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Comparative Safety Analysis of the MIR.M1 Reactor With Reference to Two Types of Low Enriched Fuel

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ABSTRACT

Presented here is comparative analysis of the results of calculation data analysis of the MIR.M1 accidents with the most conservative initiating events with reference to two types of low enriched fuel, i.e. UO2+Al and U9%Mo+Al. Analysis was performed for the loss-of-heat removal and non-authorized positive reactivity insertion accidents resulted in the worst case probable consequences. According to the results of calculation data analysis, the instantaneous break LOCA leads to melting fuel assemblies with the highest power density as to both types of nuclear fuel. The positive reactivity-induced accident does not cause any fuel failure. This experiment was conducted with funding from the Argonne National Laboratory (USA).

1. Introduction

In accordance with the Russian-U.S. Agreement [1] on studying the possibility of converting six Russian research reactors, an issue of the MIR.M1 reactor conversion to low enriched fuel was addressed. As shown previously [2], for normal operational conditions there is a principal technical capability to use low enriched fuel (19.7% enriched in ^{235}U) in this reactor without significantly deteriorating its physical parameters and application properties. The objective of this paper was to conduct preliminary comparative safety analysis with reference to two types of low enriched dispersion fuel: U9Mo+Al and UO_2+Al.

The paper addresses non-authorized positive reactivity insertion accidents and loss-of-heat removal accidents. Since the reactivity-induced accident does not result in any heavy consequences (no fuel rod failure occurs) only the main results are presented for this accident. The primary circuit pressure pipeline break accident is covered in detail when melting fuel assemblies with the highest power density and fission gas release into the reactor rooms can occur.

2. The MIR.M1 reactor

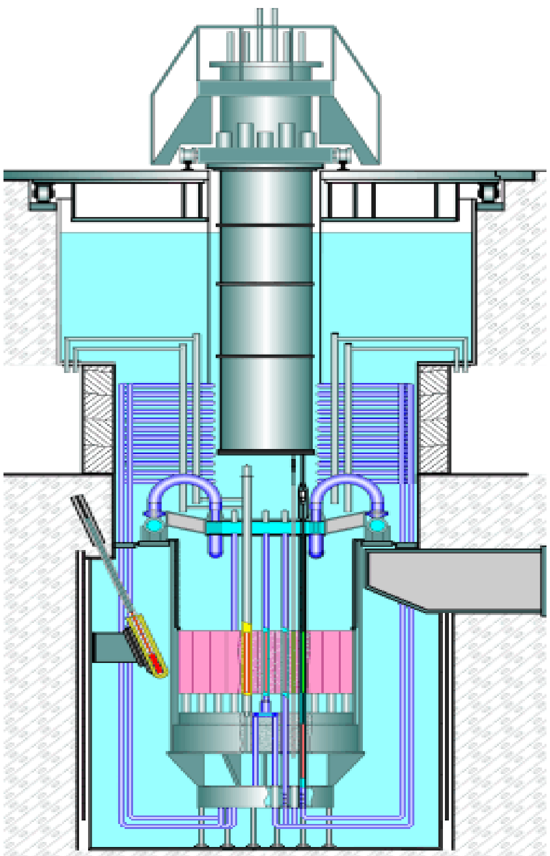


Fig. 1 – The MIR.M1 reactor

The MIR.M1 research reactor (Fig.1) is a heterogeneous loop channel-type reactor immersed in a pool of water operated at thermal neutrons. It is described by a possibility to create local regions of high neutron flux density in the core. The heat is removed from the MIR reactor following a two-circuit process including five circulation paths in the primary circuit, two paths in the pool cooling circuit and four paths in the secondary circuit. The end heat consumer is a cooling tower.

The primary circuit is designed for heat removal from the core and heat transfer to the secondary circuit, which is a water recycling circuit, as well as for maintaining the active medium.

The primary circuit includes five similar circulation paths united by inlet and pressure headers (Fig. 2). Each path consists of pipelines, shut-off and control valves, heat exchanger, and main circulation pump (MCP). The coolant from the MCP pressure header enters via two pipelines the system made of header ring and supply header connected to each other by eleven U-shaped pipes. The coolant flows downwards from the supply header to cool the operating FAs. The outlet pipelines are joined into two hot headers, from which the coolant enters an oxygen activity attenuator connected by two pipelines to a MCP inlet header. A volume compensator and gas purger are connected to the inlet header. From the inlet header the coolant flows to the heat exchangers where it transfers heat to the water recycling circuit.

