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Progress in Safety Assessment of the IR-8 Reactor During Conversion to LEU Fuel

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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. Conversion is possible using the IRT-3M fuel assemblies (FAs) with U-9% Mo LEU fuel instead of high enriched uranium (HEU) FAs with UO_2. Now the IRT-3M FA with LEU is under development and licensing of U-9% Mo fuel. The main neutron and thermal-hydraulic characteristics of the core during conversion were determined. The radiation safety analysis for the IR-8 reactor during normal operation and possible accidents was finished. Currently analysis of the postulated accidents consequences for design basis accidents (DBA) is completed and continuing for beyond DBA.

1. Introduction

After finishing "The feasibility studies of the IR-8 research reactor conversion to LEU fuel" [1] NRC "Kurchatov Institute" under the GTRI program began to evaluate the IR-8 reactor safety during conversion to LEU [2, 3].

2. Tasks for assessing the safety of the IR-8 reactor during conversion to LEU

Currently the studies to assess safety of IR-8 reactor during conversion to LEU fuel are continued. These studies include implementation of the following tasks:

- Neutronic and thermal-hydraulic calculations
- Radiation safety
- Analysis of possible accidents

Analysis is to be performed for various transients (RIA, LOFA and LOCA), including design basis and beyond design basis as required by the Regulatory Body (RTN). The list of initiating events is made on the basis of the existing SAR of IR-8 reactor for the HEU fuel, and the recommendations for research reactor safety of the IAEA.

Neutronic and thermal-hydraulic calculations

For neutron-physical calculations the MCU-PTR code with database MDBPT50 was used [4]. The MCU-PTR code with database MDBPT50 is certified for calculation neutronic characteristics of the IR-8 research reactor taking into account fuel burn-up, absorber burn-up in control rods, beryllium reflector poisoning and control rods insertion. For steady-state thermal-hydraulic calculations the ASTRA code [5] was used.

Justification of radiation safety

For calculations of fission product activity the MCU-PTR code was used.

To evaluate the radiation situation and the expected irradiation to the population at long and short-term radioactive releases into the atmosphere a certified NUCLIDE and NUCLIDE - ACCIDENT codes were used. NUCLIDE and NUCLIDE - ACCIDENT codes are parts of GARANT-UNIVERSAL software package [6—9] for calculations of atmospheric air pollution by radioactive substances.

Calculation of the radiation situation in the reactor hall by the scattered radiation from the storage of spent fuel assemblies (SFA) with assumption of complete unwatering of irradiated assemblies was carried out using the MicroSkyshine code (version 2) [10].

Analysis of possible accidents

For analysis of the consequences of pre-emergency accidents and DBA accidents the PARET/ANL Version 7.6 [11] code was used. The PARET/ANL code is designed for use in predicting the course and consequences of nondestructive reactivity insertion (RIA) and loss-of-flow (LOFA) accidents in research and test reactor cores.

A calculation analysis of basic neutronic and reactivity characteristics of the reactor and also analysis of the criticality of fuel storage ("fresh" and spent fuel assemblies at the complete unwatering of the SFA storage) were performed using the MCU-PTR code.

For analysis of the consequences of beyond DBA the following codes were used:

- PARET/ANL
- BEREZA [12]
- ATHLET [13]

The BEREZA code is designed for the analysis of transient and accident thermal regimes of research reactors pool and tank-types. The code provides the ability to calculate modes with surface or bulk boiling of the coolant, and the fuel elements cooled by natural circulation.

The thermal-hydraulic system ATHLET code (Analysis of Thermal-Hydraulics of LEaks and Transients) was initially designed for the analysis of the whole spectrum of leaks and transients in Light Water Reactors (LWR) like Pressurized and Boiling Water Reactor (PWR and BWR) types. However, the experience has shown that the code may be fully used with success for Russian WWER type reactors. The ATHLET code is under verification based on the experimental data, commissioning results of reactor WWER-1000, solution of standard international problems.

