# STATUS AND DEVELOPMENT OF RBMK FUEL RODS AND REACTOR MATERIALS

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**Abstract**

The paper presents current status and development of RBMK fuel rods and reactor materials. With regard to fuel rod cladding the following issues have been discussed: corrosion, tensile properties, welding technology and testing of an alternative cladding alloy with a composition of Zr-Nb-Sn-Fe. Erbium doped fuel has been suggested for safety improvement. Also analysis of fuel reliability is presented in the paper.

## 1. CONDITIONS OF FUEL ROD OPERATION IN RBMK-1000 AND RBMK-1500.

The conditions of fuel rod operation in RBMK-1000 and RBMK-1500 in Table 1, 2.

## 2. SOME SPECIFIC FEATURES OF RBMK FUEL ROD OPERATION

RBMK fuel assembly consists of two vertically positioned fuel rod bundles that are separated in space by a discontinuity in the core centre. The RBMK reactor is operated under conditions of FA continuous reloading. These circumstances condition the two main operation features of fuel rods, namely,

- weld joints are in the core centre;
- in the region of the fuel column discontinuity a power burst (up to ~30%) is realised.

These factors define rather rigid operation conditions for both fuel rod claddings and particularly, weld joints.

One more specific feature of RBMK fuel rod operation consists in the fact that low fuel burn-up (some 2.5-3 times lower than of PWR, WWER or BWR) is combined with the adequately long operation of fuel rods (1100-1200 eff.days) without control of the coolant water chemistry for radiolytic gases at the oxygen content of water up to 20 $\mu g/dm^3$.

This circumstance gives the priority to some factors used to assess the fuel ultimate condition, specifically, nodular and general water-side corrosion, including that under spacer grids, fuel cladding overheats due to cruds and high thickness oxide films available on them, local hydriding, influence of coolant impurities on the temperature conditions of fuel rods, influence of heat flux of a fuel rod on the rate of oxide film growth on fuel claddings etc.

## 3. CORROSION OF FUEL CLADDINGS

As regards boiling reactor fuel rods also those of RBMK, corrosion of Zr-alloys is a dangerous phenomenon that may substantially confine the life-time of fuel rods. The corrosion condition of RBMK fuel rods was studied using $FA_s$ that have operated to different burn-ups, the design burn-up included.

It is established that along with the general corrosion as a uniform dark-coloured oxide film Zr-1% Nb claddings also develop local types of corrosion, namely, nodular corrosion, drastically increased corrosion nodules under spacer grids. The acquired data on the corrosion condition of the RBMK fuel rods are tabulated in table 3.

The nodular corrosion had a typical appearance of white coloured nodules is local thickening at the background of a dense uniform film of dark coloured Zr oxide. The nodules had a lenticular shape with a bulge in the centre. The uniform oxide film thickness varies very little along the fuel rod length, however, the sizes of the local corrosion nodules increase both under and outside the spacer grid with the proximity to the core centre (table 4). No nodular corrosion was observed in the plenum region.

It can be seen from the table the local corrosion under the spacer grid is very intensive. A decrease in the fuel rod cladding cross-section in those sites may reach ~40% and more. These local thinning of claddings are stress concentrator sites and can be a cause of a crack nucleation and its further evolution to a primary penetrating defect.

TABLE 1. CONDITIONS OF RBMK-1000 AND RBMK-1500 FUEL ROD OPERATION

| № | Parameter | RBMK-1000 | RBMK-1500 |
|---|---|---|---|
| 1 | Mean heat generation rate, W/cm<br>- irradiation onset<br>- irradiation end | <br>250<br>139 | <br>405<br>215 |
| 2 | Maximum heat generation rate,W/cm | 350 | 485 |
| 3 | Coolant Temperature, $^0$C<br>at FA inlet<br>at FA outlet | <br>270<br>284 | |
| 4 | Coolant pressure, MPa | 6,7 | 6,7 |
| 5 | Maximum temperature of fuel rod cladding, $^0$C | 367 | 372 |
| 6 | Fuel burn-up, MW·day/t U<br>average<br>maximum | <br>26000<br>30000 | <br>22500<br>26800 |
| 7 | Neutron fluence (E≥1 Mev),n/cm$^2$ | $1{,}3 \cdot 10^{21}$ | $1{,}3 \cdot 10^{21}$ |

TABLE 2. CHARACTERISTICS OF PRIMARY CIRCUIT COOLANT UNDER RBMK-1000 AND RBMK-1500 NORMAL OPERATION CONDITIONS

| Parameter | Value |
|---|---|
| pH at 25$^0$C | 6.5-7.8 |
| Specific electrical conductivity at 25$^0$C, $\mu$ Sm/cm, not more than | 1.0 |
| Mass concentration of chloride-ion, $\mu$g/dm$^3$, not more than | 70 |
| Hardness, $\mu$g-ef/dm$^3$, not more than | 5 |
| Mass concentration of $SiO_3$, $\mu$g/dm$^3$, not more than | 700 |
| Oxygen, $\mu$g/dm$^3$, not more than | 20 |
| Mass concentration of iron, $\mu$g/dm$^3$, not more than | 50 |
| Mass concentration of copper, $\mu$g/dm$^3$, not more than | 20 |
| Mass concentration of petroleum product, $\mu$g/dm$^3$, not more than | 100 |

The dense oxide film at the outer surface is covered with a deposit layer the colour of which changes from light grey to dark-ginger depending on the thickness. Its thickness was within 7-10 μm. Oxide films and cruds are taken into account by the design analysis of the fuel rod temperature conditions.

