Nuclear Engineering and Technology xxx (2017) 1–10

Available online at ScienceDirect

Nuclear Engineering and Technology

journal homepage: www.elsevier.com/locate/net

Original Article

A Concise Design for the Irradiation of U–10Zr Metallic Fuel at a Very Low Burnup

Haibing Guo, Wei Zhou, Yong Sun, Dazhi Qian, Jimin Ma, Jun Leng, Heyong Huo, and Shaohua Wang*

Institute of Nuclear Physics and Chemistry, China Academy of Engineering Physics, No. 64 Mianshan Road, Mianyang 621999, Sichuan Province, China

ARTICLE INFO

Article history:
Received 25 April 2016
Received in revised form
24 November 2016
Accepted 7 December 2016
Available online xxx

Keywords:
CMRR
Irradiation Design
Neutron Radiography
Swelling
U–10Zr

ABSTRACT

In order to investigate the swelling behavior and fuel–cladding interaction mechanism of U–10Zr alloy metallic fuel at very low burnup, an irradiation experiment was concisely designed and conducted on the China Mianyang Research Reactor. Two types of irradiation samples were designed for studying free swelling without restraint and the fuel–cladding interaction mechanism. A new bonding material, namely, pure aluminum powder, was used to fill the gap between the fuel slug and sample shell for reducing thermal resistance and allowing the expansion of the fuel slug. In this paper, the concise irradiation rig design is introduced, and the neutronic and thermal–hydraulic analyses, which were carried out mainly using MCNP (Monte Carlo N-Particle) and FLUENT codes, are presented. Out-of-pile tests were conducted prior to irradiation to verify the manufacturing quality and hydraulic performance of the rig. Nondestructive postirradiation examinations using cold neutron radiography technology were conducted to check fuel cladding integrity and swelling behavior. The results of the preliminary examinations confirmed the safety and effectiveness of the design.

1. Introduction

Metallic fuel has been in use for a long time in many nuclear reactors, such as the Experimental Breeder Reactor (EBR-II) and the Dounreay Fast Reactor, owing to its enhanced thermal conductivity, large fissile nuclei density, and negative temperature feedback coefficient [1]. In the United States Integral Fast Reactor program, a series of irradiation experiments were performed in the EBR-II, the Fast Flux Test Facility, and the Chicago Pile-5 reactors to investigate the irradiation behavior of U–Pu–Zr alloy fuel [2]. The results showed that the fuel exhibited as much as 7% axial elongation when it reached a burnup of 1.5 at.% [3]. In other experiments, fuel–cladding mechanical interaction (FCMI) and fuel–cladding chemical interaction (FCCI) were studied [4]. According to previous reports, swelling of metallic fuel was mainly attributable to the accumulation of fission gas and other products, and in the event of fuel–cladding contact,

* *Corresponding author.*
E-mail address: guohaib@caep.cn (H. Guo).
http://dx.doi.org/10.1016/j.net.2016.12.005

Please cite this article in press as: H. Guo et al., A Concise Design for the Irradiation of U–10Zr Metallic Fuel at a Very Low Burnup, Nuclear Engineering and Technology (2017), http://dx.doi.org/10.1016/j.net.2016.12.005

the FCMI and FCCI began to affect the integrity of the cladding.

Recently, in China, a fusion–fission hybrid energy reactor (FFR) concept was proposed [5]; U–10Zr alloy was chosen as the candidate fuel. This alloy uses abundant uranium, and elemental zirconium has a weight fraction of 10%. The most important characteristic of this concept was the use of light water as the coolant and moderator. As a result of the use of light water, the neutron energy spectrum in the subcritical blanket was more moderated than that in a fusion reactor or a fast reactor, but it was harder than that in a thermal reactor. This enabled the adjustment of the production and depletion rates of fissile nuclei to maintain equilibrium, which is necessary in order to reach a much higher burnup and nuclear resource utilization rate [6]. In the FFR concept, the metallic fuel elements were recycled through a melting–casting process [7], in which approximately 38 wt.% of fission products were eliminated whereas all actinides and most long-term radioactive isotopes were preserved. The irradiation behavior of the fuel was one of the most important study topics, and hence, the U–10Zr fuel irradiation (UZFI) experiment was scheduled in the China Mianyang Research Reactor (CMRR).

