Irradiation Test of MA-Containing Metal Fuel – Irradiation in the Phenix Reactor and Non-Destructive Examination of Low-Burnup Test Fuels –

Background

Recent nuclear fuel cycle development programs in Japan, the United States, France, Korea, etc., address the transmutation of minor actinides (MA: Np, Am and Cm) included in the spent fuel of light water reactors in order to reduce the radioactivity of nuclear wastes and ease the environmental burden caused by nuclear power production. Ahead of these programs, CRIEPI focused on the fact that the efficient MA transmutation can be attained by using metal fuel fast reactors, and planned the irradiation test to demonstrate the MA transmutation and the feasibility of MA-containing metal fuel. The test fuels loaded with the U-Pu-Zr alloys containing up to 5 wt.% MA were fabricated by 1994 in collaboration with the Institute for Transuranium of the European Commission. The irradiation of the test fuels in the French fast reactor, Phenix, was started in 2003 after the 9-year suspension of reactor operation.

Objectives

The objectives of the test are to demonstrate the integrity of the MA-containing metal fuel up to 10 at.% burnup, and to confirm that the irradiation performance of the MA-containing metal fuel is equivalent to that of the traditional, MA-free metal fuel by the non-destructive examination of the low-burnup test fuels.

Principal Results

1. Integrity of the MA-containing metal fuel up to 10 at.% burnup

The types and structures of the test fuels are shown in Fig. 1. The cast U-Pu-Zr alloy rods with a diameter of 4.9 mm and a length of 20 - 50 mm were stacked up to form a fuel column with a height of 485 cm in the test fuel. The MA-containing fuel stack section was placed at the axial position from 285 mm to 385 mm (100 mm in height). Three test fuel assemblies were constructed, and each assembly contained three different types of test fuels. The maximum fuel cladding temperature was 570℃ and the peak linear power rate was 330 W/cm during irradiation. Three assemblies were discharged from the reactor at the respective burnup levels: low (~2.5 at.%), medium (~7 at.%) and high (~10 at.%). No fuel failure was found up to 10 at.%, and the integrity of the MA-containing metal fuel up to 10 at.% burnup was demonstrated.

2. Non-destructive examination of the low-burnup test fuels

The results of non-destructive examination of the three low-burnup (2.5 at.%) test fuels did not indicate excessive cladding deformation and anomalous irradiation behavior due to MA addition, as follows.

(1) As shown in Fig. 2, the laser profilometry of fuel cladding diameter showed a maximum cladding diameter increase of 0.2 %, which is an acceptable level.

(2) The annular gap between the fuel stack and cladding was filled with sodium, which is called “bond sodium”. As cesium is dissolved in sodium, the gamma ray intensity from the cesium-137 produced by the fission is proportional to the amount of sodium. The gamma autoradiography of the cesium-137 (Fig. 3) indicated that approximately 80 % of the sodium that had been in the annular gap was expelled above the fuel stack due to the fuel alloy swelling. This sodium behavior is quantitatively consistent with the estimation based on the MA-free metal fuel behavior.

(3) The fuel stack axial deformations of 1.9 - 2.3 % for three low-burnup fuels were measured from the result of the gamma autoradiography of the ruthenium-106 (Fig. 3). These deformations are equivalent to those for the MA-free metal fuels.

Future Developments

Post-irradiation examination of the MA-containing metal fuel will be continued and the irradiation performance data will be accumulated in order to demonstrate the MA transmutation and the feasibility of MA-containing metal fuel.

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Reference

H. OHTA, et al., “Irradiation Experiment on Fast Reactor Metal Fuels Containing Minor Actinides up to 7at.% Burnup”, Proc. Int. Conf. Advanced Nuclear Fuel Cycles and Systems, Sep. 9-13.2007.