## 4. TENSILE PROPERTIES OF CLADDINGS

As irradiated the fuel rod claddings have rather high tensile properties that do not essentially vary over the fuel rod active part (table 5). No substantial distinction of the metal properties was revealed outside and under spacer grids. The plenum area is characterised by higher ductility compared to the other areas of the fuel rod.

TABLE 3. CORROSION CONDITION OF FUEL ROD CLADDING AS OPERATED IN RBMK-1000 (LENINGRAD NPP)

| Cladding material | Fuel burn-up, Mwt/kg | Time of operation, h | Uniform oxide film thickness, μm | Maximum size of corrosion nodules, μm | | Mass fraction of hydrogen,% | |
|---|---|---|---|---|---|---|---|
| | | | | outside SG | under SG | outside SG | under SG |
| Zr+1%Nb | 1.3 | 3770 | 10-20 | 40 | 150 | $\sim 3\text{-}6 \cdot 10^{-3}$ | - |
| | 9.9 | 26440 | 10-20 | 60 | 180 | $\sim 1 \cdot 10^{-2}$ | - |
| | 19.3 | 29112 | 15-20 | 130 | 380 | $\sim 1 \cdot 10^{-2}$ | $\sim 1.5 \cdot 10^{-2}$ |

TABLE 4. THICKNESS OF CORROSION NODULES IN DIFFERENT AREAS OF FUEL ROD

| Distance from the top of upper FA rod, mm | Point of measurement | Magnitude of corrosion nodules, μm |
|---|---|---|
| 500-800 | outside SG | 40 |
| | under SG | 80 |
| 1550 - 3500 | outside SG | 140 |
| | under SG | 380 |

TABLE 5. MECHANICAL PROPERTIES OF FUEL ROD CLADDING AFTER SERVICE LIFE IRRADIATION

| Alloy | $T_{test}$, $^{0}C$ | Mechanical properties (in fuel column region) | | |
|---|---|---|---|---|
| | | $\sigma_d$, MPa | $\delta_{total}$,% | $\delta_{unif}$, % |
| Zr+1%Nb | 20 | 630 | 9,2 | - |
| | 300 | 480 | 12,0 | ~2,5 |

## 5. RESULTS OF TESTING PILOT FUEL ASSEMBLIES WITH ZR-NB-SN-FE CLADDINGS IN RBMK-1000 AT LENINGRADSKAJA NPP

The Zr-Nb-Sn-Fe alloy was designed at VNIINM early in 70-is for application as a fuel rod cladding material both in VVER and RBMK. This alloy was also under consideration for other core components, namely, guide thimbles of control and protection system (VVER), pressure tubes (RBMK) and etc.

The experimental fuel rods clad in this alloy were successfully tested in the research reactors (MIR, MR). To generate experimental data on a large massive of fuel rods under conditions of the commercial units of $NPP_s$ 38 pilot fuel assemblies were produced that in their designs did not differ from the standard RBMK fuel assemblies. As a cladding material of the fuel rods of those $FA_s$ use was made of Zr+1%Nb+1.3%Sn+0.35%Fe. The pilot fuel assemblies were tested at LNPP to reach the service-life burn-up. The results of the corrosion resistance testing of the fuel rods clad in this new alloy are tabulated in table 6.

As it can be seen from table 6 the outer Zr-Nb-Sn-Fe cladding surface was not generally subjected to nodular corrosion. The maximum local corrosion of this alloy under the spacer grids is some 3.5 times lower compared to that of Zr+1%Nb alloy.

From the comparison between the tensile properties of the cladding materials it is established that the strength characteristics of Zr-Nb-Sn-Fe alloy are much higher and its ductility is lower than those of Zr+1%Nb (see table 7).

## 6. SOME ISSUES RELEVANT TO STRENGTH OF CLADDING WELDS

As it has already been mentioned the welds of the RBMK fuel rods (one weld of the top fuel rods and one of the bottom ones) are located in the core centre where due to the fuel column discontinuity a power burst is realised. These circumstances make the conditions of the weld operation very rigid.

Under these conditions the application of the progressive resistance butt welding in 1982 made the problem of welds strength more acute. End plug detachments were systematically observed even at the initial stage of irradiation at a low fuel burn-up. These events were of adequate significance. Approximately in 5% events the cause of fuel rod leakage was plug detachments. As a rule it was the bottom plug of a fuel rod in the top bundle of $FA_s$ (table 8).

TABLE 6. CORROSIOM RESISTANCE OF Zr-Nb-Sn-Fe CLADDING OF FUEL RODS AFTER OPERATION IN RBMK-1000 (LENINGRAD NPP)

| Cladding material | Fuel burn-up MW·day/ kgU | Time of operation, h | Uniform oxide film thickness, µm | Maximum thickness of lokal corrosion nodules, µm | | Mass fraction of hydrogen, % | |
|---|---|---|---|---|---|---|---|
| | | | | outside SG | under SG | outside SG | under SG |
| Zr+1%Nb+ 1,3%Sn+ +0,35%Fe | 11,4 | 25488 | 10-20 | 0 | 0 | $7{,}5\cdot10^{-3}$ | - |
| | 19,3 | 29112 | 15-20 | 0 | 100 | $1{,}5\text{-}2{,}0\cdot10^{-2}$ | $1{,}5\text{-}2{,}0\cdot10^{-2}$ |