3. Loadings of the core during conversion from HEU to LEU

Taking the BOEC #35 [2] as a starting loading, calculations for eight equilibrium cycles with the reactor core consisting of HEU in the regime of partial reloading of 2 FAs using new calculating model with azimuthally divided fuel elements were carried out using MCU-PTR code. On the results of the neutronic and thermal-hydraulic calculations of equilibrium loadings of the reactor in the regime of partial reloading the characteristics (neutron, thermal hydraulic and reactivity) of equilibrium core with HEU, which was chosen as a starting point for reactor conversion to LEU (figure 1).

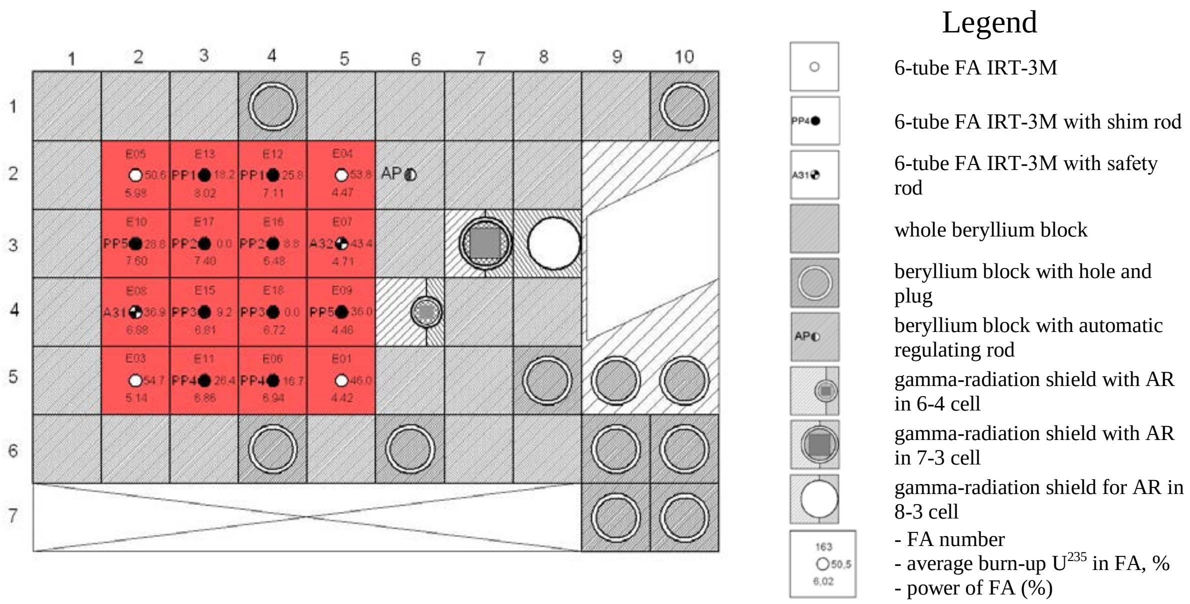

Fig. 1. The values of fuel burnup and FAs power for the loading with HEU (BOEC)

Conversion of the core from 16 FAs with HEU to LEU is to be carried out in 8 transitional cycles replacing two FAs in each cycle. Calculations of the fuel burnup in FAs and excess reactivity were performed for transitional loadings from HEU to LEU (# HLEU01, # HLEU02, # HLEU03, # HLEU04, # HLEU05, # HLEU06, # HLEU07) and first loading with LEU (# LEU08).

The thermal-hydraulic analysis was performed for four mixed loadings and the first loading with LEU in steady state. It was obtained that for mixed loadings # HLEU01 (figure 2), # HLEU03 and # HLEU05 (figure 3) reactor power should be limited to the value of 7.0, 7.5 and 7.6 MW accordingly; For the nominal power of 8 MW the safety margin to onset of nucleate boiling (ONB) for these three mixed core loadings is lower than the minimum allowed value of 1.4. The analysis for loading # HLEU07, first loading with LEU (figure 4) and the equilibrium loading of the core with LEU (figure 5) showed that operation of the IR-8 reactor with power of 8 MW is allowable.