The heat received by the water recycling circuit coolant from the primary circuit and pool cooling circuit is transferred to the cooling tower medium.

In case of the MIR.M1 reactor primary circuit loss-of-integrity there is a double reactor emergency cooling system (ECS): one system (ECS #1) is connected to the inlet header and the core, the second system (ECS #2) supplies water directly from the reactor pool to the operating channels.

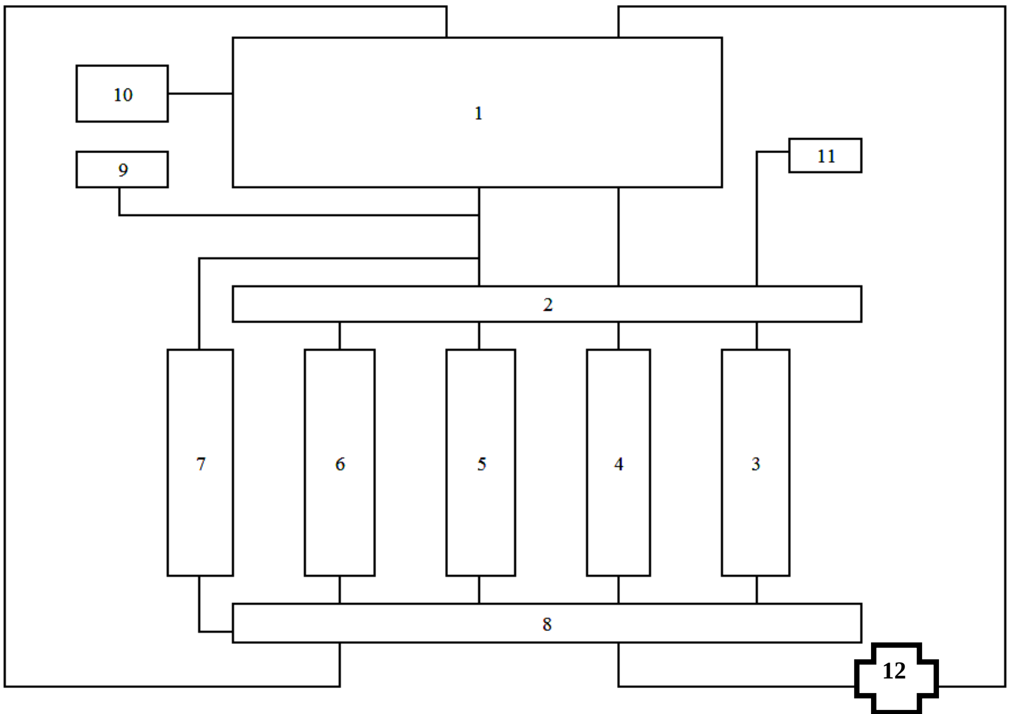


Fig. 2 – The MIR.M1 reactor primary circuit flowchart:

1 – reactor; 2 – inlet header; 3, 4, 5, 6, 7 – pipelines and equipment of the first, second, third, fourth and fifth legs; 8 – pressure header; 9 – pressurizer and degassing system; 10 – ECS #2; 11 – ECS#1; 12 – pipeline break point

3. Calculation model description

Thermal-hydraulic code RELAP5/MOD3.2 [3] is intended for calculation data analysis of a variety of accident and transient processes of the pressurized water reactors. The code was initially intended for application in the PWR reactors, however rather general code principals as well as its flexible structure, which allows simulating processes in the installations of different configurations and scale, presuppose application of the code in safety analysis calculations with reference to other reactors where water is used as coolant.

Using this code transients and accidents can be simulated that occur in the reactor primary and secondary circuits as well as modes in various components such as pumps, hydro-accumulators, valves of different designs, etc. The code allows calculating the modes of large-break accidents with reference to control system failures as well as reactivity-induced accidents.

A MIR.M1 reactor primary circuit calculation model flowchart developed for thermal-hydraulic code RELAP5/MOD.3.2 is presented in Fig. 3.

A FA model is adapted for both fuels. Each fuel rod is a three-layer tube with a fuel layer embedded on the both sides into an aluminum alloy cladding (SAV-6). Inside the inner fuel rod there is a ring-shaped aluminum displacer. The FA outer diameter is 70 mm, and its overall length is 1484 mm.