TABLE 7. MECHANICAL PROPERTIES OF Zr-Nb-Sn-Fe CLADDING OF FUEL RODS AFTER SERVICE-LIFE IRRADIATION (19.3 MWd/kg U)

| ALLOY | $T_{test}$, °C | Mechanical properties (in fuel column region) | |
|---|---|---|---|
| | | $\sigma_d$, MPa | $\delta_{total}$, % |
| Zr+1%Nb+ 1,3%Sn+0,35%Fe | 20 | 720 | 5,3 |
| | 300 | 520 | 4,0 |

TABLE 8. SOME DATA OF LOSS OF TIGHTNESS BY FUEL RODS BY PLUG DETACHMENT

(upper FA, weld №1) at units №1 and №2 of LNPP (fuel rod sealing by RBW, plug detachment in confirmed by PIE in hot cells of SB R&DIPE)

| № of FA | Unit | Burn-up, MW·day/bundle | Fabrication date | Load date | Discharge date |
|---|---|---|---|---|---|
| 1-20-7-26PT | 1 | 194 | 10.82 | 03.83 | 06.83 |
| 1-20-7025PT | 1 | 556 | 10.82 | 02.83 | 11.83 |
| 1-20-5784 | 1 | 657 | 08.82 | 11.82 | 09.83 |
| 1-20-6233 | 1 | 474 | 09.82 | 01.83 | 09.83 |
| 1-20-2319 | 1 | 308 | 01.82 | 04.82 | 11.82 |
| 1-24-0451 | 2 | 655 | .81 | 01.82 | 11.82 |

Laboratory and $PIE_S$ showed that the cause of the fuel rod failure due to a plug detachment was intensive hydrading of the weld area under conditions when the fuel column rested upon the weld burr resulting from fuel rod sealing.

The improved plug design (Fig. 1) having an internal projection thus eliminating the direct contact between the hot fuel and the burr significantly improved the state of the art making fuel failures due to a plug detachment casual and being realised as a rule by the "secondary hydrading" mechanism.

From this viewpoint of significance is the analysis of the experience gained in the operation of the fuel rods of unit 1 and 2 of Ignalina NPP put into operation in 1984. The design of the RBMK-1000 fuel rods took into account the accumulated experience and the advanced design and technological solutions were realised, namely:

- use of a plug having a projection upon which two blanket pellets of a lower $U^{235}$ content (from 0.4% to 0.7%) but not the $UO_2$ (2% enrichment of $U^{235}$) fuel column rest;
- use of fuel pellets with central holes;
- anneal of welds.

Through the blanket pellet location the region of the radial thermal mechanical pellet-cladding interaction is placed away from the plug, thus eliminating extra tensile stresses in the weld area that are induced by the side effect (Fig. 3, 4). This figure shows the distribution of the design elastic stresses simulating the radial loading of a cladding accomplished by swelling fuel (the account is also taken of the radial coolant pressure - 6 MPa) with blanket pellets available and without them. It is seen that during operation the absence of blanket pellets may result in grass tensile stresses in the weld area under the action of the side effect.

Besides, blanket pellets also lower a power burst providing an efficient protection against weld hydrading (Fig. 2).

The optimal character of this solution may be illustrated by the following data on the fuel service at INPP. At this plant during the whole period of the operation of the two RBMK-1500 units there were only four events of fuel rod failures due to plug detachments (1985, 1986, 1988 and 1995), i.e., 1 failure per ~200 thousand discharged fuel rods for the two units on the average. During the recent 8 years there was a single event of a plug detachment, i.e., 1 failure per 550 thousand discharged fuel rods. This event took place in a fuel rod of the bottom fuel bundle discharged in 1995 among 35 leaky $FA_s$. The fuel rods of those 35 $FA_s$ were overheated due to the unscheduled cooling conditions.

