RERTR 2014 – 35TH INTERNATIONAL MEETING ON
REDUCED ENRICHMENT FOR RESEARCH AND TEST REACTORS

OCTOBER 12-16, 2014
IAEA VIENNA INTERNATIONAL CENTER
VIENNA, AUSTRIA

Analysis of Beyond DBA Consequences of the IR-8 Reactor Primary Pipes Rupture during Conversion to LEU

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ABSTRACT

As a part of the GTRI program NRC KI is performing the studies to establish the feasibility of converting the IR-8 research reactor to low enriched uranium (LEU) 19.7% enrichment of U^{235} fuel with financial support from U.S. Department of Energy. In 2012 after the completion of the IR-8 conversion feasibility study the work to assess safety of IR-8 reactor during conversion to LEU fuel was begun. Currently the analysis of beyond DBA consequences is carried out. The analysis of reactor primary pressure and suction pipes rupture consequences is finished. Calculations of full instantaneous rupture of pressure pipes for initial loading of the core with HEU, four mixed loadings, first and equilibrium loadings with LEU were performed using one-dimensional BEREZA code. For confirmation of the obtained results the same calculations for initial loading of the core with HEU and equilibrium loading with LEU were performed using system ATHLET code.

1. Introduction

Currently the analysis of beyond design basis accidents during conversion of the IR-8 reactor from HEU to LEU is carried out in NRC "Kurchatov Institute". Among the beyond design basis accidents the full instantaneous rupture of the primary pipes was considered [1].

The calculations were performed using BEREZA code [2] for starting loading of the core with HEU [3], four mixed HEU — LEU loadings: #HLEU01, #HLEU03, #HLEU05, #HLEU07, for the first and the equilibrium loadings with LEU [4].

For confirmation of the obtained results the same calculations were performed using the system ATHLET code [5] for starting loading of the core with HEU and equilibrium loading with LEU.

2. Brief calculation codes description

BEREZA code

The BEREZA code is designed for the analysis of transient and accident thermal regimes of pool and tank-types research reactors. This code allows to determine the fuel element temperature, the enthalpy and the coolant temperature and heat flux from the fuel element to the coolant as the function of the longitudinal coordinate and time. This sets arbitrary functions depending on the time of power density in a fuel element, the coolant flow rate, its enthalpy and pressure at the core inlet. The code provides the ability to calculate modes with surface or bulk boiling of the coolant, and the fuel elements cooled by natural circulation. Instead the entire core the elementary channel with maximum power density in a fuel element is considered.

ATHLET code

The thermal-hydraulic system ATHLET code (**A**nalysis of **T**hermal-**H**ydraulics of **LE**aks and **T**ransients) is developed by GRS mbH (Gesellschaft für Anlagen- und Reaktorsicherheit) and was initially designed for the analysis of the whole spectrum of leaks and transients in LWR reactors like PWR and BWR types. However, the experience has shown that the code may be fully used with success for Russian WWER type reactors.

ATHLET is composed of several basic modules for the simulation of the different phenomena involved in the operation of light water reactors:

— thermo-fluid dynamics (TFD),

— heat transfer and heat conduction (HECU),

— neutron kinetics (NEUKIN) for description of one-point and one-dimensional kinetics,

— description of the equipment work (GCSM),

— FEBE method for the numerical integration implements fully implicit scheme.

Other independent modules (for example, 3D neutron kinetics) can be connected by means of a general interface.

The main TFD module is based on the five-equations system (conservation equations for mass and energy for the liquid and vapor phase separately and general equation for the angular momentum of the mixture with the flow drift) or on the six-equations system (conservation equations for mass, energy and momentum for the liquid and vapor phases at the description of two-fluid model) with the large set of closing relations. The reactor primary circuit is simulated by connecting basic thermo-fluid dynamic objects (TFO). The ATHLET code is under verification based on the experimental data, commissioning results of reactor WWER-1000, solution of standard international problems.

3. IR-8 reactor

IR-8 is a pool type research reactor with power up to 8 MW. It uses light water as moderator, coolant and top shield (figure 1). The IR-8 reactor core consists of 16 IRT-3M FAs with tubular elements of square cross section. The fuel is UO_2 with 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 (figure 2). The 13 rods with boron carbide absorber are used as control rods of CPS.

Starting loading of the core with HEU and equilibrium loading with LEU are presented in figures 3, 4.