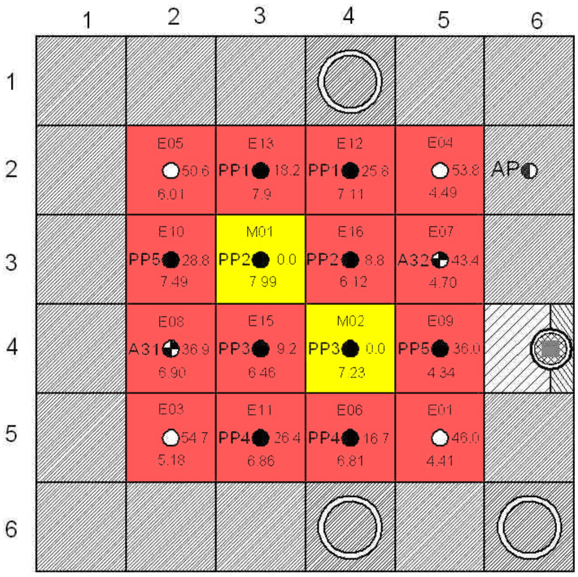

Fig. 2. Fuel burnup and FAs power for the mixed loading # HLEU01 (BOC)

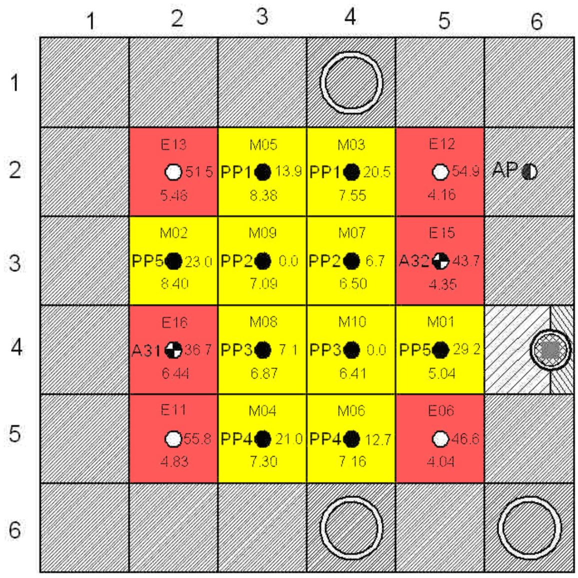

Fig. 3. Fuel burnup and FAs power for the mixed loading # HLEU05 (BOC)

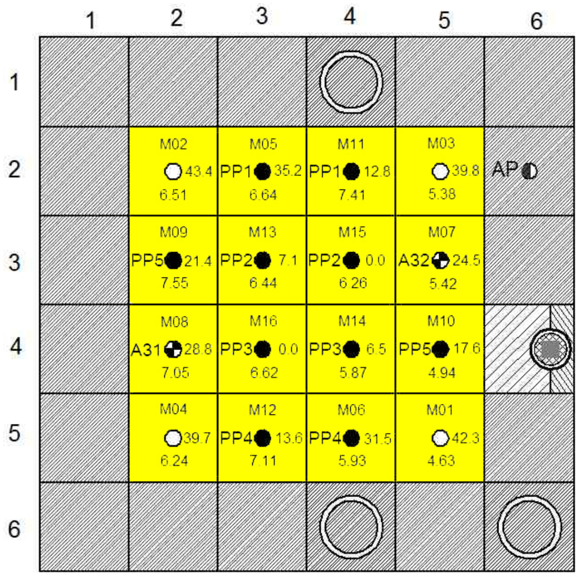

Fig. 4. Fuel burnup and FAs power for the first loading with LEU (BOC)