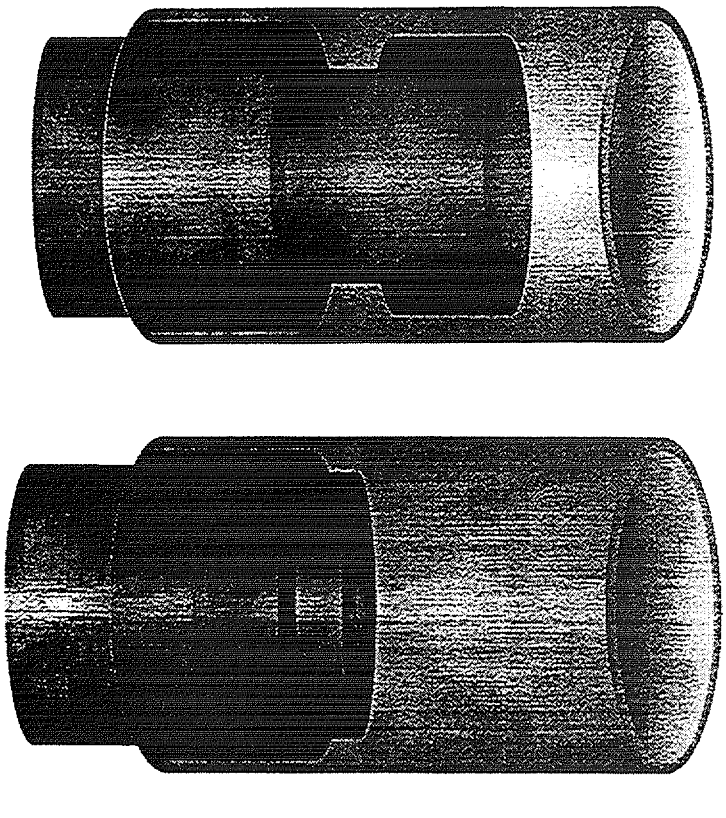
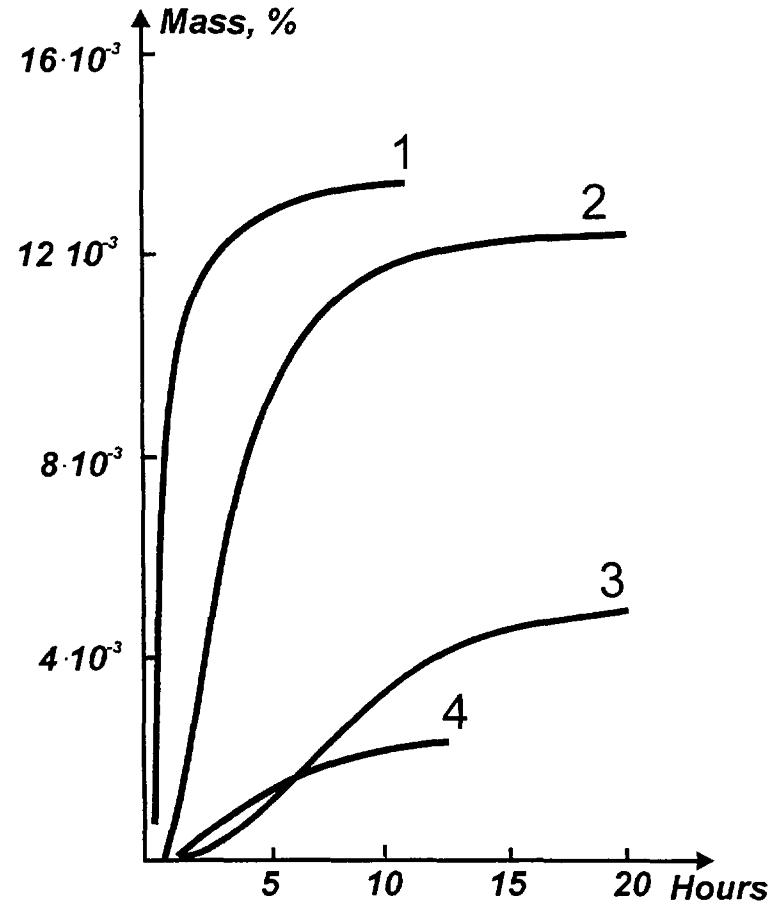


*Fig. 1. Drawing of the end plug with a "mushroom" (upper) and with a boss (lower).*




1-RBMK fuel welded joint with fuel on burr,
2-standard RBMK-1000 fuel welded joint with fuel on boss,
3-welded joint with "mushroom" plug having 3.5 mm diameter stem,
4-standard RBMK-1000 fuel welded joint having plug with boss and depleted (0.4% enriched) pellets.

*Fig. 2. Hydrogen concentration variation in resistance butt welded joints of different design in RBMK fuel rod.*

It is to be noted that there is an interrelation between the events of loss of tightness by fuel rods for the reason of a plug detachment and the total quantity of leaky $FA_s$. This circumstance may be explained by the fact that plugs are detached by the "secondary" hydrading mechanism as a result of the mixed steam-water ingress into the fuel rod interior via a small primary defect.

To-day in a pilot operation at LNPP are fuel rods having a mushroom shaped plug (Fig. 1). This mushroom plug promotes an improved resistance to weld hydriding.

## 7. OPERATION RELIABILITY OF FUEL RODS

Presently the operation reliability of the RBMK fuel rods differs very much between not only different $NPP_s$ but also different units of the same NPP.

At the same technology level the operation reliability can vary between different units within $10^{-5}$ to $10^{-4}$ per year. On the one hand, this is an evidence that the operation conditions of the RBMK units have a significant influence on the fuel rod damageability, and, on the other, that those conditions are quite different in different reactor units even for the operation routine common for all $NPP_s$.

Therefore, the fuel damageability can be reduced via the further optimisation of the RBMK core operation parameters.

From our point of view, most significant and useful for the understanding of the problem will be the analysis of the state of the art at the two units of INPP. If has already been mentioned that at this plant the fuel rods operate at heat loads a factor of 1.25-1.3 higher than in other RBMK units. Table 9 summarises the data on the number of discharged leaky $FA_s$ of INPP submitted to us by Mr. VORONTSOV B.A., the deputy chief engineer on INPP safety. It follows from the tabulated data that during 1991-1995 the level of the failures at unit 1 was $\sim 1.4 \times 10^{-5}$/year. As far as unit 2 is concerned without the account for 40 $FA_S$ discharged in 1994-1995 for the reason of the unscheduled overheating of the fuel rods the level of the failures is $\sim 2.0\text{-}2.5 \times 10^{-5}$ /year. To-day this failure level is typical of the modern cores of $NPP_S$ with VVER, BWR and PWR. To attain those high results, at INPP a large scope of work was implemented to optimise the core performance, namely, algorithms of stabilisation (within 15% of power rating fields) were introduced during $FA_S$ reloading, additional absorber discharge and replacement of water columns by $FA_S$ etc.