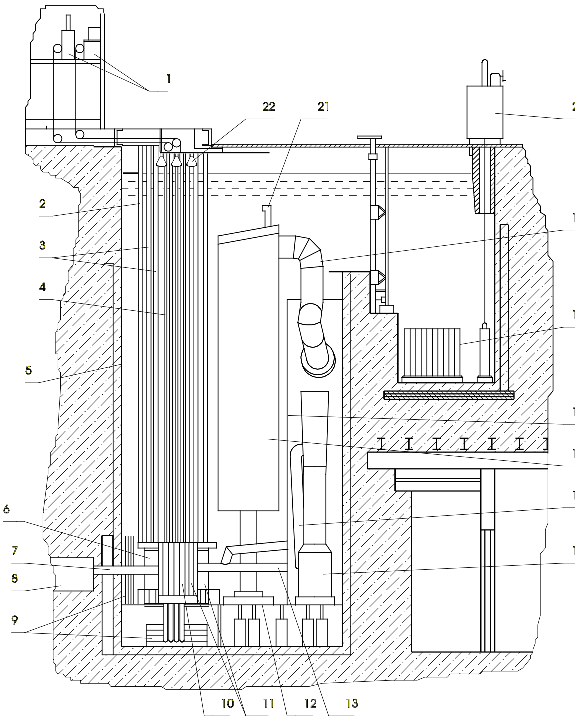


Fig. 1. Cross-section of the IR-8 reactor:
1 — CPS rods drives, 2 – channel of the CPS ionizing chamber, 3 — vertical experimental channels, 4 – the CPS rod channel, 5 — reactor tank (pool liner), 6 — reactor vessel, 7 – horizontal experimental channel — beam tube (HEC), 8 – gate of HEC, 9 – steel screens, 10 — FA, 11 – beryllium reflector, 12 – intermediate bottom, 13 — ultracold neutrons source channel, 14 – the ejector (water-jet pump), 15 – head pipeline, 16 – delay tank, 17 – vertical punched partition, 18 – cells of spent FA storage, 19 — suction pipeline, 20 — transport container, 21 -air outlet, 22 -spraying facility.

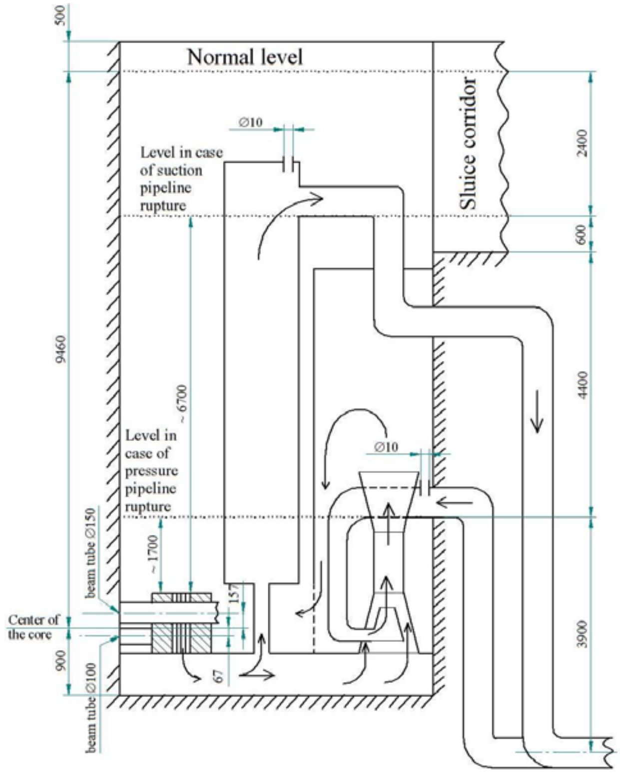


Fig. 2. Scheme of the reactor pool with pressure and suction pipes

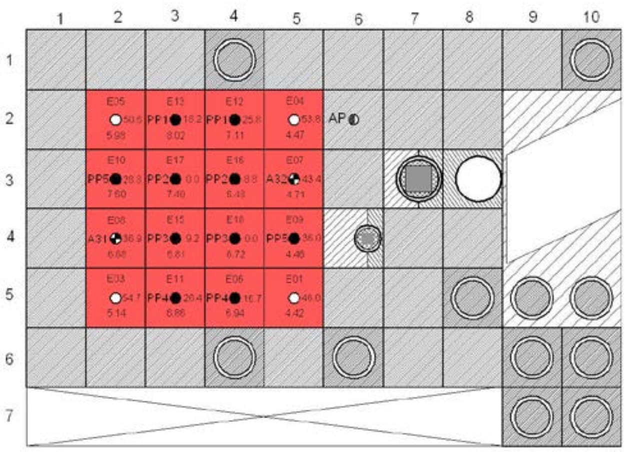


Fig. 3. Values of FA burnup and power of FAs with HEU (BOEC)