All FAs are arranged into 6 groups with similar thermal-hydraulic parameters and power density. Since a "hot" header design is divided into two parts, three groups of FAs are connected to one part of the header, and the remaining three groups – to another part. It should be noted that there is a group in the calculation model consisting of one FA, which is the maximum stressed FA. To take into account thermal flux distribution non-uniformity at the perimeter of its fuel rods (azimuthal distribution non-uniformity factor is 1.1), it can be divided into two equal parts. In one part the specific power density is by 10% higher (“hot” part), and in another part – by 10% lower (“cold” part) than the average power density in the FA so that the overall FA power remains unchanged and makes up 3.2 MW. The remaining five groups are hereinafter referred to as “averaged”.

A simulated accident development process is divided into two stages. At the first stage the reactor systems are brought to the stationary operational mode. The accident mode calculation itself is performed at the second stage after the initiating event occurs.

4. MIR.M1 reactor primary circuit loss-of-coolant accident analysis

A double-ended instantaneous full cross-section pressure pipeline break is taken as a postulated initiating event of the primary circuit loss-of-coolant accident. It is assumed that the break occurs in the region between the header ring and valve P-1 at the lower height point (Fig. 2, position 12).

In accordance with [4] the overlapping on the initiating event of an undetected uncontrolled system element failure is considered (in particular, an ECS #2 pump failure) and one failure of any safety system element – a failure of an ECS #1 valve connecting the supply line and channels. In addition, the regulatory documents [5] postulate jamming of one of the most efficient scram rod. Thus, five control rods of six possible are inserted in the core.

The initiating event occurs in the calculation at the 1600th second.

The pipeline break leads to deterioration of the core cooling conditions caused by loss of pressure and coolant flow rate. The core is protected from dewatering by the following systems and equipment:

- ECS #1;
- ECS #2;
- MCP.

In this case a mode of the coolant blowdown into the break mainly impacts these processes determining the time history of the change of pressure and coolant flow rate through the channels in the core.

There are three main phases of the accident. The first phase is reducing the coolant flow rate through a FA and multiple circulation overturn in the FA with the highest power density. This stage is described by the high power density at the sharp loss-of-heat removal, which can lead to departure from nucleate boiling in the maximum stressed FAs.

The second phase is described by the MCP shutoff, coolant backflow through the core at rather high rates in the FA (1÷2 m/s) and ECS actuation.

At the third phase there is a significant reduction of power density in the FA and cooling quasi-steady mode with the flow rates via the channels equal to the feed flow rates.

Early after the initiating event the total coolant flow rate from the primary circuit into the break sharply increases, and there is coolant backflow from the direction of the core. An expansion wave runs along the circuit and the pressure in the core drops sharply leading to the alarm signal at the 0.2th s as per pressure reduction set point in the hot header up to 0.45 MPa (Fig. 3). After the initial strong fluctuations the pressure in the circuit is partially restored and then it decreases gradually.

With a 0.05 s delay the safety rods of 2.8 β_{eff} efficiency will start inserting in the core, which leads to an abrupt reactor power decrease.

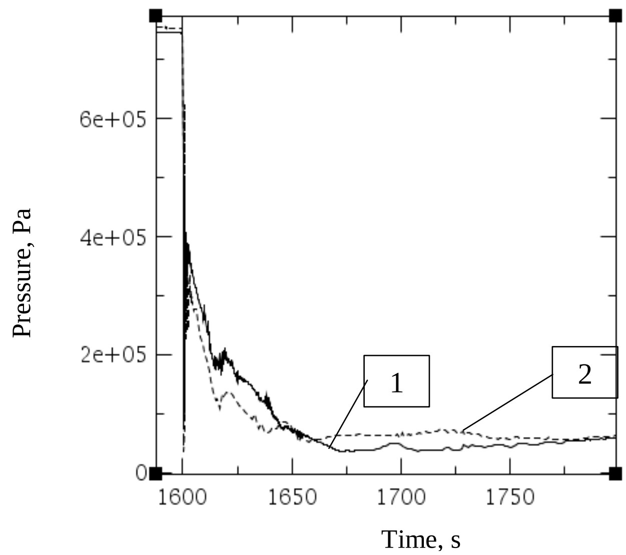


Fig. 3 – Change of pressure in hot headers

(1 – U9%Mo+Al-fuel, 2 –UO_2+Al-fuel)

