TRANSACTIONS

Poster Session 1

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Research Reactor Fuel Management

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Poster Session 1

Fuel Development, Qualification, Fabrication and Licensing
THE RESULTS OF COMPUTATIONAL STUDY OF STRESSED-STRAINED STATE KINETICS IN A RESEARCH REACTOR FUEL ELEMENT WITH (U-9%Mo)-Al DISPERSIVE FUEL IRRADIATED IN THE IVV-2M REACTOR

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ABSTRACT

The results of the computational study of the stressed-strained state (SSS) kinetics in the research reactor (RR) dispersive fuel element with (U-9%Mo)-Al irradiated in the IVV-2M reactor have been analysed. The effects of the birth defects of the “delamination of fuel composition” type on the SSS and fuel element form changes resulted from the joint action of the fuel being swollen and the gaseous fission products being released from the fuel into the defect area have been assessed.

1. Introduction

The paper presents the data relating to the conditions of the first in-reactor testing of dispersive fuel elements with (U-9%Mo)-Al and their post-irradiation examinations that have been carried out at the IVV-2M reactor (Zarechniy, Russia) [1]. The data obtained on the test conditions and the results of the post-irradiation examinations enable to assess the kinetics of the stressed-strained state (SSS) of these fuel elements in the course of irradiation and to forecast the possible form changes of a fuel element if it has got a birth defect of the “delamination of fuel composition” type. The birth defect of the “detachment of cladding from fuel composition” type has been considered in ref. [2].

2. The statement of the problem

Let us consider a research reactor (RR) fuel element, Figure 1, whose dispersive type fuel particles (of the U-9%Mo alloy in aluminium matrix) are enclosed into the cladding made of the aluminium-based SAV-1 alloy. The fuel composition is rigidly bonded with the fuel cladding through the aluminium matrix. The presence of this rigid bond between the fuel and the cladding ensures the loading of claddings from the swelling fuel with a given strain rate.

The swelling of the fuel composition, \( S_f \), was determined according to the data of the post-reactor measurements of the fuel element volume increase (\( \Delta V \)): 

\[ S_f = \frac{\Delta V}{V} \]

where \( V \) is the initial volume of the fuel element. The swelling rate of the fuel composition was
determined as \( \hat{S}_m = \frac{S_m}{\tau_p} \), where \( \tau_p \) – is the duration of the fuel element irradiation.

The design data presented in Ref. [2] show that it is the birth defects of the “fuel composition delamination” and “detachment of cladding from fuel composition” types that can have the most substantial effects upon the fuel element performance.

Cracks and delaminations in the fuel composition are the places where the gaseous fission products (GFP) releasing according to the mechanism of the direct emission out of the fuel particles can accumulate. This GFP accumulation inside the cavities like those leads to the growth of pressure and the local form changes in the “fuel-cladding” system. By way of illustration of this, the comparative computations of SSS and form changes in a RR fuel element that had got a birth defect of the “fuel composition delamination” as well as in an identical element having no defect were carried out with the input data as follows:

Dimensions of a fuel element:
- width across flats, \( S \) 35,8 mm
- thickness of cladding, \( \delta \) 0,375 mm
- thickness of the dispersive fuel composition, \( \Delta \) 0,6 mm

Fuel element materials:
- cladding material aluminium-based alloy SAV-1
- matrix material aluminium powder
- fuel particles U-9%Mo alloy

Average volumetric content of (U-9%Mo) particles in the fuel composition 32 %

Physicomechanical properties of the fuel element materials assumed for the computations:
- thermal conductivity of the cladding and matrix materials, \( \lambda_1 \) 230 W\( \cdot \)m\(^{-1}\)\( \cdot \)K\(^{-1}\)
- thermal conductivity of the (U-9%Mo) alloy, \( \lambda_2 \) 16,7 W\( \cdot \)m\(^{-1}\)\( \cdot \)K\(^{-1}\)
- Young modulus of the cladding and matrix materials, \( E \) 10\(^5\) MPa
- Poisson ratio for the cladding and matrix materials, \( \mu \) 0,3

The data on the creep of the SAV-1 alloy under the in-reactor irradiation conditions is presented in Fig. 2 [3].

The loading factors of loading are:
- volumetric energy release inside the fuel composition, \( q_v \) 3 W\( \cdot \)mm\(^3\)
- coolant temperature, \( T_c \) 60°C
- rate of fuel composition swelling, \( \hat{S}_f \) 9,3\( \cdot \)10\(^{-4}\)\%\( \cdot \)h\(^{-1}\)

For the purpose of computing, a defect of “lengthwise delamination of fuel composition” type, 10 mm wide, 50 mm long, situated in the centre of the fuel layer at one of the sides of the fuel element and symmetric about its longitudinal axis had been adopted. The fuel element cladding was considered to be loaded both by the pressure of the fuel composition being swollen and by that of the GFP having been released from the fuel composition into the delamination space.

The duration of fuel element operation, \( \tau_p \), had been adopted to be of 3650 hours.

3. The volumetric assessment of the GFP releasing into the cavities formed by defects inside the fuel

The main contribution to the total volume of the GFPs that will have passed through the surface of the delamination cavity during the first ~10 days of operation will be made by the stable isotopes of Kr and Xe. Their volume is proportional with the
irradiation time while the radioactive isotopes of Kr and Xe attain rather promptly (in ~10 or less days for the majority of isotopes) the state of equilibrium between the formation and decay of nuclei. In this case the amount of nuclei (and consequently the volume of GFPs) does not depend any more on the duration of irradiation and remains invariable.

As was shown in Ref. [4], one fission of U-235 forms eventually ~ 0,26 of a stable (Kr+Xe) isotope nucleus. The number (\( M \)) of U-235 nuclear fission acts inside the “outgassing” volume (\( V_{ov} \)) of a fuel particle per second can be determined on the ground that 1 W of power corresponds to \( 3,2 \times 10^{10} \) fiss/s: \( M = 3,2 \times 10^{10} N_{ov} \ W^{-1} s^{-1} \), where \( N_{ov} \) is the heat power generated in the “outgassing” volume fraction, expressed in watts (W) and described by the relationship: \( N_{ov} = q_v \cdot V_{ov} \) (W), where \( q_v \) is the volumetric heat power generated in (U-Mo) particles, W \( \cdot \) mm\(^{-3} \).

The “outgassing” volume fracture, \( V_{ov} \), can be determined as \( V_{ov} = S_f \cdot \delta_1 \), where \( S_f \) is the area of the open surface of (U-9 %Mo) fuel particles, and \( \delta_1 \) – the average free path of fission fragments: \( \delta_1 \approx 0,01 \) mm.

The number of stable isotope nuclei formed per second and released into the defect-simulating cavity (\( N_{st.n} \)) is:

\[
N_{st.n} = M \cdot 0,26 \ \text{nuc} \cdot s^{-1}.
\]

The amount of moles of an isotope released into the defect cavity per second (\( \sum n_i \)) is:

\[
\sum n_i = N_{st.n}/A_v \ \text{mole} \cdot s^{-1},
\]

where \( A_v \) is the Avogadro number, \( 6,02 \times 10^{23} \).