Also the problems were resolved relevant to a reduction of axial deviations of power density fields.

TABLE 9. NUMBER OF LEAKY FUEL ASSEMBLIES DISCHARGED FROM UNITS 1 AND 2 OF IGNALINA NPP (1988-1995)

| YEAR | UNIT 1 | UNIT 2 |
|---|---|---|
| 1988 | 10 | 5 |
| 1989 | 9 | 6 |
| 1990 | 9 | 8 |
| 1991 | 1 | 7 |
| 1992 | 3 | 5 |
| 1993 | - | 1 |
| 1994 | 2 | 5 |
| 1995 | 1 | 35 |

*** The data are submitted by Mr. Voronzov V.A., Chief engineer deputy on safety, INPP**

## 8. RBMK FUEL DEVELOPMENTS AND PRODUCTION USING ERBIUM AS BURNABLE ABSORBER

After the Chernobyl accident to improve the safety of RBMK type reactors the steam void reactivity coefficient was much reduced via loading some extra absorbers into the core which resulted in a lower burn-up of fuel as well as in an increase of the mean and maximum power of FA channels. The design studies aimed at reducing the steam void coefficient showed that an erbium burnable absorber as $Er_2O_3$ added to the fuel lowers down the steam void coefficient of reactivity to the level which does not require any extra absorbers to be loaded into the core. Besides, the burnable absorber available in fresh $FA_S$ reduces significantly their power and the reload effected reactivity. This simplifies the reloading procedure and control of the energy distribution within the core. The lower non-uniformity of the power rating allows a higher enrichment, thus, extending the fuel burn-up /1, 2, 3, 4/.

To-day the production of the U-Er oxide fuel has been mastered by Electrostal plant as applied to RBMK-1000 and -1500. U-Er fuelled $FA_S$ have been fabricated and now under test at Ignalina NPP (150 $FA_S$) and Leningrad NPP (200 $FA_S$)/. The work is under way to switch all the reactors to the fuel of this type. The erbium content of the oxide fuel is 0.41% mass. $U_{235}$ enrichment of the fuel was increased from 2.4% to 2.6% for RBMK-1000 and from 2.0% to 2.4% for RBMK-1500.

The process flow outline of the U-Er fuel pellet manufacture contemplates the use of uranium dioxide and erbium oxide powders as initial materials. The initial powders of uranium dioxide and erbium oxide in the specified ratio are blended in a vane mixer. A binder is added to the as sieved powder mixture. The uniformity of the erbium oxide and carbon binder is controlled. The further process does not essentially differ from the process used to manufacture the standard fuel. Thus, the only distinction is the additional technological operation of blending the initial uranium dioxide and erbium oxide powders. On the one hand, this operation is very important since a high degree of uniform erbium distribution within the pellet basic material is required, on the other, it is adequately sophisticated as the quantity of erbium introduced is low.

The adopted blending process provides for the needed uniformity of the mixture. From the results of the manufacture the degree of the fuel uniformity is such that the pellet distribution in terms of their total erbium contents has a relative mean square root deviation of 2.0%. The distribution of erbium within the volume of individual fuel pellets as determined by the x-ray spectral microanalysis is also uniform.

Based on the work performed the specifications for the U-Er pellets were established, analytical techniques of control were developed and metrologically certified; they are to control the erbium content of fuel pellets, the total content of uranium, the O/Me ratio, impurities such as nitrogen, hydrogen, carbon, fluorine, chlorine .

At Electrostal plant an isolated bay was created to manufacture U-Er oxide fuel. The technological process of U-Er pellet manufacture was designed and several pilot lots of this fuel were produced of the total mass>50t for testing in INPP and LNPP.

The properties of the U-Er fuel were studied both under laboratory conditions and in-pile. Laboratory scale investigations were carried out to study the thermal conductivity, thermal expansion coefficient, strength, thermal creep, U-Er fuel pellet - Zr+1%Nb cladding compatibility. The corrosion resistance of the U-Er fuel was investigated.

Measurements of the pellet thermal expansion within 25-1500$^0$ C showed that within the temperature range studied the linear thermal expansion of the U-Er fuel is systematically less than of $UO_2$ which may be related to a smaller ionic radius of erbium. However, the differences do not exceed 0,6%.

The thermal conductivity of pellets was determined in the temperature range of 300-1500° C. It is found out that with the introduction of Er the thermal conductivity of the fuel is lowered in the whole temperature range studied, however, it does not exceed 10%. Aside from the laboratory studies of the U-Er fuel thermal conductivity, in-pile investigations were performed. Specifically, in the IVV-2M reactor the fuel temperatures were measured under irradiation using instrumented U-Er fuel rods and the standard fuel [1]. The analysis of the acquired results implemented within the programme RET (TR) made it possible to introduce corrections into the temperature dependence of the U-Er fuel as derived under laboratory conditions (see Fig. 5).