Legend

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 and plug

beryllium block with automatic regulating rod

gamma-radiation shield with AR in 6-4 cell

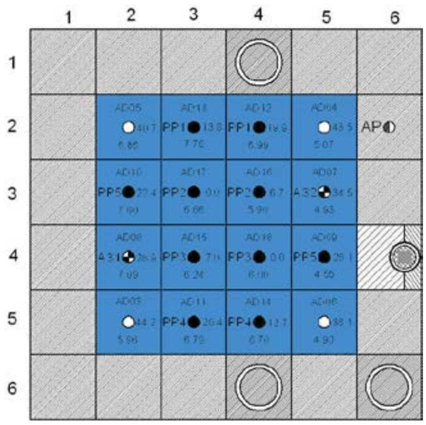
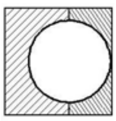
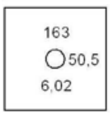


Fig. 4. Values of FA burnup and power of FAs with LEU (BOEC)



gamma-radiation shield for AR in 8-3 cell



— FA number
— average burn-up U^{235} in FA, %
— power of FA (%)

 gamma-radiation shield with AR in 7-3 cell

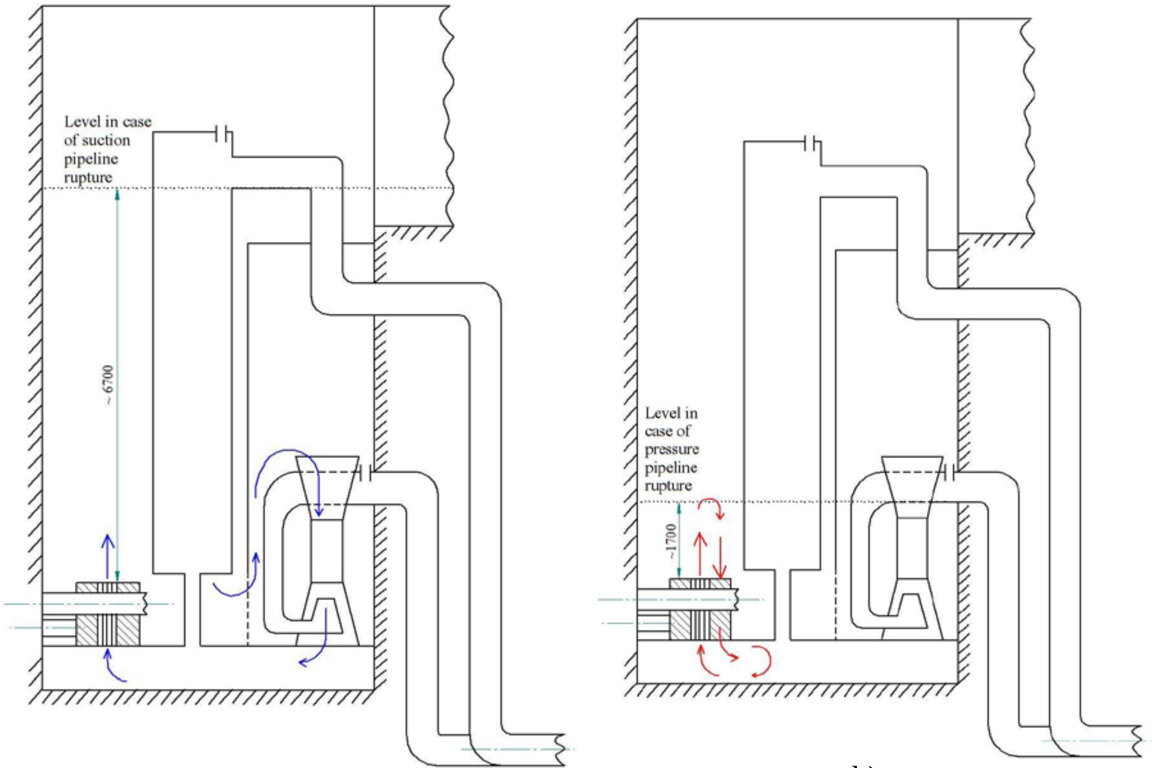
4. Full instantaneous rupture of the primary pipes

In case of full instantaneous cross-section rupture of the primary pipes (suction or pressure) out of the pool tank, the reactor will be shut down as a result of scram due to deviations from the normal value of the primary coolant flow rate or pressure, or drop of the core pressure or pool water level.