The total volume of the isotope moles released into the defect cavity per second is:

\[
V \cdot \sum n_i = 22,4 \times 10^6 \ \text{nm} \cdot \text{m}^{-3} \cdot \text{s}^{-1}.
\]

So the volume of the moles released into the defect cavity during the operation time \( \tau \) is:

\[
V \cdot \sum n_i = 22,4 \times 10^6 \cdot \tau \ \text{nm} \cdot \text{m}^{-3} \cdot \text{s}^{-1}.
\]

Then the volume of stable (Kr+Xe) isotopes (\( V_{stab} \)) released into the defect-simulating cavity during the whole operational time \( \tau \) is:

\[
V_{stab}(\tau) = V \cdot \sum n_i \cdot \tau \ \text{nm}^3.
\]

The computation of the GFPs release from the fission areas was carried out with the following assumptions:
- the area of the “outgassing” surface of fuel particles (\( S_f \)) was assumed to be equal to 0.32 of the delamination surface area and to remain constant during the whole operation time;
- the volumetric generation of heat in (U-9%Mo) fuel particles, \( q_v \), was assumed to equal 9 W \( \cdot \) mm\(^{-3} \).

The rate of the GFPs release into the defect area under these assumptions was:

\[
V_{stab} \approx 3,35 \times 10^{-2} \ \text{nm}^3 \cdot \text{h}^{-1}.
\]

4. The results of computing the stressed-strained state and the form change of RR fuel elements without any defect and with the birth defect of the “delamination of fuel composition” type

The results of computing the SSS in a fuel element without any defect are presented in Figures 3, 4 and 5.

As can be seen in the Fig. 3, the strained state in the fuel element cladding becomes stabilizes at a
level (~105 MPa) where the rates of fuel composition swelling and cladding material creep become equalized.

At the end of the fuel element operation the maximum creep strain in its outer cladding can attain the value of 0.32% (Figure 4). The stresses in the fuel claddings throughout its entire operation time are tensile, but the fuel composition remains compressed (Fig. 5).

The results of the SSS and form change computations for the fuel element with the “fuel composition delamination” birth defect are presented in Figures 6, 7, 9 and 10.

The displacements of the upper and lower sides of the defective fuel claddings are shown in Figures 6 and 7. The axial length-dependent displacements of the fuel element sides at the end of the operation time being characteristic for the fuel elements with the defect under consideration can be seen in Figs. 6, 7. The principal attention is to be paid here to the displacement of the upper side above the defect as it is more significant than that of the lower one. The change in the field of stresses $\sigma_x$ (MPa) in the course of the operation is presented in Fig. 8.

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**Fig. 5.** The field of stress intensity (MPa) in the fuel element without any defect

**Fig. 6.** The plot of Y-displacements along the outer (1) and inner (2) sides of the defective fuel element.

**Fig. 7.** The field of Y-displacements (mm) in the defective fuel element

**Fig. 8.** The field of stresses $\sigma_x$ (MPa) in the non-defective fuel element

**Fig. 9.** The plot of stress intensity (MPa) along the outer side of cladding of the defective fuel element

**Fig. 10.** The EOL strain intensity along the outer side of cladding of the defective fuel element at the end of its operation
The plots of the stress and strain intensities along the outer side of the cladding at the end of the fuel element operation can be seen in Figs. 9 and 10. As can be seen in Fig. 10, the localization of the creep strain is going on in the fuel element cladding above the centre of the fuel delamination. The behaviour of the change in the GFPs pressure in the birth defect area during the irradiation is shown in Fig. 11.

5. Conclusion

The results of computing the stressed-strained state as well as the form change in the research reactor fuel element having the birth defect of the “fuel composition delamination” type have shown that:

- due to the presence of a defect of the said type, the strain in the fuel cladding becomes localized above the delamination centre, which can result in the loss of the sealing ability of the cladding;
- the local form change (“bulging”) of the fuel elements with defects of that type can lead to the reduction of the flow area between the fuel elements to the degree regarded as unallowable from the points of view of hydraulics and heat-sink cooling.

6. References


THERMO-MECHANICAL MODELLING OF U-Mo FUELS WITH MAIA

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ABSTRACT

CEA has developed in close collaboration with ANL a 2D thermo-mechanical code, called MAIA, for modelling the behaviour of U-Mo dispersion fuel. MAIA uses a finite element method for the resolution of the thermal and mechanical problems. The physical models (U-Mo/Al interaction layer growth, PF swelling) are the ones used in the ANL code, PLATE. Several correlations are available for the modelling of the growth of the oxide layer on the cladding. The thermal modelling takes into account the volume fraction of each meat constituent that evolves with time due to the growth of interaction layer, densification and fission product swelling in the U-Mo particles and in the interaction layer. The thermal properties of the meat are then homogenized. The mechanical modelling can use plasticity in the meat and creep in the Al alloy cladding. MAIA is used to calculate temperatures, stresses and deformations within a U-Mo plate. It also calculates the oxide layer thickness of the cladding, the layer thickness of interaction product between U-Mo and the aluminium matrix and the fraction volume of each constituent of the meat during time. The results of the code are compared with post irradiation examinations performed on FUTURE, an irradiation of the French U-Mo Group.

1. Introduction

MAIA is a 2D thermo-mechanical code using a Finite Element Method (FEM) dedicated to U-Mo/Al dispersion fuel plates. It is based on CAST3M, a FEM code developed by CEA. The geometry is a 2D plate (plane or curved). The meat is treated as a homogeneous material for the thermal and mechanical resolution but the evolution of its composition is calculated throughout the irradiation taking into account the disappearing of as-fabricated porosities, the swelling of the fuel due to fission products and the interaction between the U-Mo particles and the Al matrix. MAIA’s development has been made in close collaboration with Argonne National Laboratory (ANL).

2. MAIA code modelling

2.1 Meat thermal conductivity homogenisation

For the thermal FEM resolution, the meat is considered as a homogeneous material. The thermal conductivity of the meat is homogenised in several steps. At first, its degradation due to the gaseous swelling in U-Mo is taken into account [1]:

\[ k_g = k_{UMo} \exp(-2.14 S_{gas}) \]

where \( k_g \) is the thermal conductivity of U-Mo with gaseous fission product bubbles, \( k_{UMo} \) is the thermal conductivity of U-Mo and \( S_{gas} \) is the U-Mo swelling due to gaseous fission products. Then, the U-Mo particles are supposed to be spherical with an external uniform layer of interaction. Under these
conditions, the equivalent conductivity of the U-Mo particles with the interaction layer can be analytically calculated:

\[
k_f = \left(1 - \frac{e}{r_k} + \frac{e}{k_{\text{interaction}}} \right)^{-1}
\]

where \(k_f\) is the effective thermal conductivity of the fuel (with the interaction phase), \(e\) is the thickness of the interaction phase, \(r\) the radius of the U-Mo particle and \(k_{\text{interaction}}\) the thermal conductivity of the interaction phase. For the homogenisation of the Al matrix with these inclusions, an auto-coherent law [2] is used:

\[
k_{100} = -k_f + 3V_f k_f + 2k_m - 3V_f k_m + \frac{8k_f k_m + \left(k_f - 3V_f k_f - 2k_m + 3V_f k_m \right)^2}{4}
\]

where \(k_{100}\) is the thermal conductivity of the fully dense meat, \(k_m\) is the thermal conductivity of the matrix and \(V_f\) is the volume fraction of the fissile particles (U-Mo and the interaction phase).