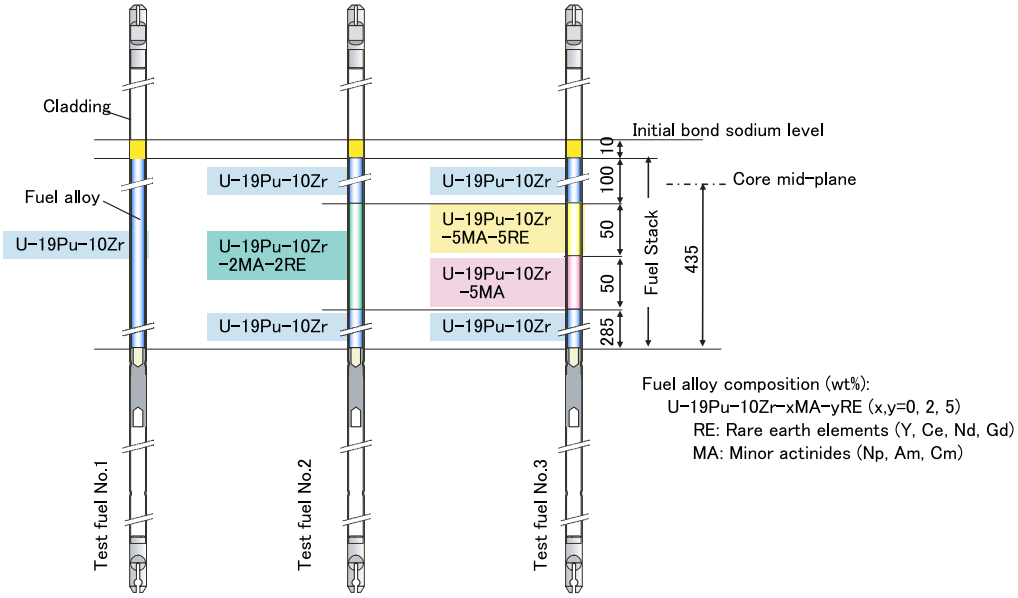

Fig.1 Types and structures of the test fuels

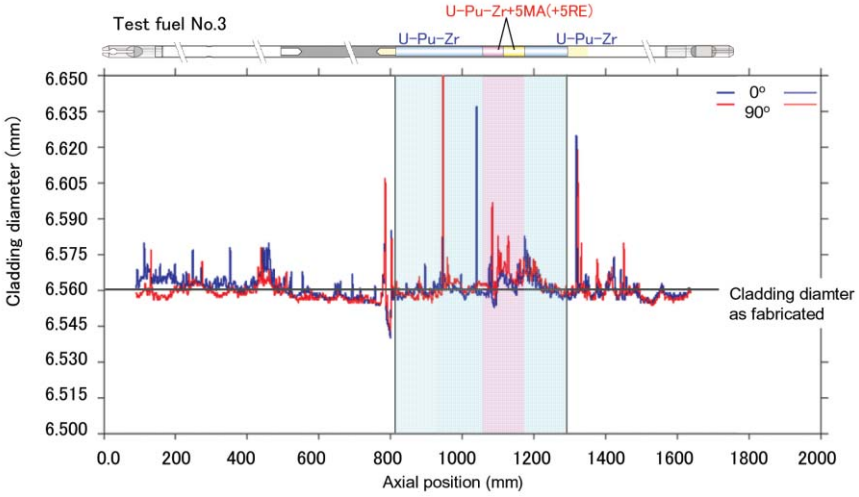

Fig.2 Laser profilometry of fuel cladding diameter

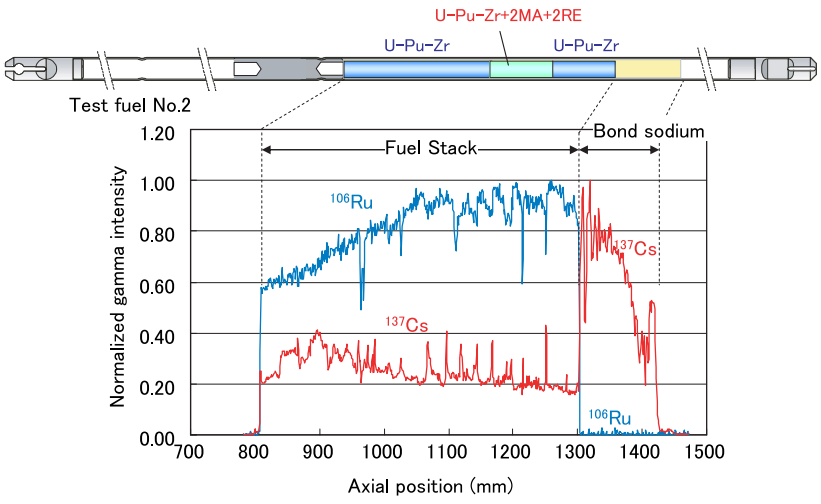

Fig.3 Gamma autoradiography of the cesium-137 and ruthenium-106