Rupture of one or both primary pipes does not lead to a complete unwatering of the reactor pool.

4.1. Lowering of the water level in the reactor pool at rupture of pressure and suction pipes

In case of the suction pipe rupture the water will flow from both pools to pump room through the core (and parallel through holes of the pressure box and through the ejector in the direction opposite to normal), through the delay tank and part of the suction pipe. There is a hole in the top of the delay tank for elimination of the “siphon effect”. Therefore after 1.5 minutes after rupture, when the level in the pool drops below the top of the delay tank, drain of the water stops, and the level is established at 6.7 m above the top of the core (figure 5).



a) b)

Fig. 5. Scheme of water movement at natural circulation:
a) after rupture of the suction pipe, b) after rupture of the pressure pipe

In case of the pressure pipe rupture after the heat exchangers, water will flow by gravity from the reactor pool into the pump room through the ejector nozzle and part of the pressure pipe.

In the reactor pool the main space, the pressure box and the delay tank will drain simultaneously because they are interrelated. Until the level is above the bottom of the sluice corridor between the pools, except for the water from the reactor pool, the water from storage and sluice corridor will also drain. Except for the flow by gravity, until the water level will drop below the top of the delay tank, and the air will enter into a hole in the top, the water will be pumped from the pool to pump room by the primary pumps.

Below the mark of the pressure pipes passage through the wall of the pool the level will not decrease because near this passage the port for disruption of a siphon drain is provided in pipes, which is connected to the pool and additionally the vent pipe.

The air will enter the port, filling the upper pipe elbow and draining will stop. Above the core the ~1.7 m layer of water will remain (figure 5). Time from the moment of rupture to the moment of level decrease down to the passage mark of the pressure pipes through the wall of the pool will be equal to ~4.4 min.

When the drain of water from the pool is terminated, then forced movement of water through the reactor (by gravity) stops. At rupture of the suction pipe water passed the core up from the bottom, returns to its lower face through the ejector and through the reflector. At rupture of the pressure pipe (more severe case), as the level drops below the outlet section of the ejector, the water returns only through the reflector.

Further cooling of the reactor occurs by natural circulation. Decay heat, removed from the core by natural circulation, accumulates in the pool water.

Decay heat is determined using the MCU-PTR code [6].

4.2. Calculation of the core cooling at the pressure pipes rupture using BEREZA code

The results of the calculations of temperatures after pressure pipes rupture at reactor operation on power of 8 MW are given in table. The maximum power density in the meat of the most heat-stressed FA (in cell 3-2) — 3574 MW/m^3 for equilibrium loading with HEU and 2844 MW/m^3 for equilibrium loading with LEU.

Table. Pool water temperature (t_{pool}), water temperature at outlet of the most heat-stressed gap (t_{gap}) and fuel element (t_{FE}) after primary pipes rupture

Time after rupture	t_{pool}	Loading with HEU		Loading with LEU	
		t_{gap}	t_{FE}	t_{gap}	t_{FE}
5 minutes	51	89	94	91	96
20 minutes	55	86	89	87	91
1 hour	60	84	86	85	88
3 hours	70	89	91	90	92
6 hours	82	98	99	99	100
9 hours	91	104	105	104	105
13.5 hours	104	104	105	104	105

