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Cermet fuel in a light water reactor: a possible way to improve safety. Part II. Testing the fuel

V. Troyanov, V. Popov, Iu. Baranaev

State Research Center of Russian Federation, Institute of Physics and Power Engineering
IPPE, Bondarenko sq., 1, Obninsk, , Kaluga region, 249032, Russia

Abstract

This paper summarizes what is done for the experimental testing of cermet fuel with various matrix materials. Low neutron absorption, high heat conductivity, good corrosion resistance in water, low chemical interaction with cladding (zirconium alloy) and UO_2 in normal and accident conditions, technological ability – are the requirements of the matrix material [1]. Suitability of the proposed solutions to the cermet fuel design with respect to these requirements was proven through a series of experiments simulating fuel operating and accidental conditions. © 2001 Published by Elsevier Science Ltd. All rights reserved.

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1. Introduction

The practical implementation of a new nuclear fuel in a nuclear reactor requires comprehensive experimental substantiation using both laboratory and integrated in-pile tests. The important fuel properties should be investigated and proven in a variety of normal and abnormal reactor operating conditions. The series of experiments carried out in the process of cermet fuel development include:

- out-of-pile long term tests of fuel pins at operating temperature and dynamic tests in design basis accident conditions,
- in-pile life-time testing and testing of fuel behaviour under accident conditions.

*Corresponding author: V.M. Troyanov: tel: +7 08439 98540; fax: +7 095 8833112; e-mail: dmt@ippe.rssi.ru

2. Compatibility of fuel pin materials during isothermal exposure

For these experiments simulators of cermet fuel (60% vol. UO_2 + 40% vol. silumin (Si,Al)) pins have been prepared according to the technology described earlier [2]. They were tested in the process of a prolonged isothermic exposure (τ = 2700 hours) at 775 K. After the test, absence of cracks, bowings, distortions or other defects in the fuel pin simulators have been noted. The simulators retained their geometric stability. Autoradiographic investigations showed that the diffusion penetration of uranium into the matrix did not exceed 10 μm.

No difference in fuel structure has been revealed in comparison with the initial structure. The distribution of the main elements of fuel composition after tests is presented in Fig.1. One can see that the uranium concentration at the UO_2 particles - matrix boundary reduces practically to zero values in less than 10-15 μm.

X-ray phase analysis of the cermet fuel showed that the main phases of the fuel remained the initial components of the cermet fuel: UO_2 and Al, Si. However, exposure of fuel pin simulators at a temperature of 775 K yields UAl_3 and $USiO_4$. The amount of these components was lower.

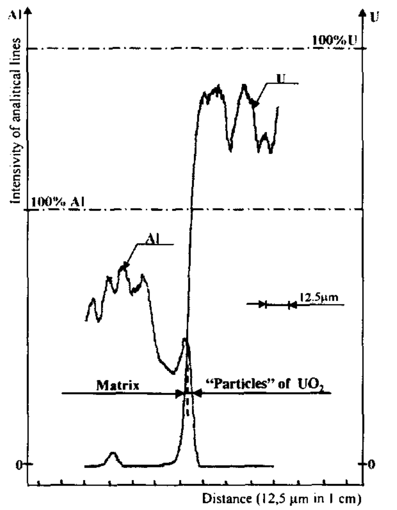

Fig. 1. Distribution of Al and U on the boundary of UO_2-matrix after testing at 775 K

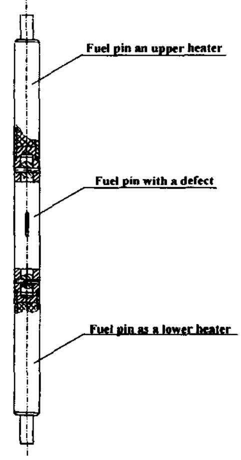

Fig. 2. Sectional fuel pin with an artificial defect

3. Tests of fuel in design-basis and in severe accident conditions

To study the cermet fuel pin behaviour under severe accident conditions the fuel pin mock-ups have been tested in an argon-steam mixture atmosphere at temperatures up to 2000 K. The failures of fuel pin mock-ups took place in the temperature range of 1620-1800 K as a result of total oxidation.

Hydrogen and fission product release in the temperature range up to 1475 K has been studied for the irradiated cermet fuel pins. Fuel pins were pre-irradiated in the AM research reactor (Obninsk, IPPE). No fission product release except ^3H and ^{85}Kr was identified. Only hydrogen was released from the fuel pin in the temperature below ~ 1300 K. ^3H release was noted between 1280-1440 K only.