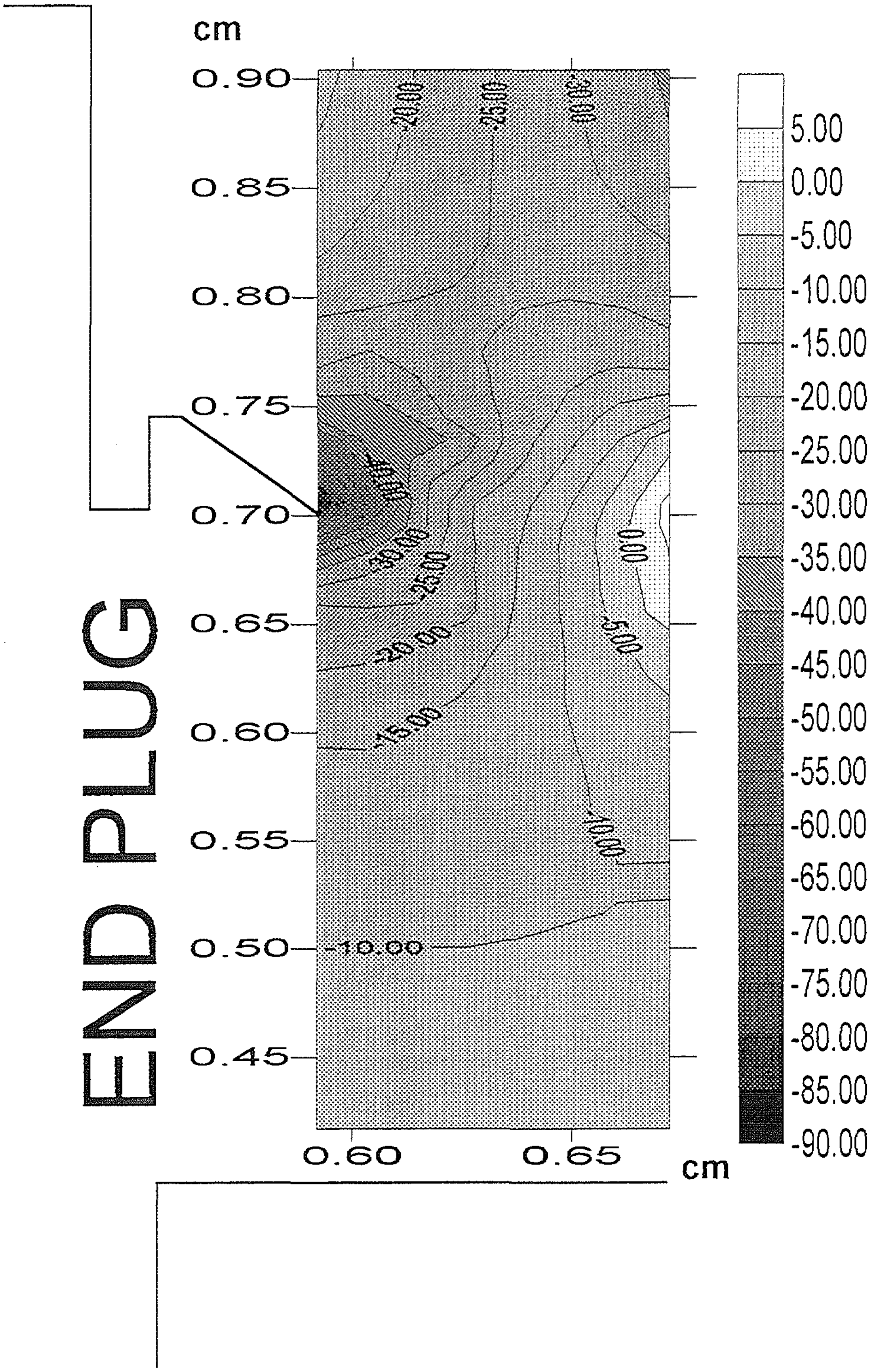



*Fig. 3. Longitudinal stresses in fuel rod cladding with blanket pellet in welded joint area, MPa.*