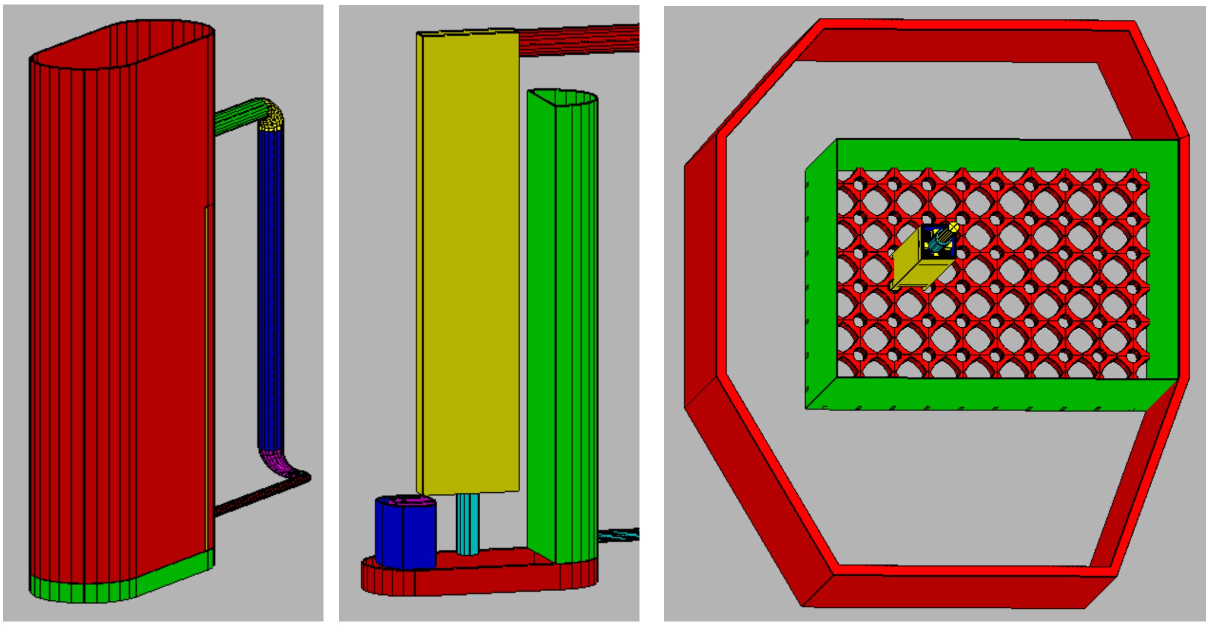
When water pressure in the core is ~ 1,2·10^5 Pa, the saturation temperature is 104 °C. As seen in table, the water temperature at the outlet of the most heat-stressed gap reaches this value after 9 hours, after which boiling begins. Approximately after 13.5 hours all the water in the pool heats up to 104 °C, boiling will occure throughout the core. Water temperature in the FA gaps reaches 104 °C, and temperature of fuel elements will be almost the same.

Water and fuel element temperatures for four mixed HEU — LEU loadings and for the first loading with LEU slightly differs from the values given above.

4.3. Calculation of the core cooling at the pressure pipes rupture using ATHLET code

This report, related to beyond design basis accident analysis, in fact is the first full-sized hydro-dynamic modeling of the IR-8 research reactor and includes practically all the main objects of the reactor. During the creation of the input data block for the improved assessment code ATHLET the technique [7] was used, which allows the high accuracy description of the geometrical characteristics of the objects and their hydro-dynamic interrelation. The point is that in the beginning the 3D-model of the objects is created based on the available drawings, pictures etc., and then based on them and according to the specifications designed for ATHLET the data blocks are formed, which are responsible for object description and interrelation (topology), control of the system etc. For these purposes the special preprocessor is designed, which finally automates this work and allows rather simple and fast making changes in the formed data, if necessary. In this case at the performing of the calculations not only the reactor is modeled but the primary and secondary circuit with control system with necessary detailing for certain calculations [8—11]. Quite successfully this method was applied at modeling of the UPTF experimental system [12], which was used for research of efficiency of the safety systems of some reactors types in Germany, USA and Japan.

In figures 6, 7 the intermediate result of the preprocessor (three-dimensional representation of the main hydro-dynamic objects and thermal structures) for the preparation of input data for ATHLET code is presented.



a) b) c)

Fig. 6. Basic macro objects of the reactor pool:

lower and upper parts of the reactor pool with the primary circuit pipe (a); bottom of the pool and internal elements of top of the pool — the core with the reflector, the delay tank with inlet pipe and part of the primary circuit suction pipe, the pressure box with part of the primary circuit pressure pipe (b); reactor vessel with fuel assembly and CPS channel on the supporting grid (c).