And finally, a correction is applied due to the porosity of the meat:

\[
k_{\text{meat}} = k_{100} \cdot \exp(-2.14 P)
\]

where \(k_{\text{meat}}\) is the thermal conductivity of the meat and \(P\) is the porosity.

### 2.2 Interaction layer

The U-Mo particles react with the aluminium matrix to create interaction compounds. The code predicts the thickness of this interaction layer with the correlation developed by ANL [5]. Then, the volume of this product is calculated assuming that U-Mo particles are spherical. With a mass balance, the volumes of each constituent of the meat are calculated during each time step of the calculation. The user must specify the stoichiometry of the interaction compound (U-Mo)\(\text{Al}_x\).

### 2.3 Fission product swelling

Fission product swelling is calculated using also a correlation developed by ANL [5]. In the U-Mo, this swelling is made of a solid part and a gaseous part. In the interaction product, a fission product swelling is also calculated.

### 2.4 Fabrication porosity

The fabrication porosity of the meat is supposed to accommodate the swelling due to the interaction product creation and the fission products as long as it has not completely disappeared.

### 2.5 Cladding Oxidation

The oxidation of the cladding is a strong barrier for the thermal flux evacuation due to the low thermal conductivity of the boehmite compared with the one of an aluminium alloy. Several correlations are available such as ORNL models ([3] and [4]) to evaluate the growth of this layer during irradiation.

### 2.6 Mechanical calculation

For the mechanical calculation, the user can either choose an elastic behaviour or a viscoplastic one (plasticity in the meat and creeping in the cladding). The swelling calculated in the former models (interaction product creation, fission products and porosity depletion) is supposed to be isotropic and is transformed in a strain imposed for the mechanical calculation.

### 3. Validation

The validation of MAIA is under progress. The first part consists of analytical tests to confirm the good behaviour of the models with their specifications. The second part consists of the comparison of experimental data with MAIA results. MAIA has already been used to compare the results of the code
with post-irradiation measurements made on IRIS 1 [6]. In this paper, the code is used to calculate FUTURE [7], an other irradiation of the French U-Mo Group, whose first post irradiation examinations are now available. The validation will continue this year on other irradiations as soon as other data and post-irradiation examinations will be available.

The extra volume expansion observed in a part of the FUTURE plate is of course not modelled in MAIA as this phenomenon is not yet understood. The code does not pretend to predict the irradiation behaviour of the plate but to be a useful tool to help for the interpretation of these irradiations. The code is also very useful for parametric studies.

4. FUTURE

4.1 Irradiation conditions

The “FUTURE” irradiation is a part of the French U-Mo Group program. Two low enriched uranium fuel plates have been irradiated two cycles (40 full power days) in the FUTURE irradiation rig of BR2 [7]. These plates consist of U-7wt% Mo atomised powder with low enriched uranium (~20% $^{235}$U) and has a density loading of 8.5 g U.cm$^{-3}$. The cladding of the fuel plates is an AG3-NET alloy.

<table>
<thead>
<tr>
<th>Width (mm)</th>
<th>Plate</th>
<th>Fuel meat</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thickness (mm)</td>
<td>64.5</td>
<td>49.8</td>
</tr>
<tr>
<td>Porosity</td>
<td>1.21</td>
<td>0.53</td>
</tr>
</tbody>
</table>

Table 1: As-fabricated geometry of U-Mo modelled fuel plate (U7MTBR06)

4.2 Hypotheses

The MAIA calculation of FUTURE has been made at the maximum flux plane (MFP) (Cf. Fig 2 - 3).

Due to the first post irradiation examinations, the interaction product is assumed to be a compound of global composition “(U-Mo)Al$_4$”. For IRIS 1, in accordance with the post irradiation examinations [6], it was assumed to be “(U-Mo)Al$_7$”. The oxide layer growth is imposed with a linear law with a final value of 33 µm (the maximum oxide thickness measured after irradiation) because the other available correlations underestimate this growth. Along the width of the plate, a correction is applied to the history of power density of the hottest point (cf. Fig 1) to take the power profile into account [8].

For the thermal calculation, the coolant temperature is imposed at 40°C. A convection condition is applied between the coolant and the cladding with a heat coefficient transfer of 39500 W.cm$^{-2}$ to get a maximum external cladding temperature of 126°C at the beginning of irradiation. The mechanical calculation has been made with an elastic behaviour of the meat and the cladding and with the hypothesis of plane strains.

4.3 Results

The main results are summarized in the figures 4 to 7 and in table 2.
The maximum temperature of the meat evolves from 135°C at the beginning of the irradiation to 215°C after 40 days (cf. Fig 4 - 5). At the cladding-boehmite interface, the temperature evolves from 126°C to 150°C. MAIA allows localizing the hottest part of the plate that does not exactly fit with the maximum power due to side effect in the cooling.

The post irradiation image analyses show a good agreement with the calculated Al matrix final volume fraction but a slight underestimation of the interaction product and a slight over-estimation of the U-Mo particles final fraction volume (cf. Fig 7). However, the interaction layer thickness calculated by MAIA is in a good agreement with the post irradiation measurements (cf. Table 2).

The total strain of Fig. 6 is the strain imposed for the mechanical calculation in each direction of the plate. It is the sum of the strains produced by the interaction product creation, the fission products swelling and the porosity depletion. The creation of the interaction product is the first cause of swelling in front of fission products.

<table>
<thead>
<tr>
<th>Interaction layer thickness (µm)</th>
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<tbody>
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<td>Post irradiation examinations</td>
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<td>MAIA</td>
</tr>
</tbody>
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Table 2 Interaction layer thickness at the end of irradiation in the centre of the meat (O)

5. Discussion
6. Conclusion

French irradiations IRIS 1 and FUTURE are the first results of plate irradiations calculated with MAIA. These results are globally in rather good agreement with the examinations but some improvements in the modelling are still needed to reduce the differences between calculations and measurements. The validation has to be continued as soon as other post irradiation examinations will be available.

The code does not pretend to predict the irradiation behaviour of the plate but to be a useful tool to help for the interpretation of these irradiations.

7. References

MICROSTRUCTURE OF U$_3$Si$_2$ FUEL PLATES SUBMITTED TO A HIGH HEAT FLUX.

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ABSTRACT

In order to gain insight on the performance limits of U$_3$Si$_2$ fuel with aluminum cladding, fuel plates with a fissile material density of 5.1 and 6.1 gU/cm$^3$ have been irradiated in the BR2 reactor of SCK•CEN in Mol. The plates were intended to be subjected to severe conditions corresponding to a maximum mean hot plane heat flux of 520 W/cm$^2$, leading to a cladding surface temperature of 180 - 200°C and fuel temperatures of 220 - 240°C. The irradiation program was stopped after the second cycle based on the visual inspection and wet sipping tests of the elements, correspondingly showing degradations on the outer aluminum surfaces of the U$_3$Si$_2$ plates and the release of fission products. The maximum fuel burnup was 29% and 25% $^{235}$U respectively.