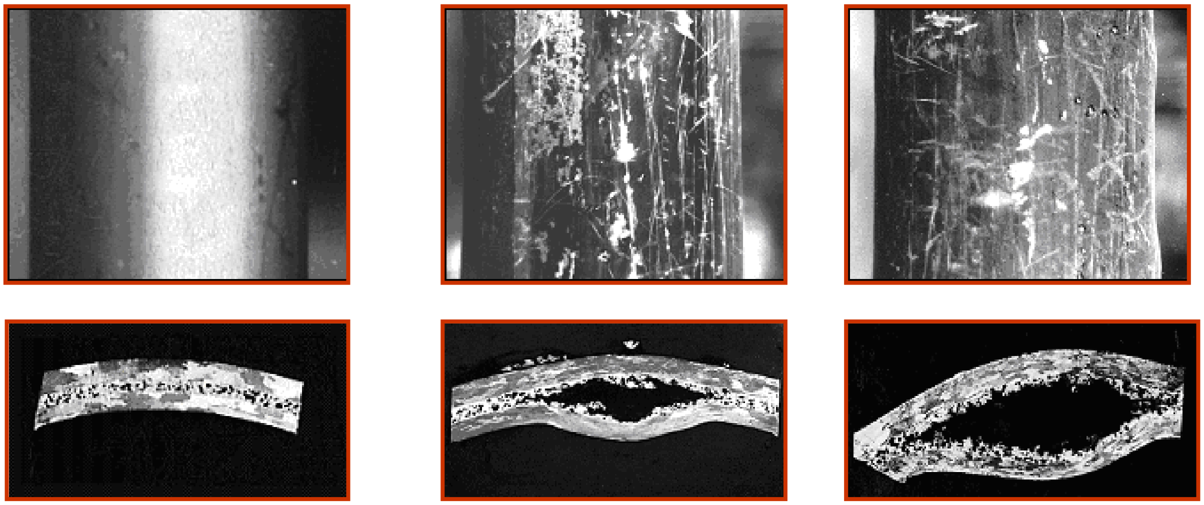
ECS #2 is actuated in case of U9%Mo+Al-fuel at the 54th second of the process, and in case of UO_2+Al-fuel – at the 47th second when an alarm set point is achieved to reduce the level up to 1500 mm in the pressurizer. When the emergency signal starts ECS #1 is also actuated with a 25 s delay, however, the main coolant flow from this system runs into the break, and there is the overlapping of a failure on valve KG-5 in the hot header supply line resulting in no coolant flowing into the core. As per the same set point the signal comes to close the valve and the pressurizer is cut off from the circuit during 33 seconds.

As a result of an abrupt drop of pressure and coolant flow rate via the maximum stressed FA, in some regions in the gaps between the fuel rods a low heat transfer factor flow mode is implemented (~100-300 W/m^{2}xK). During the 6th second of the process the aluminum matrix and cladding melting temperature is achieved (660 ˚C (933 K)) for both FA types due to residual power density in the fuel in the separate maximum stressed regions of the fuel rods. These FAs are not considered in further thermal-hydraulic calculations.

For realistic analysis it is necessary to account the blistering effect of fuel elements. After achieving the temperature ~ (460-550) ˚C due to blistering effect the outer fuel tube contacting the channel body wall, as well as, the internal fuel tube also contact each other [6].

This effect increases radial thermal conductivity and residual power density is removed to the reactor pool water. Thus, the maximal temperature that can be achieved on the fuel rods is determined by the blistering temperature (460-550 ˚C). The effect is possible due to the unique design feature of MIR.M1 reactor (the channel-type reactor placed in water pool and tube-type fuel).

Out-of-pile tests of the MIR irradiated fuel had shown that at temperature of (450÷480) ºC there starts the process of gaseous swelling and blistering (Fig. 4).



Non-heated
fuel element Ø61 mm

Heated to 480°C
fuel element Ø52 mm

Heated to 515°C
fuel element Ø43 mm

Fig. 4 – The process of gaseous swelling and blistering of the MIR irradiated fuel

During the first seconds after the initiating event the averaged groups of FAs with UO_2+Al-fuel show a slight increase in the temperature of the fuel rod claddings (Fig. 5) not exceeding 430K (157 ˚C) due to changes in the flow mode caused by fluctuations of pressure and coolant flow rate. From the 130th second of the process fluctuation modes of coolant flow in the channels are set leading in their turn to the fuel cladding temperature fluctuations. At that, the second and third less-stressed groups of FAs show the maximal coolant heating (up to the saturation temperature) as well as coolant boiling along the entire length of the fuel rods.