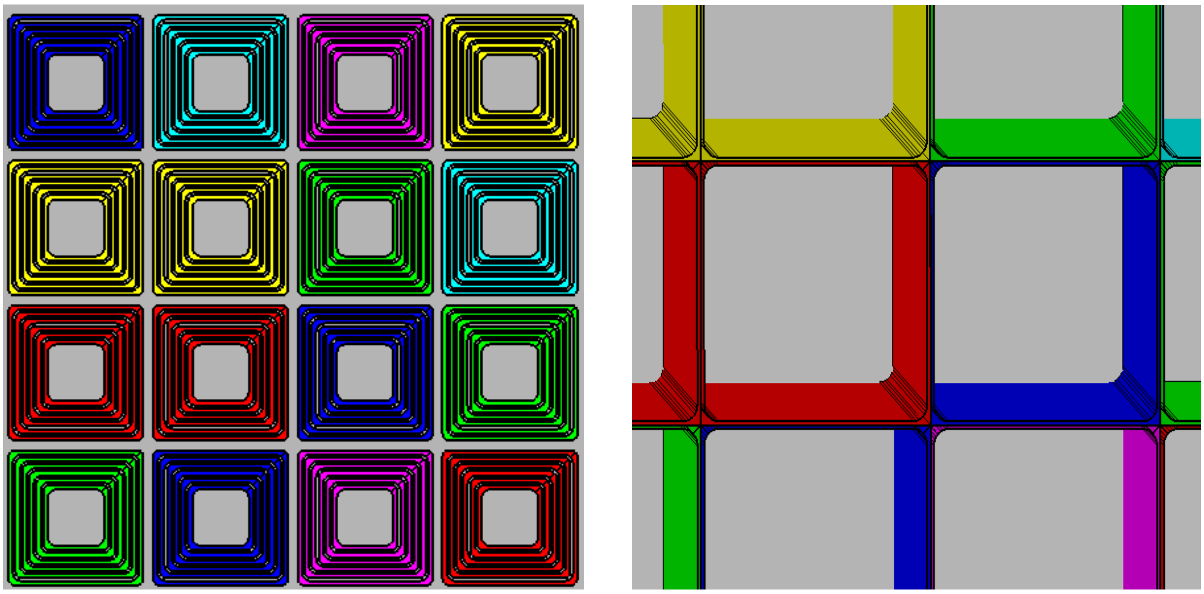
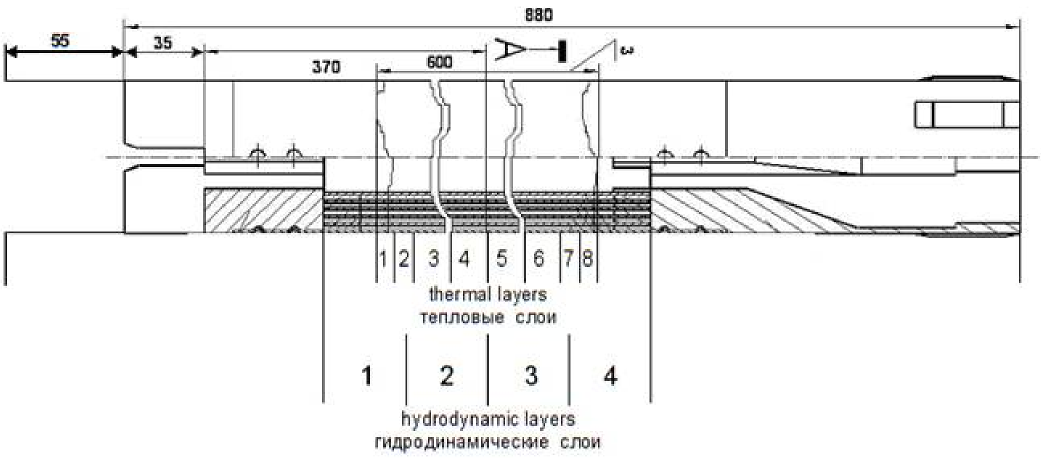


Fig. 7. Hydraulic objects of FAs (without the channels with control rods) and the fragment of hydraulic objects between FAs

The coolant flows are in the following path. In the primary circuit of the reactor (it is modeled more simply than real circuit, the only main elevations and the volume are kept) the pump system is installed, after which the flow passes through the heat exchangers and into the reactor pool — in the ejector nozzle.

At its top the pool is connected with the room, filled with air, which is indirectly related to the environment. It can be assumed that in this room the constant pressure and temperature are maintained. This fact is also taken into account during creating a design scheme. All the objects above are included as a part of the microobjects design scheme with its internal detailing, mainly associated with the partition along the axis. The feature of the whole system is that the volume of connected macroobjects may vary between a few orders of magnitude, from a few cubic meters (the top of the pool, pressure and suction boxes), to the tenths of the liter (objects in the space between the reactor fuel elements). The legend for calculation zones in FAs of the core for calculations using ATHLET code is shown in figure 8.

BOT



TOP

Fig. 8. Legend for calculation zones in FAs of the core

The results of the calculations for equilibrium loadings with HEU and LEU after pressure pipes rupture are shown in figures 9—12.