In a post-irradiation examination (PIE) program the microstructural causes for this degradation are analysed. It is found that the failure of the plates is entirely related to the corrosion of the aluminum cladding, which has caused temperatures to rise well beyond the calculated values (>300 ºC for the cladding and >400 ºC for the fuel). In all stages, the fuel grains have retained their integrity and, apart from the formation of an interaction phase with the aluminum matrix, they do not demonstrate deleterious changes in their physical properties.

1. Introduction

The design of the French Réacteur Jules Horowitz (RJH) is based on low enriched uranium (LEU) fuel ($\leq 20\%$ $^{235}$U/$U_{total}$). In LEU fuel the reduction of the enrichment or fissile material is compensated by a higher uranium loading density [1,2]. As manager of the RJH project, the Commissariat à l'énergie atomique (CEA) has a vital interest in the qualification of LEU fuels. Such a program requires irradiation tests to demonstrate the ability of the fuel to withstand research reactor operating limits (i.e. high power densities) up to economically viable burn-up values. Among the fuels considered is the U$_3$Si$_2$ fuel dispersed in an aluminum matrix [3]. U$_3$Si$_2$ has been found to perform well under irradiation tests, even with uranium densities up to 5.0 g/cm$^3$ [4].

To study the performance limits of this fuel type, U$_3$Si$_2$ fuel plates have been irradiated in the BR2 reactor of SCK•CEN in Mol in 2 fuel elements under conditions corresponding to the upper boundary limit of the regime at which the RJH reactor is planned to operate. After irradiation, visual inspection of the fuel elements showed important degradations of the outer surfaces of the U$_3$Si$_2$ plates, including visible swelling. Wet sipping test demonstrated loss of integrity for one of the fuel plates. Several samples have been cut from the fuel plates and investigated in a post-irradiation campaign. In this article, the results of the microstructural examination of the fuel plate with the highest U loading (6.1 gU/cm$^3$) are presented.
2. Experimental

2.1. Fuel plates and irradiation history

The fissile material or meat of the fuel plate consists of U$_3$Si$_2$ grains dispersed in a pure aluminum matrix with a mass ratio (U$_3$Si$_2$/Al) of 6.6. The enrichment of the uranium amounts to 35% $^{235}\text{U}/\text{U}_{\text{total}}$. The cladding of the fuel elements is made of AG3-NET Al-Mg alloy (2.5 – 3 wt% Mg), which corresponds to a nuclear grade Al5754 alloy.

The fuel plate was incorporated in the BR2 reactor during two irradiation cycles. Thermal hydraulic calculations showed that the mean heat flux in the axial hot plane would approach the requested 520 W/cm$^2$. From the temperature calculations it was found that the external cladding surface temperature would reach around 160°C. However, the azimuthal distribution of the heat flux in the axial hot plane of the fuel element shows a peak of 550 W/cm$^2$ in the first cycle.

After the second cycle (prematurely terminated), wet sipping tests showed the release of fission products. Visual inspection revealed degradations of the outer surface and swelling of the plate. Based on this information, the irradiation program was stopped before the mean burn-up target had been reached. The U$_3$Si$_2$ plate showed a maximum burn-up of $1.29 \times 10^{21}$ fissions/cm$^3$ (25% $^{235}\text{U}$).

2.2 Post-irradiation examination

Visual examination at the reactor site hot cell revealed that the deformed zone on the fuel plate consists of 3 concentric half-circular stretched areas.

The result of thickness measurements shows that the swelling is located at the left side of the plate, at a position azimuthally located near the stiffener, where the fission density shows a maximum. Given the as fabricated plate thickness 1.28 ± 0.01 mm, the swelling amounts up to about 0.33 mm.

Based on this profilometry, several samples were cut from the fuel plate. The samples are embedded in an epoxy resin in such a way that the complete section of the fuel (meat and cladding) could be observed. The samples are polished with SiC paper of successively finer grain size, finishing on cloth with diamond paste of 3 µm and 1 µm.

3. Results

3.1. Cladding corrosion

The ceramographic observation of the fuel section that has been submitted to a relatively low heat flux (approximately 450 W/cm$^2$), shows no visible degradation of the cladding or the fuel (Figure 1a). The detailed image of the fuel (Figure 1b) reveals an interaction phase (average thickness of 2 µm), between the U$_3$Si$_2$ particles (dark gray) and the aluminum matrix (white). Such an interaction layer is regularly observed for this type of fuel and has been identified as an U(Al,Si)$_3$ phase [5,6].

With increasing heat flux (approximately 500 W/cm$^2$) the oxidation of the cladding at the waterside surfaces becomes visible (Figure 1c). A dense oxide layer has developed, followed by a layer with a granular aspect which is most probably caused by pitting corrosion. The fuel is relatively intact, with the exception of several cracks passing through the meat (Figure 1d), which should be attributed to the thermal cycling of the fuel. The interaction between the U$_3$Si$_2$ particles and the aluminum matrix has progressed slightly further (average thickness of interaction layer equals 3 µm) but most of the particles are still separated by pure aluminum.

The image of the fuel plate section that has been submitted to 550 W/cm$^2$ (Figure 1e) shows that the corrosion of the cladding has proceeded up to the meat. A clear increase in plate thickness of approximately 0.3 mm can be measured, which is in good agreement with the profilometry (0.33 mm). As also visible in Figure 1 a,c,e, the swelling of the fuel plate should be attributed to the corrosion of the cladding and not to an increase of the meat thickness. In the meat, all the aluminum matrix is consumed by the fuel and large voids are present (Figure 1f). The average thickness of the interaction layer has increased to 5 µm.
Figure 1 Image of the fuel plate over the fuel width and a micrograph of the meat obtained from the section of the fuel plate that has been submitted to \( \approx 450 \text{ W/cm}^2 \) (a,b), \( \approx 500 \text{ W/cm}^2 \) (c,d) and \( \approx 550 \text{ W/cm}^2 \) (e,f).

X-ray maps of the fuel plate sample submitted to 550 W/cm\(^2\) are recorded to reveal the distribution of Al, Mg, Si and O in the corroded cladding. From Figure 2 it is clearly seen that the corroded cladding consists of several layers. A first layer (Figure 2L1) is a dense oxide layer, without precipitates but containing some Mg. Below this layer, a pitting corrosion layer with characteristic corrosion pits can be seen (Figure 2L2). Underneath, a network of grain boundaries is clearly delineated in the Mg and O X-ray maps (Figure 2L3). It is also noticed that in this layer the grain interiors contain oxidised Mg. In layer 4 (Figure 2L4) only the grain boundaries have been affected and the grain interior shows clustering of Mg but as yet unoxidised. At this and the next position (Figure 2L5), one can observe that Mg and Si are associated and form precipitates (Mg\(_2\)Si). Close to the meat (Figure 2L5), a layer of relatively intact cladding is still present. At the interface between cladding and the meat, oxidised Mg is observed (Figure 2L6).