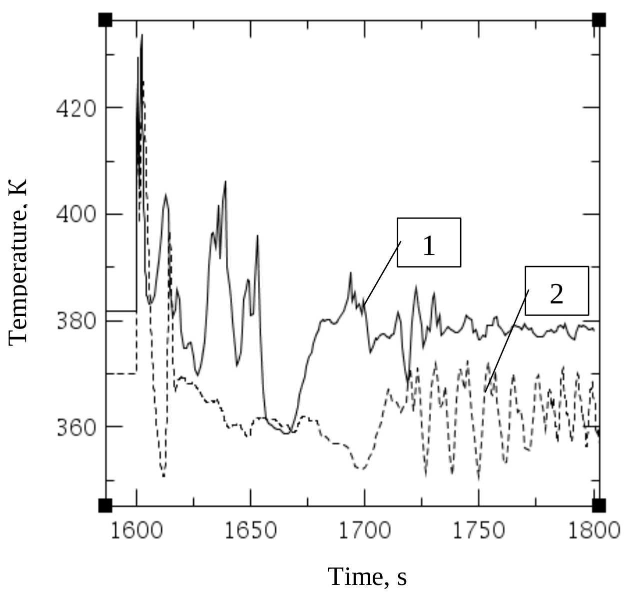


Fig. 5 – Maximal cladding temperature of the averaged groups of FAs
(1 – U9%Mo+Al-fuel, 2 –UO_2+Al-fuel)

In the averaged FA groups with U9%Mo+Al-fuel the cladding temperature does not exceed 440K (167 ˚C). At the 150th second there is a relatively stable distribution of the coolant flow rate in the channels under calculations that results in some stabilization in the temperature mode of the fuel rods. The second less-stressed FA group shows the maximal coolant heating (up to the saturation temperature) as well as coolant boiling at the fuel rod top.

ECS #2 compensates sufficiently the coolant loss in the reactor channels, which under a significant decrease of power density provides safe heat removal from 5 FA groups after the 200th second (except for the maximum stressed FA).

Thus, as a result of the calculation study of the accident with a MIR reactor primary circuit pipeline break of the with actuation of the emergency cooling system from the pool cooling circuit it is shown that ECS coolant supply from below into the FA operating channels ensures distillate entering the core. That is why after the residual power density reduction the core will be filled with water, which will provide safe heat removal from the FAs (except for the maximum stressed FA).

5. Calculation results of the accident with a non-authorized extraction of the maximum efficiency shim rod

The accident initiating event (shim rod non-authorized extraction) occurs at the 100th second.

According to the calculation results the accident temporal development with moving a shim rod is practically the same for both fuels (Table 1).

Table 1 – Consequence of the key event during the accident development

Duration of process for fuel U9%Mo+Al, s	Duration of process for fuel UO_2+Al, s	Description of the event
0	0	All the reactor system are in the stationary state
0+	0+	Positive reactivity insertion - shim rod is extracted
2.2	2.28	Pre-alarm set point is achieved to actuate scram rods in as reactor power increases
3.5	3.5	Emergency set point is achieved to actuate scram rods
3.55	3.55	Scram rods drop into core, driving FAs with an absorber start inserting into the core (accounting the delay time).
3.6	3.8	Reactor power achieves the maxima value 0f 97.lMW; fuel and coolant temperature increases
4.0+	4.0+	Power decreases. The reactor is in the sub-critical state
100+	100+	New core cooling mode is stabilized

Figure 6 presents the time history of the maximal temperature of fuel and the maximum stressed fuel rod cladding as well as washing coolant temperature at the elevation with the maximal temperature of fuel and cladding for both fuels.

The temperature values for UO_2+Al at the peak load at the 3.8th second are 181 ºC (454 K) – for the fuel meat and 172 ºC (445 K) – for the fuel rod cladding. The temperature of the coolant washing the stressed fuel rod at that point makes up 89 ºC (362 K) not exceeding the saturation temperature at the set pressure. The minimal departure from nucleate boiling ratio (Bernat’s

correlation) is achieved at the maximal fuel rod heating making up K ~1.7.
The temperature values for U9%Mo+Al at the peak load at the 3.6th second make up 185 ºC (458 K) – for the fuel meat and 176 ºC (449 K) – for the fuel cladding. The maximal temperature of the coolant washing the stressed fuel rod at the same point is 88 ºC (361 K) that does not exceed the saturation temperature at the set pressure. The minimal departure from nucleate boiling ratio (Bernat's correlation) is achieved at the maximal fuel rod heating making up K ~1.5.