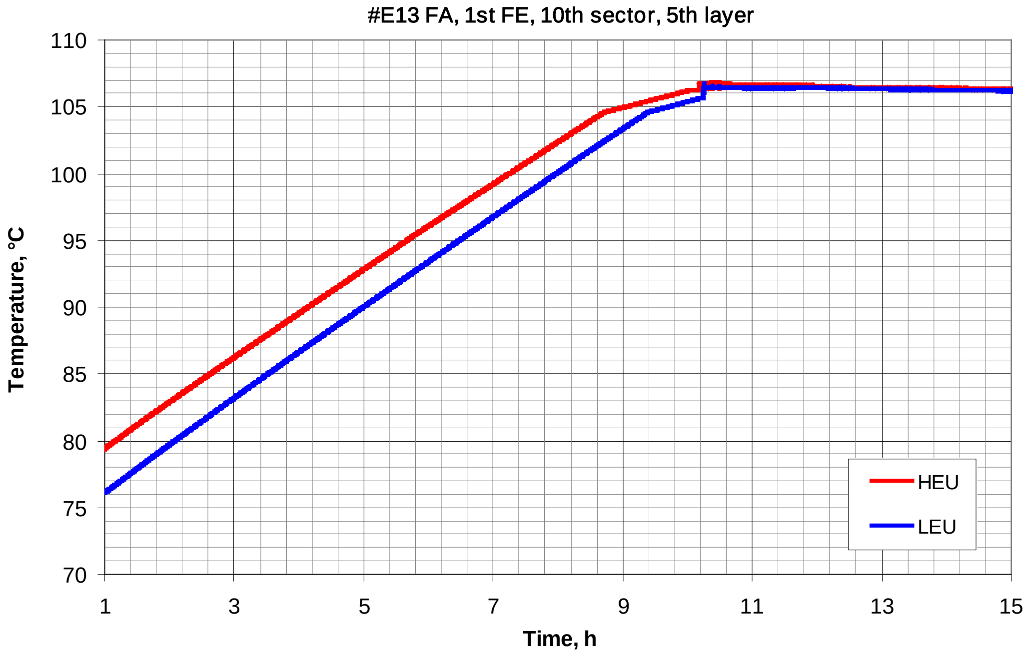


Fig. 9. Change in maximum fuel element inner surface temperature after pressure pipes rupture

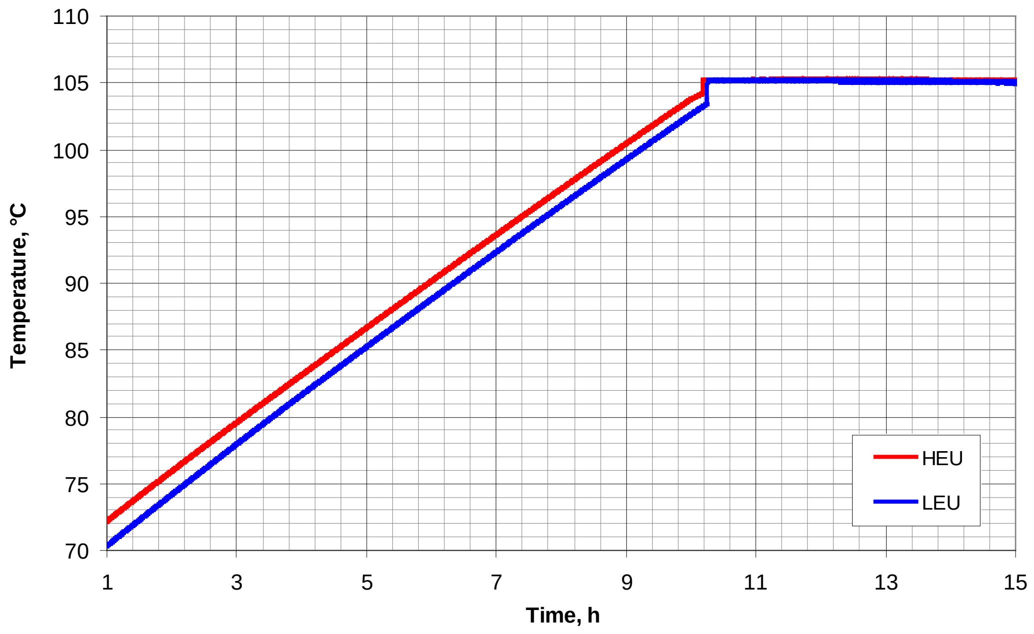


Fig. 10. Change in maximum coolant temperature in gap between the 1st and the 2nd fuel elements after pressure pipes rupture

