Performance Evaluation of Metallic Fuels Irradiated in BOR-60 to 7 at.%

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1. Introduction

Sodium-cooled fast reactor (SFR) has been researched as one of the most primary candidates of the generation IV reactors since it has advantages as enhanced reactor safety, fuel cycle economy, and environmental protection. These advantages are, to some extents, the direct results of using a metallic fuel because it has superior inherent safety and fuel-efficiency characterized by high thermal conductivity and high heavy metal density. In terms of this, metallic fuels have been considered as one of the most probable options for the advanced reactors such as SFR-type Small Modular Reactor (SMR). KAERI is developing a SFR-type advanced SMR named as SALUS (Small, Advanced, Long-cycles and Ultimate Safe SFR) using U-10Zr metallic fuel, and it is necessary to perform in-pile performance verification tests under fast neutron environment to ensure that the metallic fuels can retain their integrity to high burnup. Hence, metallic fuels made of U-10Zr fuel and ferritic martensitic stainless steel claddings (HT9, FC92B, and FC92N) were fabricated in KAERI, and irradiation tests have been performed in BOR-60 experimental fast reactor located in Russia. Recently, the irradiation tests to the maximum burnup of 7 at.% have been completed, and fuel performance analyses were performed using irradiation histories to ensure that the fuel rods retain their mechanical integrity. Here, these fuel performance analysis results will be presented.

2. Irradiation Tests Performed in BOR-60

Metallic fuels made of U-10Zr (19.75% U-235 enrichment) and ferritic martensitic stainless steel claddings (HT9, FC92B, and FC92N) were fabricated in KAERI, and irradiation tests were performed in BOR-60 located in Russia. HT9 wires were wrapped around the fuel rods and their diameter was 0.5 mm. The fuel test rig was designed to allow sodium coolant flow adjacent to the fuel rods. Fig. 1 shows linear heat generation rate (LHGR) and peak inner cladding temperature (PICT) of the fuel rods during the irradiation test. The maximum and time-averaged LHGR were 362 and 330 W/cm, respectively. The time-averaged maximum PICT was 642 °C. Effective full power day (EFPD) was 747 days. The maximum peak flux was $1.86 \cdot 10^{15}$ n/(cm$^2 \cdot$s).

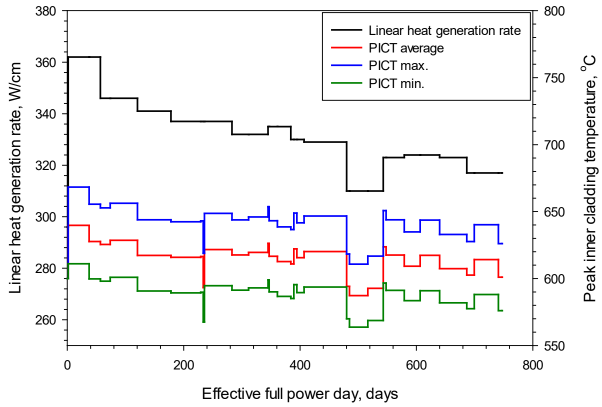

Fig. 1. Linear heat generation rate and peak inner cladding temperature of fuel rods irradiated in BOR-60.

3. Performance Evaluation and Conclusion

Integrity of three fuel rods (HT9, FC92B, and FC92N claddings) was evaluated using LIFE-METAL code, which has been developed jointly with Argonne National Laboratory (ANL) [1-2]. Creep correlations reflecting in-reactor creep test results of HT9, FC92B, and FC92N were developed and implemented in the LIFE-METAL [3]. Three criteria were adopted to evaluate the integrity of the fuel rods [4]. First, fuel maximum temperature should be lower than the solidus temperature of U-10Zr (1,237 °C). Second, the maximum strain of claddings should be lower than 1%. Third, cumulative damage fraction (CDF) should be lower than 0.05.