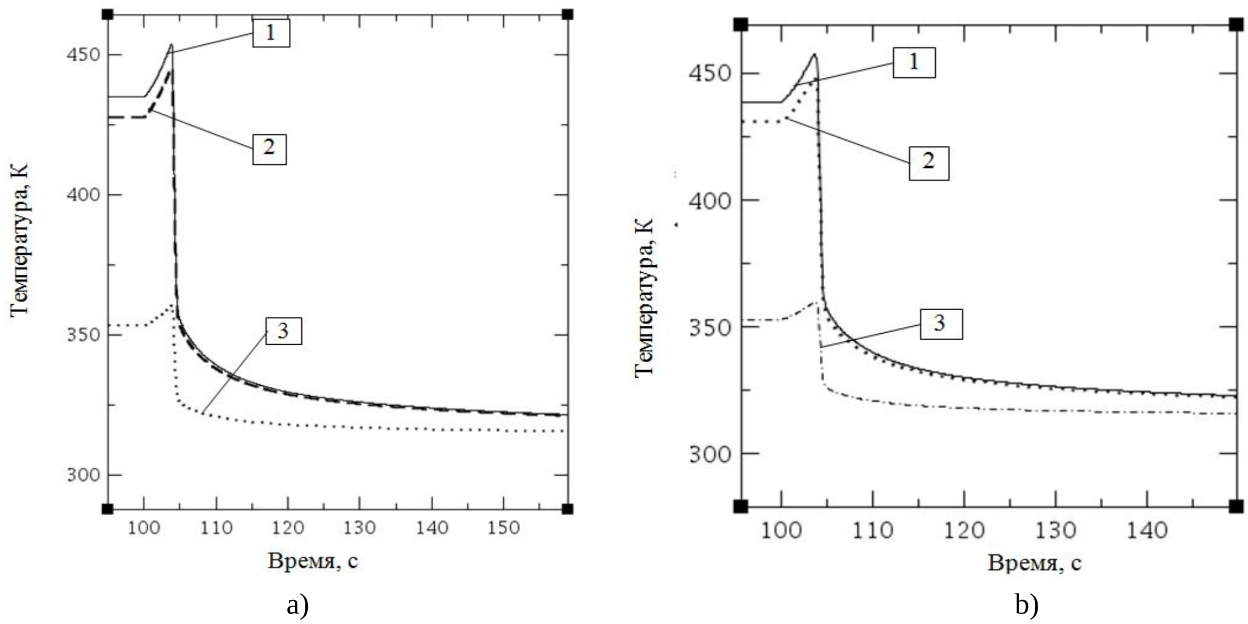


Fig. 6 – Change in the fuel maximal temperature (1), fuel rod cladding (2) and coolant (3): a - UO_2+Al; b - U9%Mo+Al

6. Conclusion

The comparative analysis of the MIR.M1 reactor safety with LEU fuel based on UO_2+Al and U9%Mo+Al was performed using the calculation data analysis during the design-basis accidents with loss-of-heat removal from the core in one case and excess reactivity insertion in another case.
In case of a non-authorized extraction of a shim rod at the design rate the safety requirements are met for both fuels. No fuel failure occurs. The safety system is capable to bring and maintain the core in the sub-critical state.
In case of loss-of-heat removal from the core due to the primary circuit pipeline break a failure of the stressed FA may occur in the cores with both fuels. From other FAs the heat removal is provided by the reactor emergency cooling system.
As it was shown in feasibility of the MIR.M1 reactor safety, in the core with highly-enriched fuel this accident results in a failure of the maximal stressed FA and fission product release beyond the limits of the primary circuit. At that, the radioactive fission product release into the environment does not exceed the established limits. Therefore, in the core with both LEU fuels in case of the loss-of-heat removal from the FAs the radioactive fission product release will not exceed the established values and no measures are needed to be taken for the staff and population protection from the overexposure.

7. Gratitude

The authors express their gratitude to the ANL experts (Dr. Jordi Roglans-Ribas, Dr. Nelson Hannan and Dr. Patrick Garner) for their assistance in conducting this research.

8. References

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