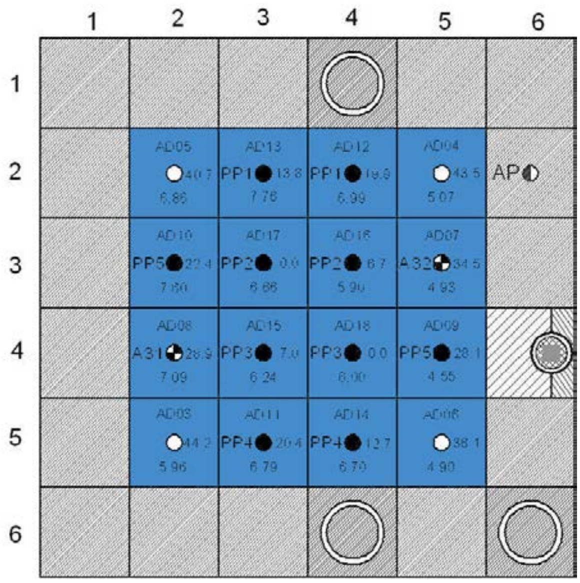

Fig. 5. Fuel burnup and FAs power for the equilibrium loading with LEU (BOEC)

For four mixed loadings # HLEU01, # HLEU03, # HLEU05, # HLEU07, for the first loading with LEU and for the equilibrium loading of the core with LEU the main neutronic and reactivity characteristics were calculated.

4. The main results of calculation analysis of possible accidents consequences

At normal operation of the IR-8 reactor the radiation doses to the population due to external and internal exposures, even at the nearest border of the NRC "KI" vicinity will not exceed the value of 10^{-2} mSv/yr, which is much lower than allowable value equal to 1 mSv/yr.

The calculation results of radiation consequences of the accident showed that total annual individual dose to the population living in the vicinity of IR-8 reactor will not exceed the value of 0.32 mSv/yr, which is lower than allowable value equal to 1 mSv/yr. Therefore no additional protective actions during conversion from HEU to LEU are required.

The analysis of radiation consequences of the accidents was performed for initial loading of the core with HEU, four mixed loadings HEU-LEU, the first loading with LEU and for the equilibrium loading with LEU.

Analysis of the consequences of pre-emergency accidents

Analysis of the consequences of RIA type accidents showed that in the considered cases calculated coolant temperature reaches the value not above 106 °C, which does not exceed the saturation temperature equal to 119 °C at pressure of $1.9 \cdot 10^5$ Pa in the core. The cladding surface temperature can reach up to ~146 °C during the first second of the transient (in case with fall of fuel assembly into the core). The departure of nucleate boiling (DNB) will not occur. Since the Fuel Element (FE) temperature increases during a very short period of time and the temperature is much lower than the clad damage temperature, FE cladding damage will not occur.

Except the case of FA drop at the reloading, the results for mixed HEU-LEU loadings and loadings with LEU are very similar to the results for loading with HEU.

Analysis of the consequences of design basis accidents (DBA)

The results of calculation of the fuel element (with HEU or LEU) cladding damage have shown that radiation consequences of cladding damage will be visible on the background activity of fission products into the coolant from surface contamination by uranium of fuel claddings with a total area of more than 20 cm^2. The sensitivity of the methods to control the activity levels of fission products in the coolant and in the ventilated air from under-deck space is sufficient to determine the presence of leaking fuel elements. In case of exceeding of the allowable activity level in the primary circuit water, the reactor automatically shuts down and cladding damage monitoring is carried out.

The results of calculation of the radiation consequences of the accident have shown that in case with the blockage of coolant flow through a FA with LEU fuel doses to the personnel of the NRC “KI” and the population in the vicinity of the reactor IR-8, will be essentially the same as in a similar accident with HEU. For both HEU and LEU cases the total annual individual dose to the population living in the vicinity of the reactor IR-8 is lower than allowable value equal to 1 mSv/yr.

A calculation analysis of the IR-8 primary coolant flow reduction has shown that DNB in this case will not occur. A short time boiling on the surface of the fuel elements is possible, since the surface temperature of the fuel elements in the "hot spot" within the first seconds is close to the surface temperature of the ONB. Consequently, in the case with the primary coolant flow reduction, even without insertion of the control rods into the core, damage of fuel element will not occur.

