Irradiation of Metallic Fuels in BOR-60 Reactor and Non-destructive Post-Irradiation Examinations at 3 at.%

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1. Introduction

To verify the metallic fuel performance at a high burnup, irradiation experiments have been conducted with fuel rods made of U-10 wt.%Zr fuel slug and ferritic martensitic stainless steel claddings in the BOR-60 test reactor. BOR-60 test reactor is located in Russia Federation providing fast neutron spectrum. The objective of the irradiation is to achieve maximum burn-up of 10 at.% with maintaining the maximum cladding temperature in the rage of 650±20 °C. Seven fuel rods were irradiated up to 3 at. %, and among them, only three fuel rods were discharged, and non-destructive post irradiation examinations were conducted to confirm the integrity of the fuel rods.

2. Irradiation of Metallic Fuels in BOR-60

Fuel rod consists of U-Zr fuel slug and ferritic martensitic steel (HT9, FC92B and FC92N) cladding. Fig.1 shows cross section of fuel test rig and the axial dimension of the fuel rods.

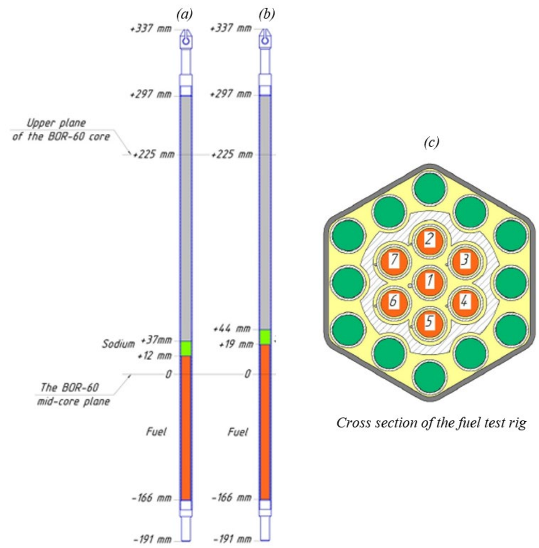

Fig. 1 Axial diemension of fuel rod (a) cetral fuel rod, (b)peripheral fuel rod and (c) cross section of fuel test rig.

The fuel slug length of central fuel rod is slightly shorter than peripheral fuel rods. Seven fuel rods were mounted in the fuel test rig and irradiated up to 3 at% burn-up. Among the rods, the peripheral fuel rods No.2(HT9), No.4(FC92N), and No.6(FC92B) were discharged from the reactor, and non-destructive examinations such as visual inspection, fuel rod outer diameter measurement, and gamma scanning were conducted for the discharged fuel rods.

3. Non-destructive Post Irradiation Examination

3.1 Visual inspection and diameter measurement

Visual examination and photographing of irradiated fuel rods were conducted with the use of optical system. The surface of the claddings is mainly reddish brown in the region of fuel slug and brownish outside the fuel slug region. Fig. 2 shows the photos of fuel rod No.2 at 0^{o} in azimuthal orientation. The surface of claddings seems to be in satisfactory condition, and there are no any abnormal things observed.

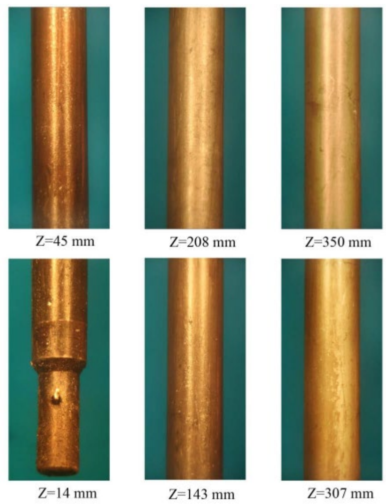

Fig. 2. Fuel rod No.2 appearance after removal of the wrapping wire, orientation 0^{o}.

Profilometry of fuel rods were measured by two inductive transducers (DB-810B). Diameter of the fuel rods were measured at every 1 mm axial increments at four azimuthal orientations (0^{o}, 45^{o}, 90^{o} and 135^{o}). The manufactured tolerance of cladding diameter is ±0.02 mm. Diameters of fuel rods No.4 and No.6 are within the manufacturing tolerances. However, the diameters of fuel rod No.2 throughout its length are positioned above the upper limit of the tolerances specified in the

KAERI's fuel fabrication report. This phenomenon is shown regardless of cladding temperature, i.e., in the regions of lower part and gas plenum of the fuel rod. Judging from this, the cladding of fuel rod No.2 was likely to be manufactured above the upper limit of diameter tolerances. Claddings of all the fuel rods under examination exhibit ovality (out-of-roundness) that is evidenced by measurements of the diameter in two perpendicular orientations 0º–90º and 45º–135º. Especially, the cladding of fuel rod No. 2 has more pronounced ovality around the upper end of its fuel slug. A metallography on its cross section taken through a subsequent destructive PIE can give more detailed information.

3.2 Gamma spectroscopy

Gamma scanning measurements of fuel rods were performed with the use of semi-automatic gamma scanning equipment with high purity Ge detector. The emitted intensity from isotopes was measured throughout the length of fuel rods using a collimator with a slot 1 mm wide. The intensity was recorded at displacement step of 1 mm. Time of exposure at every point of gamma scanning was 60 seconds. Fig. 3 shows the gamma scanning measurement data of the fuel rod No.2. Gamma ray energies from ^{137}Cs, ^{134}Cs, ^{95}Zr, ^{103}Ru and ^{106}Ru isotopes with energies of 661.7 keV, 795.9 keV, 724.2 keV, 497.1 keV and 621.8 keV, respectively, were recorded during the gamma scanning measurements. Distribution of a non-migrating fission product ^{95}Zr dissolved in uranium crystal lattice generally shows the distribution of power density and fuel burnup over the fuel rod length. No abnormalities in the ^{95}Zr distribution show that there is no mass transfer of the fuel components over the fuel column length. Distribution of ^{103}Ru and ^{106}Ru also shows the distribution of power density and fuel burnup over the fuel rod length, and evidences of no big inclusions of metallic fission products in fuel. There was axial migration of ^{134}Cs and ^{137}Cs from fuel areas into plenum regions as Cs is dissolved in the bond sodium. The majority of migrated cesium isotopes are above the fuel in sodium, and their small amount migrated into the lower area of the fuel slug.

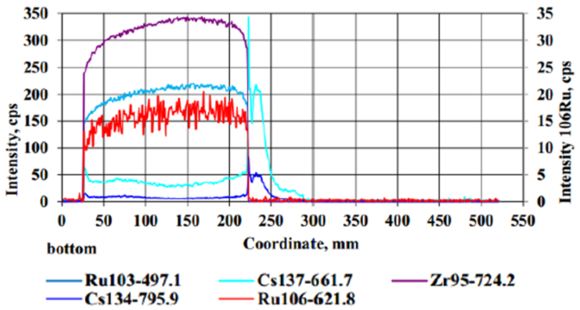

Fig. 3. Gamma ray intensities of fuel rod No. 2 in axial direction

4. Summary

Three irradiated U-10 wt.%Zr fuel rods were discharged, and non-destructive examinations were performed.

-Visual inspection and profilometry have shown no evidence of abnormal characteristics.

-Intensities of Cs isotopes indicate the Cs migration occurs in liquid state.

-It is concluded that all the fuel rods preserved their integrity during their irradiation testing.

Acknowledgement

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