Figure 2 Al K\( \beta \), Mg K\( \alpha \), Si K\( \alpha \) and O K\( \alpha \) X-ray maps of a third of the fuel plate section, from the corroded cladding surface to the meat. The various layers (L1 to L6) are described in the text.
3.2. Fuel behavior

The U, Al and Si X-ray maps in the meat of the sample that has been submitted to a heat flux of 550 W/cm$^2$, indicate that the consumption of the aluminum matrix is complete and that a U-Si-Al interaction phase is present. Voids have formed between the fuel grains and the oxygen map (Figure 2) shows the oxidation of the inner surfaces of these porosities. This indicates that water has intruded in the fuel causing the oxidation of the surface of the voids.

From the X-ray micrographs (Figure 3) it is seen that the fission product concentration is lower in the U-Si-Al mixed phase, but at the interface with the Al cladding, a clear rise in fission product concentration can be observed. This is reflected in the linescan data (Figure 4) where a steep local rise of fission product concentration can be seen at this (boundary between zone 2 and 3 in Figure 4). Inside the grain (zone 1), the concentrations are nearly constant. In between the mixing phase of this grain and the adjacent grain (zone 2) the fission product signals rise again, producing the halo-effect. Furthermore from Figure 4 it is seen that the ratio of the atomic percentages of U and Si remains constant at around 1.5, even inside the U-Si-Al mixing phase.

![Figure 3: Local X-ray map (respectively of Al Kβ, U Mα, Si Ka, Xe La, Nd La) of a U$_3$Si$_2$ particle in the Al matrix. The secondary electron image (SE) indicates the position at which a semi-quantitative linescan is measured.](image)

![Figure 4: Semi quantitative linescan covering the Al matrix and a U$_3$Si$_2$ particle defined in Figure 3. The linescan can be divided in different zones: zone 1 is inside the grain, zone 2 is the interaction layer and zone 3 is the Al matrix. It should be noted that the amounts of main fuel constituents and fission products are expressed as wt% while the lower graph represents the ratio between At% U and At% Si.](image)

4. Discussion

It seems obvious that the failure of the fuel plate is entirely related to the deterioration of the cladding and not to the degradation of the fuel. The microstructural analysis of samples submitted to different heat fluxes, shows the evolution of the fuel plate as the corrosion of the cladding progresses.

Taking into account the heat flux of 550 W/cm$^2$, it is estimated that at the spot close to the stiffener, the temperature at the surface of the plate must have reached values around 190 °C. At such high temperatures, general corrosion of the Al-Mg alloy will occur. The resulting corrosion layer, consisting of an alumina barrier layer on top of a porous boehmite (γ-AlO(OH)) bulk corrosion layer [7,9], has a lower thermal conductivity (2.25 W/mK) than AG3 NET (130 W/mK) [8]). The increase in temperature of the cladding following the buildup of this duplex layer, will accelerate the corrosion process. The X-ray map in Figure 2 reveals that an oxide layer of approximately 100 µm is formed. With a temperature increase of 1.5 to 2 degrees per µm of oxide, the estimated cladding temperatures would reach over 300 °C. At these high temperatures, progressing sensitisation of the aluminum-
magnesium cladding leads to the decoration of the grain boundaries with \( \text{Mg}_2\text{Al}_3 \) precipitates, making the cladding susceptible to grain boundary corrosion [9,10]. The meat temperature also rises and the formation of the U-Si-Al interaction zone will speed up, leading to a complete consumption of the Al matrix. This leads to a deterioration of the thermal conductivity of the meat, which will further increase the temperature. Eventually, as the cladding becomes completely corroded, water will be able to penetrate the fuel matrix and will start to oxidise the meat.

Regarding the fuel meat, the behavior of the \( \text{U}_3\text{Si}_2 \) fuel grains seems to have been exemplary. Except for the advanced interaction with the Al of the matrix, they have withstood extreme temperatures for this fuel type, without any deleterious changes in their physical properties. Considering the effect of the 100 \( \mu \text{m} \) thick oxide layer, the calculated centerline temperature in fact rises to 425°C.

Such extreme temperatures substantiate the observed extent of the interaction layer [11,12]. The composition of the interaction phase is reported to be \( \text{U(Al, Si)}_3 \). With the observed constant \( \text{U/Sl} \) ratio in the interaction phase (Figure 4), the formula would be close to \( \text{U}_3\text{Al}_7\text{Si}_2 \).

Concerning fission gas release, the fuel shows a stable behavior with formation of small dispersed fission gas bubbles in the interior of the grains at this stage of the irradiation. There is no evidence of important release of fission gases.

The fission product distributions show nearly constant concentrations in the grain interiors, except where fission gases have begun to precipitate in bubbles. Fission products that have been ejected out of the grains, have been swept up by the formation of the U-Si-Al interaction phase and have accumulated at the interface between this phase and the remaining aluminum or between two interaction phase fronts of adjacent grains.

The origin and development of the observed voids located in the former Al matrix, is not clear. Diffusion phenomena associated with the formation of the interaction phase (the so-called Kirkendall effect [13,14] owing to the diffusion of Al in \( \text{U}_3\text{Si}_2 \)) and the accumulation of fission products at the interaction phase fronts are probably responsible for the creation of the voids. Upon water intrusion, considerable oxidation and lixiviation of the remaining Al matrix could occur. Because the irradiation was stopped shortly after the cladding had failed, the oxidation has remained limited to the surface of the voids.

5. References

IN-PILE TESTS AND POST-REACTOR EXAMINATIONS OF FUEL ELEMENTS WITH URANIUM-MOLYBDENUM FUEL OF REDUCED ENRICHMENT

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ABSTRACT

The results of the in-pile tests and post-reactor examinations of dispersed fuel elements with fuel made of alloy U-9%Mo with 36% enrichment by $^{235}$U and uranium concentration of 5.4 g/cm$^3$ manufactured by extrusion technique have been presented. The fuel elements have been tested during 107 effective days at IVV-2M reactor to reach average burnup of 40%. The maximum heat flux density was 0.69 MW/m$^2$, while the maximum design temperature of fuel cladding has not exceeded 80°C. The post-reactor examinations have been conducted using such techniques as visual inspection, profilometry, volume measurements, gamma spectroscopy, metallography and X-ray structure analysis. The following aspects have been examined: state of fuel elements; changes in shape, size and volume of fuel elements; structure and phase composition of the oxide film; character and depth of corrosion damage to fuel cladding; composition and structure of the dispersed fuel core; phase composition and width of the interaction areas of uranium-molybdenum fuel and the aluminium matrix.

1. Introduction

In the frame of the Russian RERTR (Reduced Enrichment Research and Test Reactors) program [1] there have been designed three-layer hexagonal tubular-type fuel elements (FE) for IVV-2M reactor with fuel based on (U-9%Mo)+Al and envisaged for manufacturing by means of extrusion. As the initial stage of maturing such fuel, there have been produced the experimental fuel elements with fuel of 36% enrichment by $^{235}$U and uranium concentration of 5.4 g/cm$^3$.