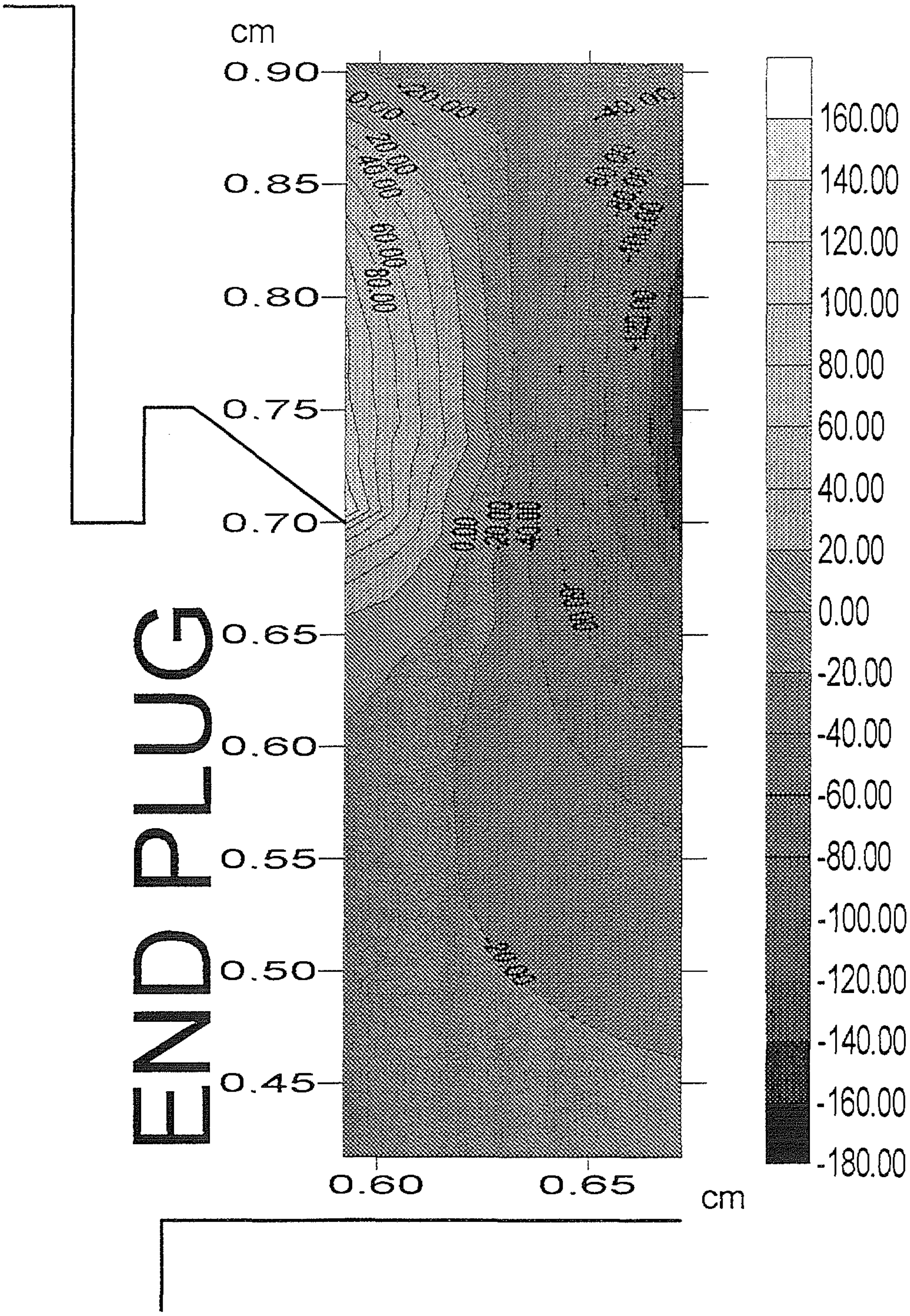



*Fig. 4. Longitudinal stresses in fuel rod cladding without blanket pellet in welded joint area, Mpa.*

The room temperature mechanical strength of U-Er fuel pellets (0.41% Er) was determined by measuring the maximum load endurable by pellets upon compression to fracture. The measurements revealed that the mean fracture strength of pellets is $\sigma$ =237.7±11 MPa. This value is above that of the standard RBMK pellets, i.e., $\sigma$ = 211± 3 MPa.

Investigations of the U-Er fuel thermal creep were launched. The set of the data on the creep as a function of stress and temperature is qualitatively described by the relationship:

$$\varepsilon \sim \exp(Q/RT+\alpha\sigma)$$

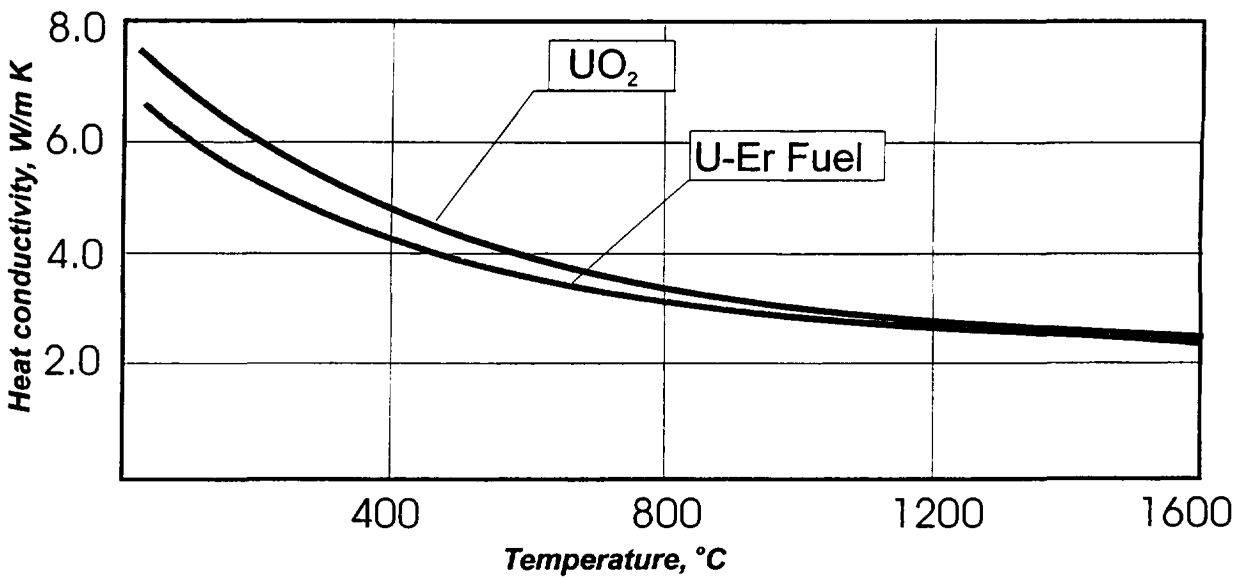



*Fig. 5. Heat conductivity of U-Er fuel.*

where ε is creep rate, σ is stress, Q is activation energy.

The activation energy of creep is equal to 460 Kτ/mole at temperatures above $1350^0$ C and 165 Kτ/mole at lower temperatures. The investigations are in progress.

The tests for the U-Er fuel - E110 alloy cladding compatibility at $800\text{-}1200^0$ C and the diffusion anneal time of 4 and 9h showed that the total thickness of the interaction zone layers is smaller for U-Er fuel compared to that for the standard $UO_2$ fuel.