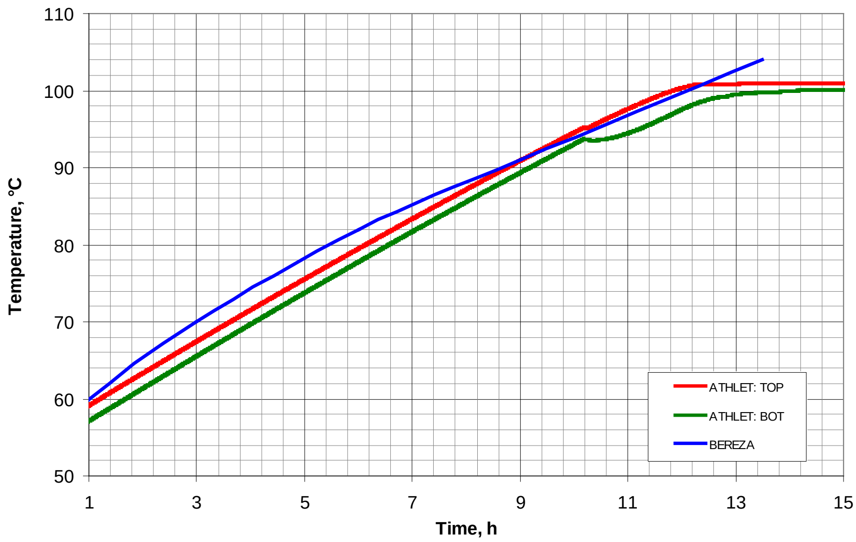


Fig. 11. Change in pool water temperature after pressure pipes rupture (HEU)

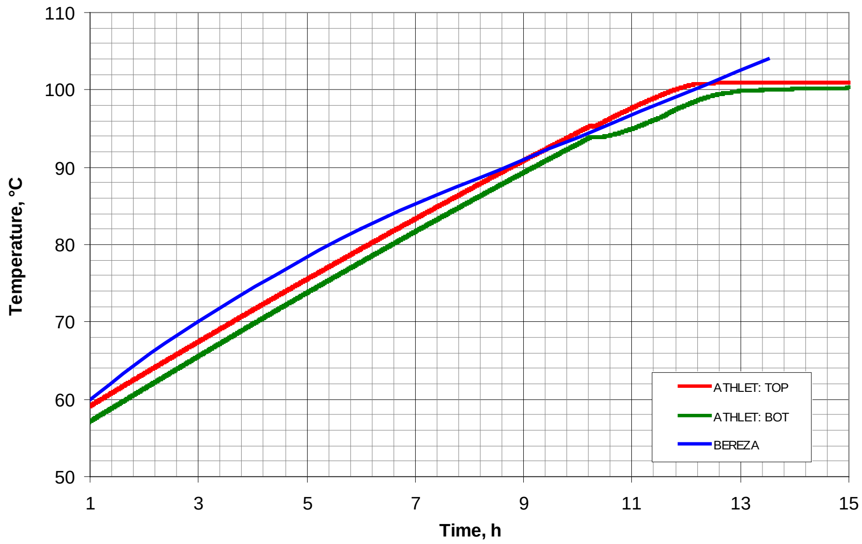


Fig. 12. Change in pool water temperature after pressure pipes rupture (LEU)

5. Conclusions

The analysis of the full instantaneous rupture of the primary pipes has been fulfilled. Based on the calculations, the following results were obtained. In case of rupture of suction or pressure pipe the water level at the pool is reduced, respectively, in ~ 1.5 or ~ 4.4 minutes to the mark of 6.7 or 1.7 m above the top of the core. During this time the core is cooled by forced movement of draining water, and after that — by natural circulation. Using the BEREZA code, a cooling by natural circulation at an accident with rupture of pressure pipe (i. e. worst case) was considered. The calculations have been fulfilled for the starting loading with HEU fuel, four mixed loadings HEU — LEU, and for loadings with LEU fuel (the first and the equilibrium). There is no significant difference in obtained results for all this loadings. Water temperature at the outlet of the heat-stressed gap reaches the saturation level (104 °C) after ~9 hours, then boiling begins. Approximately after 13—15 hours all the water in the pool warms up to temperature of 104 °C, and the boiling throughout the whole core begins. Water temperature in the FA gaps reaches the 104 °C, and temperature of fuel elements is almost the same.

In order to confirm the obtained results similar calculations were performed using system ATHLET code for starting loading of the core with HEU and equilibrium loading with LEU. The obtained results are quite similar to the results of calculations using BEREZA code, the values of temperature differ by several °C.

6. References

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