The use of fuel with 36% enrichment in the experimental fuel elements seems more preferable (as compared to fuel with 19.7% enrichment) in terms of safety of the experiment and allows to build up more fission products in fuel under investigation at lower values of fast neutron fluence both in this fuel and in fuel used in the standard fuel elements. Fuel with the above-mentioned enrichments offers also an advantage of reducing almost two times the duration of the experiment. The fuel is composed of spherical particles of 63±160 µm which are distributed almost uniformly in the aluminium matrix. Fuel claddings are made of aluminium alloys SAV-1 and AMg-2. In the period from November 2001 till April 2002 the successful in-pile tests have been conducted at the research reactor IVV-2M. Two experimental fuel elements (of types 1 and 2 which differ in "turnkey" size) were included in the standard fuel assemblies so as to accumulate fission products to the amount equivalent to ~40% burnup for fuel with 19.7% enrichment.
This article presents the results of the in-pile tests and post-reactor examinations of the experimental fuel elements with fuel based on (U-9%Mo)+Al of 36% enrichment by $^{235}$U and uranium concentration of 5.4 g/cm$^3$.

2. In-pile tests

The experimental fuel elements have been tested in the reactor as the part of the standard fuel assemblies being placed in the core of IVV-2M reactor. During the in-pile tests the water chemistry of meets the standards. The test parameters are summarized in the table and in Fig. 1.

Table. The brief characteristics of the in-pile tests of the experimental fuel elements

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Type 1</th>
<th>Type 2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel element power (average/maximum), kW</td>
<td>60.4 / 77.6</td>
<td>71.3 / 91.6</td>
</tr>
<tr>
<td>$q$ (average/maximum), MW/m$^2$</td>
<td>0.53 / 0.68</td>
<td>0.54 / 0.69</td>
</tr>
<tr>
<td>$q_l$ (average/maximum), kW/m</td>
<td>120.9 / 155.3</td>
<td>142.7 / 183.2</td>
</tr>
<tr>
<td>$q_V$ (average/maximum), kW/cm$^3$</td>
<td>2.29 / 2.94</td>
<td>2.31 / 2.96</td>
</tr>
<tr>
<td>Coolant temperature at inlet to FA, °C</td>
<td>31-34</td>
<td></td>
</tr>
<tr>
<td>Maximum wall temperature, °C</td>
<td>80</td>
<td>74</td>
</tr>
<tr>
<td>Duration of fuel life, eff. days</td>
<td>107</td>
<td></td>
</tr>
</tbody>
</table>

Fig. 1. Distribution of neutron flux density ($a$) and temperature ($b$) of coolant $T_w$, internal $T_{in}$ and external $T_{ex}$ fuel claddings and heat flux density $q$ along the fuel element of type 2:
Fuel cladding integrity monitoring of the experimental fuel assemblies (FA) have been detected during the in-pile tests and post-reactor cooling in the FA storage well by means of the standard fuel cladding failure detection system (FCFD) provided at IVV-2M reactor. The following parameters of FCFD have been monitored: counting rates of delayed neutrons in FA with the experimental fuel elements coolant outlet, concentration of $^{235}\text{U}$, the specific activities of inert radioactive gases and solid fission products in the primary coolant, $\beta$-activity of the primary coolants in the reactor and in FA storage well.

There were detected no significant deviations in terms of delayed neutron counting rates, concentration of solid fission products and $^{235}\text{U}$ in the primary coolant during testing of the experimental fuel elements as opposed the values typical for the core with the leak tight fuel elements. Following the in-pile tests the experimental fuel elements have been subjected to test for leak. As a result of the leak tests no release of noble radioactive gases and $^{137}\text{Cs}$ has been revealed. These data and the in-pile tests set the basis for the conclusion on tightness of the experimental fuel elements.

3. Post-reactor examinations

The post-reactor examinations have been conducted using such techniques as visual examination, profilometry, volume measurements, gamma spectroscopy, metallography and X-ray structure analysis. The state of fuel cladding and fuel core has been examined in six axial cross-sections of the fuel element $H$ (0.1$H$, 0.25$H$, 0.5$H$, 0.75$H$, 0.9$H$ and near to the upper plug).

**The state of the fuel elements.**

The fuel elements have retained their shape, there were found no bend distortion of the fuel element or swelling of fuel cladding. The outer surface of the fuel elements is mat, of light gray colour and without any peeling of the oxide film. Maximum registered depth of the oxide film is $\sim 7.3$ µm. In terms of its phase composition the oxide film is the mixture of two hydroxide phases of aluminium, i.e. bayerite $\beta$-Al(OH)$_3$ and bemite $\gamma$-AlOOH in a 3:1 ratio. The dominance of low-temperature modification of aluminium hydroxide $\beta$-Al(OH)$_3$ indicates that fuel cladding temperature did not exceed 100°C. In accordance with the data of gamma spectroscopy the profiles of energy release distribution along all sides of both fuel elements are almost symmetrical with respect to their centres. The difference in energy release at different sides of the fuel elements does not exceed 10% that is the evidence of almost uniform distribution of fuel particles in the fuel core. The axial component of the peaking factor $K_Z$ for the various sides of both fuel elements falls in the interval of 1.12-1.22.

**The changes in volume and geometrical sizes.**

The change in "turnkey" size for both fuel elements has a sinusoidal shape with its maximum being at $\sim 50$ mm below the centre of the core. Maximum deviation of the size does not exceed $\sim 0.4$ % of its initial value. The profile of change in "turnkey" size is similar to the profile of heat flux density (Fig. 2).

![Fig. 2. The change in "turnkey" size along the length of the fuel element of type 2:](image-url)
Based on data of hydrostatic measurements the volume of both fuel elements has increased by \(~1.6\%\) in the first fuel element and by \(~1.2\%\) in the second fuel element. The estimated reduction of the flow area in between the fuel elements characterized by the above changes in volume would make \(~2\%\).

More detailed material science investigations have been performed for the fuel element of type 2 which had experienced higher thermal loads during the in-pile tests. Fig. 3 shows the plots of the changes in FE wall and cladding thickness along the fuel element of type 2.

![Graph showing changes in FE wall and cladding thickness](image)

Fig. 3. The axial changes in FE wall thickness $\delta_{\text{wall}}$ and thickness of external and internal FE claddings $\delta_{\text{clad}}$

The measured thickness of FE wall is in the range of 1.32-1.34 mm that is within the tolerance of 1.35-0.25 mm in accordance with the drawings which means that depth of uniform corrosion is insignificant. Thickness of both FE claddings is almost the same along the entire fuel element (excluding the area of the upper plug) and falls in the following ranges:

- external cladding from 0.310 mm to 0.610 mm;
- internal cladding from 0.340 to 0.655 mm.

Thickness of the active layer varies from 0.225 to 0.565 mm.

Corrosion of FE cladding is uniform and local. Depth of uniform corrosion does not exceed 14 $\mu$m and of local corrosion not deeper than 20 $\mu$m. Total depth of corrosion is not in excess of 9% of the initial thickness of FE cladding.