Based on the methodology and criteria explained above, cladding CDF, cladding strain, and fuel centerline temperature were calculated based on the maximum PICT and LHGR shown in Fig. 1. Fig.2, Fig.3, and Fig.4 show the calculated cladding CDF, cladding strain, and fuel centerline temperature cladding depending on burnup, respectively. The maximum CDF of the fuel rods with HT9, FC92B, and FC92N claddings were $1.81 \cdot 10^{-5}$, $1.61 \cdot 10^{-5}$, and $1.66 \cdot 10^{-5}$, respectively. The maximum cladding strains of the fuel rods with HT9, FC92B, and FC92N claddings were 0.16, 0.11, and 0.10, respectively. The maximum fuel temperatures of the fuel rods with HT9, FC92B, and FC92N claddings were 802, 799, and 802 °C, respectively. Even though we used the maximum PICT for the fuel performance analysis, the calculated values were much lower than the criteria. Hence it can be concluded that the fuel rods retain their mechanical integrity. Integrity of fuel rods was also confirmed through non-destructive examination for fuel rods which reached 7 at.% burn-up. Further extended irradiation

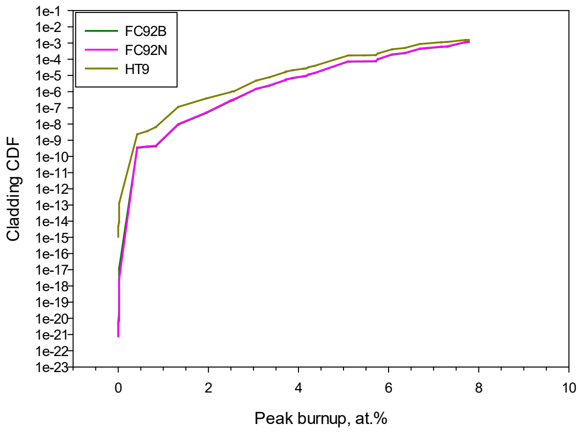

Fig. 2. Cladding CDF of the fuel rods irradiated in BOR-60 to 7 at.% burnup.

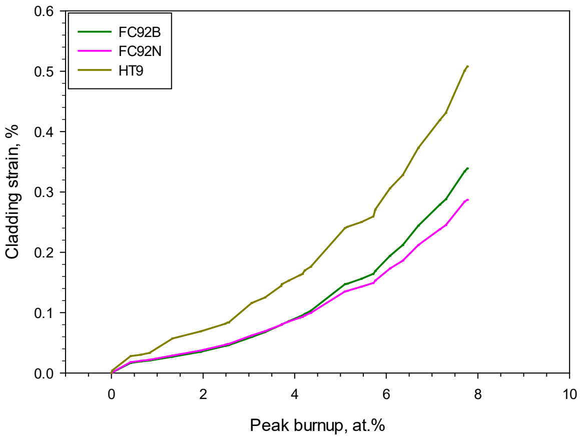

Fig. 3. Cladding strain of the fuel rods irradiated in BOR-60 to 7 at.% burnup.

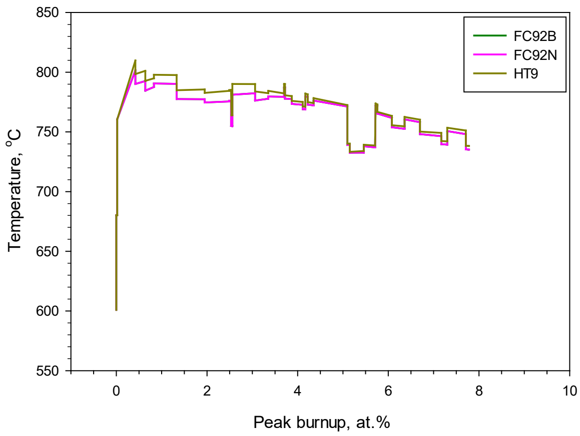

Fig. 4. Fuel maximum temperatures of the fuel rods irradiated in BOR-60 to 7 at.% burnup.

tests to the target burnup of 10 at.% will be completed by the end of 2024.

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