This paper presents the results of neutronic and thermal–hydraulic analyses performed for the design of this experiment in a concise manner. Several nondestructive examinations and test results are explained to show the quality of the capsules and the validity of the experiment design. The postirradiation integrity of the fuel pins was checked using cold neutron radiography (CNR) technology [8]. Other postirradiation examinations (PIEs), including metallic phase observation, exact shape profiling, and measurement of thermophysical properties, are under way. The described methodology, which is used for the design of a fuel irradiation experiment in a research reactor, is general and can be applied to other cases.

2. Irradiation capsule design

2.1. China Mianyang Research Reactor

The CMRR is a pool-type research reactor owned and operated by the Institute of Nuclear Physics and Chemistry, Mianyang, China. The light water in the pool is used as the coolant; it flows downward through the core. Plate-type fuel elements are used in the reactor, and it is operated under atmospheric conditions. The coolant temperatures at the inlet and outlet of the core are 35°C and 38°C, respectively [8].

In the UZFI experiment, two center symmetric positions in the core are used for irradiation. The experiment conforms to the criteria approved by the CMRR Safety Guard Committee. The main criteria are as follows: (1) the coolant mass flow in an irradiation rig should be less than 2.78 kg/s; (2) the maximum temperature of the irradiation sample surface should be lower than 84°C; (3) the reactivity introduced by irradiation rigs should be less than 0.5% $\Delta k/k$; and (4) any factor that may lead to melting of the internal materials should be avoided.

2.2. Design of the irradiation rig

The target maximum fission burnup of uranium nuclei in the U–10Zr fuel in the UZFI experiment was 0.5 at.%, including contributions from U-235 and U-238. In order to speed up the irradiation process and allow sufficient time for the radioactivity to decay and for the conducting of PIEs, a larger fission reaction rate and consequently a faster depletion rate of fissile nuclei are required. In the research reactor, which has a thermal neutron spectrum, these requirements can be met using U-235 enriched fuel slugs.

For a specific type of fuel with a given density and heavy metal fraction, the use of enriched uranium results in higher volumetric power density (VPD). The temperature rise between the outer surface and the centerline of a cylindrical body is proportional to the linear heat rate (LHR) and is independent of the radius. So, the thermal criteria will basically provide a restriction to the value of LHR, which is obtained by integrating VPD over the cross-sectional area. If VPD is expected to be high, in order to satisfy the thermal criteria, the fuel slugs were required to have small cross-sectional areas. Considering the convenience of mechanical fabrication and remote handling during PIEs, the dimensions of a fuel slug were set at $\varphi 5 \times 15$ mm, which is almost the lower limit of mechanical fabrication and remote handling.

Two types of irradiation samples and capsules were designed to investigate the free swelling behavior and fuel–cladding interaction, respectively. Schematic pictures of samples and their computer tomography and digital radiography (CT–DR) images are shown in Fig. 1. The main parameters are listed in Table 1, in which the smeared density is estimated by the ratio of the cross-sectional area of fuel slug to the total inner cross-sectional area of the cladding. The space occupied by the solid grain of aluminum powder is eliminated from the total inner cross-sectional area.

Molten salts (such as FLiBe and FLiNaK) or molten metals (such as sodium, lead, lead–bismuth eutectic, and lead–lithium) were not selected as the bonding material considering the low-temperature environment in the water-cooled core and the corrosion of other materials. Instead, pure aluminum powder was chosen to fill the gap between the inner slug and the sample shell. This material had two functions: (1) it acted as a thermal bond that enhanced thermal conduction, and (2) it worked as a buffer against fuel swelling.

The aluminum powder was loosely filled to 68% of the theoretical density with almost homogeneous porosity in the 1.5-mm-thick gap. The porosity decreases when an external force acts on the powder, because aluminum is soft and can even be extruded when the temperature approaches 168°C. This is because, when heating a metal, well before its melting point the recrystallization process will increase the ductility and decrease the resistance. The recrystallization temperature of aluminum is approximately 0.46 T_m, where T_m is the melting temperature of aluminum in units of K. Furthermore, aluminum powder poses little risk to the water-cooled reactor and has a lower radioactive dose after irradiation when cooled for a short time. Consequently, with properly designed