**Composition and microstructure of the active layer.**

In accordance with the metallography data the dispersed fuel core is characterized by the strong diffusive adherence to FE claddings along the entire length of the fuel element. No lamination was revealed inside the fuel core. The band of damage and the area of interaction between fuel and aluminium matrix were found around almost all fuel particles. The particles of uranium-molybdenum fuel have no pores of gaseous fission products which are typical for dioxide uranium fuel at the similar level of burnup.

**4. Conclusions**

The successful in-pile tests have been conducted for the experimental fuel elements containing metal fuel (U-9% Mo)+Al of 36% enrichment by $^{235}$U and with uranium density of 5.4 g/cm$^3$.

In the course of the effective 107 days of tests the average burnup in the fuel elements amounts to 22% by $^{235}$U. In terms of accumulated fission products it is equivalent to 40% burnup for fuel with 19.7% enrichment by $^{235}$U.

The fuel elements have retained their shape. The change in FE volume does not exceed 1.6%, while the change in "turnkey" size is not above 0.4%.
The fuel elements are covered by the thin even oxide film of light gray color. No peeling or spalling of the oxide film has been revealed. Corrosion of FE claddings is uniform and local. Depth of uniform corrosion does not exceed 14 μm and of local corrosion not deeper than 20 μm. Total depth of corrosion is not in excess of 9% of the initial thickness of FE cladding.

A certain irregularity has been detected in thickness of both FE claddings. However, their minimum thickness along almost entire FE (excluding the area near to the upper plug) is within the lower limits according to the design documentation for fuel elements in the initial state. Neither lamination of the fuel core, nor its peeling off FE cladding has been found. No peeling of the fuel core matrix off the fuel particles has been revealed.

Isolated small non-continuities with maximum size of 0.1 mm have been found in the fuel core. However, any negative impact of such non-continuities on the fuel element performance has not been identified.

Insignificant interaction has been detected between the aluminium matrix of the fuel core and the fuel particles made of U-9% Mo. Maximum width of the contact area not exceeding 4 μm has been found at the centre of the fuel element. Based on data of X-ray phase analysis the interaction area consists of uranium aluminide UAl₃.

Most of the fuel particles made of U-9% Mo alloy are of spherical type. No significant defects, including those induced by radiation (pores) have been found in the fuel particles.

Based on the positive results of the in-pile tests and the data of the comprehensive post-reactor material science investigations it can be concluded that the fuel elements with metal uranium-molybdenum fuel ((U-9% Mo)+Al) with uranium concentration of 5.4 g/cm³ are serviceable to the burnup levels of 40% by ²³⁵U.

5. Reference

TECHNICAL PROJECT FOR RECONSTRUCTION
OF THE RESEARCH REACTOR IRT - SOFIA

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ABSTRACT

As a result of analyses and investigations and on the basis of the prepared report, the
Government has taken the Resolution № 552/17.05.2001 for refurbishment of the IRT-200
Research Reactor into a Low Power Reactor up to 200 kW. The IRT-200 Research
Reactor was commissioned in 1961. In this paper the design features and experimental
capabilities of the IRT-200 pool-type research reactor are reviewed. In this reactor the IRT-
2M type fuel assemblies (FA) containing 36 % enriched uranium will be used instead of the
EK-10 fuel elements (FE). The main characteristics of the IRT-2M type FA are given, as
are the characteristics of the IRT-200 reactor. Use of the IRT-2M type FA will make it
possible to operate the reactor with a core volume of 42-48 litres. The reactor should
provide also for possible application of the “Boron neutron capture therapy /BNCT/” for
brain tumors treatment. The selected design features will significantly improve the
operational safety of the reactor.

For the implementation of this modernization project, the INRNE called upon the services
of INTERATOM (consortium of AtomENERGOPERPROJECT/SKODA/Kurchatov Institute),
of the consortium BELGATOM/SCK•CEN, of the NRI-Rez and of the EQE-Sofia.

1. Introduction

The basic objective of the IRT-2000 reconstruction is its future utilization as a low power reactor
complying with the nuclear and radiation safety requirements. It would be utilized for: teaching
physics, nuclear engineering and biological science students, specialization of the staff, neutron
activation analysis, radiochemistry, production of radioactive isotopes for medicine, neutron capture
therapy, scientific investigations within nuclear physics, physics of solid states, neutron physics, radiation biology, the industry and the agriculture, instrument testing and calibration.

The ideas which govern the proposed reconstruction can be summarized as:
- operation of the research reactor as a low power reactor, which would guarantee heat removal and fuel element integrity preservation during all steady-state and transient conditions;
- compact arrangement of the reactor systems within an earthquake resistant building;
- replacement of the obsolescent equipment with a new one meeting the up-to-date requirements and consequently making possible the operation of all the reactor systems even in case of beyond design-basis events.

2. Features of the IRT-200 reactor design

The IRT-200 reactor construction solutions are in conformity with certain existing elements of the IRT-2000 reactor and some basic elements of the IRT-5000 project have also been used [1]. The vertical section of IRT-200 reactor construction is shown in Figure 1. The horizontal section is shown in Figure 2. The reactor replaces the existing reactor IRT-2000 reactor. The tank of the reactor pool, the in-pool devices and the transportation channel are new constructions. Some existing elements, such as parts of the horizontal channels and their shutters will be kept and will be used also after the reconstruction. The proposed construction is in conformity with the necessity for the reactor to serve not only for scientific research and experimental activities but also for medical purposes (BNCT).

The new reactor vessel is arranged over the delay tank in order to ensure the possibility of maximal utilization of the existing experimental channels. Depending on the needs, the configuration of the reactor core could be changed substantially. The reactor vessel consists of the following parts: supporting plate, body with frame and bushes of channels.

The supporting plate with dimensions 760 x 640 mm is made of aluminium EN AW–AlMg3 EN 573-3 and is thick 55 mm thick. It provides a supporting grid with 54 cells (6 x 9).

The body with the frame of the reactor vessel, 870 mm high is fixed over the supporting plate. It is a welded construction of aluminium sheets EN AW-AlMg3 EN 573-3 with 8 mm thickness. At its bottom part there is a flange for fixing to the supporting plate, while at its upper part there is a frame for fixing of elements put in the reactor vessel. The bushes (5 pieces) of the channels are joined to the bottom flange and are used as leaders of the vertical channels with diameter 54 mm. Holes with 55 mm diameter are foreseen in the frame for possible utilization of short experimental channels (EC).

The elements to be put in the reactor vessel are: fuel assemblies (FA), beryllium blocks of different types and other elements (displacements) for different purposes.

Four-tubes and three-tubes FA of IRT-2M type [1] have been used in the IRT-200 reactor. The FA are arranged in a square grid with 7.15 cm pitch. The channels with control and protection system (CPS) rods and the experimental channels are placed in the three-tube FA. The main parameters of the IRT-2M type FA are provided in the Table 1 [1].

