 2{,}31 \cdot 10^{-6}$ s$^{-1}$ – leak parameter, s$^{-1}$.

The inhalation effective dose of any nuclide was calculated by (13). Total inhalation effective dose is 0.084 mSv for HEU and LEU fuel.

Under assumption that building leak rate is 100% per day (instead of 20% per day) effective internal dose will be 0.12 mSv for HEU fuel and LEU fuel, the effective external dose will be 0.14 mSv for HEU fuel and LEU fuel.

## 5. Conclusions

The analysis of radiological consequences of the Beyond Design Basis Accident – fuel channel blockage (drop of foreign object into the inlet of the core) for IRT MEPhI was performed. Effective external and internal doses to the operating staff and the member of public were estimated. The results of dose calculations for the reactor with HEU and LEU fuel are presented in Table 3.

**Table 3. Effective external and internal doses to the operating staff and the member of public (HEU / LEU)**

| Category | Conditions (special ventilation, distance) | Exposure time | Effective whole body dose, mSv | | |
|---|---|---|---|---|---|
| | | | External | Internal | Total |
| Staff | special ventilation off, 0 m (reactor room) | 10 min | 92.63/ 87.92 | 7.95/ 7.95 | 100.6/ 95.9 |
| Population | special ventilation on, 200 m | 1 min | 0.055/ 0.053 | 0.082/ 0.081 | 0.14/ 0.13 |
| Population [a] | special ventilation off, 200 m | 30 days | 0.046/ 0.045 | 0.084/ 0.084 | 0.13/ 0.13 |
| Population [b] | special ventilation off, 200 m | 30 days | 0.14/ 0.14 | 0.12/ 0.14 | 0.26/ 0.28 |

[a] For building leak rate 20% per day. [b] For building leak rate 100% per day.

Total effective dose to the member of public for both considered release scenarios (special ventilation on/off after the accident) is less than the annual maximum permissible effective dose to critical group of population during normal operation of radiation-dangerous object determined by radiation safety standards RSS-99/2009 (1 mSv). The distance from the release stack of 200 m is conservatively considered. The closest permanently inhabited house is approximately 500 m from the reactor building.

Annual maximum permissible effective dose to operating staff for planned higher exposure (100 mSv) can be obtained during ~10 minutes of the residence in the reactor room after the accident. The IRT MEPhI emergency plan orders the staff to leave the reactor room during 5 minutes after the considered accident. So presented estimation of effective dose to operating staff is conservative.

Radiological consequences of the considered accident are the same for the reactor with HEU and LEU fuel (under condition that HEU and LEU case have the same reactor power, the same FA number in the core and the same burnup of melted FA).

The analysis performed shows that for low-power reactor without containment (with big air leakage from the reactor room) the accident scenario with operating special ventilation may be preferable from the point of view of the reduction of the doses to population. In this case the release occurs at higher point and its duration is less in comparison with the case of special ventilation shutdown. Moreover, when the special ventilation operates iodine filters can be put into operation and it will reduce the iodine release into the environment.

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