The comparative study was continued of the corrosion properties of U-Er and standard fuel (P-10-E and P-10, respectively) in water at $350^0$ C and the pressure of 16.5 MPa for 25, 100, 500 and 1000h and in steam at $500^0$ C and the pressure of 7.0 MPa for 25 hours (see table 10).

To study the release of fission gas products (FGP) R&DIPE and Sverdlovsk branch R&DIPE performed in-pile tests of ventilated U-Er and standard fuel rods that allowed the assessment of unstable nuclides released inside a fuel rod cladding. The analysis of the results within the programmes RET (TR) and VKI that was carried out at VNIINM showed that at the fuel temperatures of 870-$1200^0$ C and burn-up < 3.5 MW.day/kgU (the reached burn-up, the experiment is in progress) erbium introduced into the fuel does not influence the release of FGP [2].

For the analysis of the serviceability of pilot U-Er fuel rods a series of calculations was performed, involving thermophisical calculations within the RET (TR) programme, the strength calculations under normal operation conditions and calculations of the cladding stability within the programmes START-3 and BUSTER, an analysis of the fuel rod behaviour under conditions of the severe design basis accident within the RAPTA programme. When implementing the calculations the needed refinement of the programmes was accomplished for taking into account the changes in the fuel properties induced by the introduction of erbium. The calculations were performed proceeding from the conditions of the highest heat density fuel operation that is the top bundle fuel rod of RBMK-1000 and - 1500 FAs. For the sake of comparison the thermophysical design of the standard fuel rod was carried out. The operation conditions and design parameters of the RBMK-1000 fuel

TABLE 10. RESULTS OF COMPARATIVE TESTS OF STANDARD (P-10) AND U-Er PELLETS (P-10-E) IN WATER AND STEAM

| Type of test | Type of fuel | № of specimen | Weigth gain after testing, mg/cm$^2$ | | | |
|---|---|---|---|---|---|---|
| | | | 25 h | 100 h | 500 h | 1000 h |
| Water: 350 $^{o}$C 16,5 Mpa | P-10 | 1 | 0,20 | 0,27 | 0,33 | 0,45 |
| | | 2 | 0,14 | 0,24 | 0,30 | 0,42 |
| | | 3 | 0,17 | 0,19 | 0,35 | 0,51 |
| | | 4 | 0,14 | 0,27 | 0,50 | 0,62 |
| | | 5 | 0,24 | 0,45 | 0,54 | 0,68 |
| | | mean | 0,18 | 0,28 | 0,40 | 0,54 |
| | P-10-Э | 1 | 0,41 | 0,68 | 0,69 | 0,80 |
| | | 2 | 0,32 | 0,47 | 0,54 | 0,66 |
| | | 3 | 0,18 | 0,20 | 0,33 | 0,51 |
| | | 4 | 0,33 | 0,63 | 0,66 | 0,77 |
| | | 5 | 0,39 | 0,65 | 0,72 | 0,74 |
| | | mean | 0,33 | 0,53 | 0,59 | 0,70 |
| Steam: 500 $^{o}$C 7,5 MPa | P-10 | 1 | 1,46 | | | |
| | | 2 | 1,55 | | | |
| | | 3 | 1,89 | | | |
| | | 4 | 2,49 | | | |
| | | 5 | 3,12 | | | |
| | | mean | 2,10 | | | |
| | P-10-Э | 1 | 1,48 | | | |
| | | 2 | 2,55 | | | |
| | | 3 | 2,95 | | | |
| | | 4 | 4,69 | | | |
| | | 5 | 1,82 | | | |
| | | mean | 2,70 | | | |

TABLE 11. COMPARISON BETWEEN CONDITIONS OF RBMK FUEL ROD 9 WITH STANDARD AND U-Er FUEL) OPERATION

| PARAMETER | RBMK-1000 FUEL ROD | | RBMK-1500 FUEL ROD | |
|---|---|---|---|---|
| | U-Er fuel | Standard | U-Er fuel | Standard |
| Mean heat generation rate, W/cm<br>- beginning of irradiation<br>- end of irradiation | <br>250<br>139 | | <br>360<br>210 | <br>405<br>215 |
| Fuel burn-up , Mw·day/tU<br>- average<br>- maximum | <br>26000<br>30000 | | <br>22550<br>26800 | |
| Maximum heat generation rate, W/cm | 350 | | 430 | 485 |
| Maximum cladding temperature, $^{0}$C | 367 | | 371 | - |
| Maximum fuel temperature, $^{0}$C<br>- beginning of irradiation<br>- end of irradiation | <br>1366<br>755 | <br>1411<br>760 | <br>1660<br>1010 | <br>1775<br>1050 |
| FGP release (end of cycle), % | 1,4 | 1,4 | 6,6 | 7,0 |
| Gas pressure in fuel rod (end of cycle), MPa | 1,85 | 1,83 | 3,1 | 3,1 |

rod are similar to those of erbium containing fuel rod for the exception of the standard fuel pellets without central holes. In the RBMK-1500 fuel rod the design parameters of the both fuel rods are identical, the distinctions refer to the operation conditions that at the initial moment of irradiation are more rigid (see table 11) for the standard fuel rod. Table 11 contains the design thermophysical characteristics of the standard and U-Er fuel rods of RBMK-1000 and - 1500.

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