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Four-tube FA</th>
<th>Three-tube FA</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{235}$U content (nominal) in the FA, g</td>
<td>230</td>
<td>198</td>
</tr>
<tr>
<td>Water volume in the FA</td>
<td>0.726</td>
<td>0.649</td>
</tr>
<tr>
<td>$^{235}$U concentration, g/l</td>
<td>77.6</td>
<td>66.8</td>
</tr>
<tr>
<td>Fuel enrichment, %</td>
<td>36</td>
<td></td>
</tr>
<tr>
<td>FE thickness, mm</td>
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</tr>
<tr>
<td>Thickness of gap between FEs, mm</td>
<td>4.5</td>
<td></td>
</tr>
<tr>
<td>Clad material</td>
<td>Aluminum alloy CAB-1</td>
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<tr>
<td>Meat material</td>
<td>$\text{UO}_2 – \text{Al}$</td>
<td></td>
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<tr>
<td>Meat length, cm</td>
<td>60</td>
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</tr>
</tbody>
</table>
Fig 1. Reactor IRT-200.

Vertical section
1. Reactor tank (stainless steel);
2. Reactor core;
3. Delay tank;
4. Ejector tank;
5. Pipeline of primary cooling circuit;
6. Transport channel;
7. Supporting platform;
8. Actuators of control and protection system (CPS);
9. Experimental channel Ø 190 mm;
10. Experimental channel Ø 54 mm;
11. Platform with parapets.

Fig 2. Reactor IRT-200. Horizontal section
1. BNCT channel; 2. Channel Ø 150 mm; 3. Channel Ø 100 mm; 4. Ejector; 5. IC channel;
The horizontal cross-sections of the IRT-2M type FA are shown in Figure 3. In the reflector of the IRT-200 reactor beryllium blocks with 6.9x6.9 cm cross-section are applied.

![Four-tube FA and Three-tube FA](image)

Fig 3. The horizontal cross section of the IRT-2M type FA.
1 – fuel elements; 2 – channel of the CPS rod; 3 – CPS rod; 4 – displacement tube

In the CPS of the IRT-200 reactor, modernized actuators of the UR-70 type are used. This concept of a “rigid rod” replaces the original design of the IRT-2000 reactor equipment control by means of cable wire, where in case of an accident there was a risk that the absorber to be pushed up the core by a steam bubble. In the UR-70 actuator the absorber rod is connected with the drive by means of a ball-coupling controlled from above.

In the CPS 3 channels are provided for measurement of reactor power. They provide measurement of neutron flux density. Signals from the channels are fed to operator desk in the control room. Two identical independent channels are foreseen, signals from which are fed to desk of the emergency Standby station.

The channel for BNCT consists of the following parts: channel and aluminium plate. The channel is made of stainless steel plate, 15 mm thick, with a rectangular profile of size 700×550 mm welded onto the pool bottom section wall. The channel face is made of stainless steel sheet of 2 mm thickness and from the pool side face is reinforced by screwed-on aluminium plate of 90 mm thickness which at the same time forms the first part of the neutron filter.

Reactor cooling is provided by pool water flowing downstream via the core and reflector into the delay tank of a volume of 2 m³ by the ejector and pump working in parallel.

3. Main parameters of the initial working loading of the IRT-200 reactor

The initial core configuration of the reactor is shown in Figure 4. This configuration permits simultaneous operation of the BNCT channel and other horizontal experimental channels as well as vertical experimental channels. The excess reactivity of the initial working loading when the filters are presented in the BNCT channel (50 cm Al) and temperature - 20°C is equal to 3.2% Δk/k.

The calculated reactivity value of the CPS rods in the initial working loading (% Δk/k):
- safety rods: A3-1 = 2.4, A3-2 = 2.4, A3-3 = 0.33;
- shim rods: KO-1 = 4.05, KO-2 = 4.05, KO-3 = 0.48, KO-4 = 0.48, KO-5 = 0.33;
- automatic regulating rod: AP = 0.16.

When 3 safety rods are fully out of the core, 5 shim rods and AP rod are fully inserted in the core, the reactor subcriticality equals 6.35 % Δk/k.
When the most reactive shim rod (KO-1 or KO-2) is fully out of the core, the reactor shutdown margin will be not less 5.5 % \(\Delta k/k\) (by minimum necessary 4.2 % \(\Delta k/k\)). Scram shutdown margin (worth of A3-2 and A3-3, when rod A3-1 failure) will be not less than 2.75% \(\Delta k/k\). Temperature effect (20°C – 50°C) – 0.3 % \(\Delta k/k\). Temperature factor of reactivity – negative, 1·10^{-2} % \(\Delta k/k\) / °C. Neutron life time – 40 µs.

The maximum density of thermal neutron flux (E < 0.625 eV) is reached in VECs located in beryllium blocks in cells D3 and D4 and is equal to 8.40·10^{12} cm^{-2}s^{-1}. The maximum density of fast neutron flux (E > 0.821 MeV) is reached in VECs located in three-tube FA in cells C3 and C4 and equals 2.80·10^{12} cm^{-2}s^{-1}.

The maximum density of thermal neutron flux (E < 0.625 eV) is reached on the entrance of the HEC 4 and is equal 1.6·10^{12} cm^{-2}s^{-1}. The neutron flux density (0.625 eV – 0.821 MeV) for the BNCT channel to the position just behind the 9 cm aluminum plate equals 8.2·10^{11} cm^{-2}s^{-1}.

The neutronic calculations have been carried out by following codes: MCNP4C, WIMS-ANL, REBUS-ANL. The WIMS-ANL code was used for cross-sections libraries preparation in the ISOTXS format.

4. References

DEVELOPMENT OF IRT-TYPE FUEL ASSEMBLY WITH PIN-TYPE FUEL ELEMENTS FOR LEU CONVERSION OF WWR-SM RESEARCH REACTOR IN UZBEKISTAN

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ABSTRACT

The design of new IRT-type fuel assembly (FA) with pin-type fuel elements has been developed. The IRT-type FA consists of the end details and a square shroud, in which 176 pin-type fuel elements are placed by means of two spacer grids. Fuel element is a square cross-section pin with four ribs curled around longitudinal axis. A large set of technological investigations of fuel element and FA fabrication processes has been completed, including:
- development of drawing-design and technological documentations;
- development of the design of spacer as a top and bottom grid;
- manufacturing of needed tools and rigs;
- development of fabrication process with fuel element simulators;
- development of fabrication process for outer and inner shrouds;
- development of the process of FA assembling.
As a result two full-scaled dummy of FA for hydraulic tests have been manufactured and ability for manufacturing of fuel element and assembly have been confirmed under conditions of industrial production.
To define the hydraulic characteristics of IRT-type FA with pin-type fuel elements the hydraulic tests of two full-scaled dummy of FA have been carried out. A large set of neutronics, thermal hydraulic and strength calculations for the WWR-SM reactor in Uzbekistan has been performed on substantiation of FA design. As a result:
- the dispersion LEU fuel U-Mo in Al matrix has been chosen
- the optimum U-235/FA loading has been defined;
- the optimal parameters of pin-type fuel element and fuel assembly have been determined.
These research of FA design are being performed in close cooperation with ANL within Russian and international RERTR programs.