Analysis of the consequences of beyond design basis accidents (BDBA)

The list of considered initiating accidents for the analysis of the consequences of beyond design basis accidents includes:

1. Spontaneous withdrawal of the automatic regulation rod (AP) with subsequent loss of flow and failure of the safety rods.
2. Full instantaneous rupture of the primary pipes.
3. Full instantaneous pump-down of the fuel storage.
4. Full instantaneous rupture of a beam tube (failure of beam tube).

Spontaneous withdrawal of the automatic regulation rod (AP) with subsequent loss of flow and failure of the safety rods

Calculation analysis of the spontaneous withdrawal of the automatic regulation rod (AP) with subsequent loss of flow and failure of the safety rods was performed using the PARET/ANL code. The results showed that the maximum power is achieved after 10.0 s, i. e. at the time of total withdrawal of the AP rod and then the power decreases under the influence of temperature reactivity effects until scram occurs on decrease of flow (note that it was assumed that scram signals on power level and period fail). Maximum coolant and fuel cladding temperatures are achieved at the beginning of insertion of the control rods. Analysis of temperature conditions showed that for all the considered loadings of IR-8 brief surface boiling in the most heat-stressed channel is possible. Departure of nucleate boiling (DNB) will not occur, because the margin to critical heat flux will be at least ~ 3.1 and cladding damage will not happen.

Full instantaneous rupture of the primary pipes – Loss of Coolant Accident (LOCA)

The cooling regime for an accident with rupture of a pressure pipe as the worst case was considered. The water level in the pool is reduced ~ 4.4 minutes to the mark 1.7 m above the top of the core. During this time the core is cooled by forced movement of draining water. After that, draining from the pool stopped because of disruption of the siphon drain and the core is cooled by natural circulation. Calculations for the natural circulation have been fulfilled using the BEREZA code.

There is no significant difference in obtained results for all loadings. Water temperature at the outlet of the most heat-stressed gap reaches the saturation level (104 °C) after ~9 hours, and 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 all core begins. Water temperature in the FA gaps reaches the 104 °C, and temperature of fuel elements is almost the same.

For confirmation of the results obtained with the BEREZA code the same calculations were performed using the code system ATHLET for starting loading of the core with HEU and equilibrium loading with LEU. Similar results were obtained for both codes [14].

Full instantaneous pump-down of the fuel storage

Calculation analysis of the criticality of fuel during storage of spent FA (SFA) with HEU and LEU in the case of the storage unwatering was performed using the MCU-PTR code. It was obtained that in case with unwatering of SFA storage pool with HEU and LEU the condition K_{eff} <0.95 is obtained. Based on the results of calculations and taking into account the regulations it can be concluded that the safety of SFA with HEU and LEU fuel storing in the SFA storage pool of the IR-8 reactor is maintained.

Calculation of the radiation situation in the reactor hall by the scattered radiation in case with unwatering of SFA storage pool was performed using MicroSkyshine code -version 2 [10]. The results of estimation of radiation situation in the reactor hall near the storage of spent FA show that for the accident with unwatering of the storage pool the dose due to scattered radiation from spent fuel assemblies with HEU or LEU fuel does not exceed ~ 55 mSv/h.

Full instantaneous rupture of a beam tube (Failure of beam tube)

It is considered the worst case for IR-8 reactor: the formation of gap around the full perimeter of the channel with diameter of 100 mm or 150 mm, which can be caused, for example, by the destruction (separation) of beam tube end. Calculation analysis of consequences for the destruction of the beam tube end is performed using the ATHLET code. This work is in progress.

5. Conclusions

From the performed transient analysis (RIA and LOFA) it follows that FE damage when using the LEU fuel in all the considered situations will not occur.

Storage, as under normal conditions, and in the case of SFAs both with HEU and LEU fuel draining, in the storage pool of SFA storage and in cells of temporary storage of the IR-8 reactor is safe for all postulated conditions.

The results of radiation consequences analysis of the IR-8 reactor during normal operation and considered accidents showed that conversion from HEU to LEU fuel will not lead to decrease of the radiation safety of the reactor and therefore it is not required to perform any protective actions.

Currently, work on the safety assessment of the reactor IR-8 during the conversion from HEU to LEU fuel continues for beyond design basis accidents (BDBA).

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