4. Tests of unsealed cermet fuel pin

4.1 Autoclave tests

To determine the corrosion resistance of the cermet in the coolant, autoclave tests of fuel pin simulators and fuel composition specimens with unsealed cladding have been carried out under the following conditions: temperature 583 K and coolant pressure 10.6 MPa. No changes of the fuel pin external diameter were observed. The dimensions of the slots in specimens remained practically unchanged. Analysis of water in which these fuel pin simulators were tested showed an absence of uranium in it (the detection limit was 0.3 mg m^{-3}). Autoclave tests of cermet fuel composition (60% vol. UO_2 + 40% vol. silumin) without cladding during 10 to 200 hours were also carried out. Post-test examination showed that with the increase of contact time between specimens and coolant a non-significant increase of oxides in the matrix took place.

4.2 In-pile tests

A special channel-loop was built in the AM research reactor. The channel contained 3 fuel pins. Every fuel pin consisted of three parts (Fig. 2). The overall length of a fuel pin was 930 mm. The middle part of the fuel pin had a through slot in its cladding, with slot dimensions of 0.45 x 9.0 mm. The conditions of testing were the following: q_l =138 W cm^{-1}, outlet coolant temperature ~ 500-503 K, cladding temperature T_c = 569 K, fuel temperature $T^{max}{}_f$ = 617 K.

To study the release of fission products from a cermet fuel the following methods were applied:

- γ-spectrometry for direct (without sampling) measurement of radionuclides in the coolant;
- sampling of coolant with subsequent gamma-spectroscopic analysis of radionuclides;
- continuous measurement of exposure dose rate on the operational section of the test loop.

In-pile tests were carried out for two months. It was found that the main mechanism of fission product release from fuel into the coolant is direct recoil of fission products. The coolant *pH* affected the dissolution of the fuel and, accordingly, fission products release. Decrease of the coolant *pH* from 6 to 3 increased the dissolution, but its increase from 3 to 10 stopped fission product release through the pin slot. Post irradiation experiments did not reveal any significant increase of slot dimensions and fuel dissolution [3] for cermet fuel under normal operation conditions.

5. In-pile testing of cermet fuel

5.1 RIA (Reactivity Initiated Accident) testing

A RIA test was carried out in the IGR impulse reactor (Kazakhstan). Three fuel assemblies were tested. The energy deposited during a pulse was of 200 to 230 cal g^{-1} fuel. No fuel damage was observed. It was concluded that the failure threshold for cermet fuel pins should be higher than this value.

5.2 Long-term irradiation

Since November 1997, two cermet fuel assemblies are being irradiated in the MIR reactor (Dimitrovgrad, Russia). The first one consists of twelve 1000-mm length fuel pins. The second one consists of shorter fuel pins. The test conditions are those corresponding to the WWER-440 core under operation. The goal of the tests are to assess the lifetime, various fabrication processes and technologies, structural material and fuel burn up influence on the fuel pin performance. The achieved burn up is 35-55 MW d kg^{-1} of U depending on the fuel pin positioning in the MIR reactor.

5.3 Cladding creep study

In-pile creep for Zr+1%Nb claddings was studied in the BR-10 reactor (IPPE). The results are used for calculation analysis of cermet fuel thermal mechanics. It was found that the creep rate of Zr-1%Nb alloy does not depend on the neutron fluence in the range of 4 x 10^{19} – 6 x 10^{19} cm^{-2} and strongly depends on the stress in the range of $\sigma > 0.85\ \sigma_y(T,\Phi)$, where $\sigma_y(T,\Phi)$ is the yield stress under the temperature T and neutron fluence Φ.

6. Conclusion

Progress was made in the research of cermet fuel properties and its behaviour under normal operation and accidental conditions. The results of the experiments form the basis for further large scale testing of cermet fuel and fuel assemblies in operating WWER reactors.

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Appendix: Calculation modeling of the cermet pin behaviour

The following calculation results were obtained for a WWER-440 cermet fuel pin:

- Total inelastic strain of cladding material is $\varepsilon_i = 2.3$ % after 4-years operation. This value is ~ 3 times less than deformation ability of the Zr – 1 % Nb alloy under irradiation.
- Maximum hoop stress in the cladding $\sigma^{max} \approx 100$ MPa which is much less than the yield stress of irradiated Zr – 1 %Nb alloy.
- Power cycling leads to thermal-mechanical stressing on the cladding $\Delta\sigma^{max} = 85$ MPa. The permitted number of power cycling is much more than 10^4.
- It has been calculated that the radial stress component σ_r is always negative. So, the cladding is always pressed to the fuel composition and there is no danger of splitting the fuel and cladding.