Analysis of the U.S. SFR Metallic Fuel Qualification Basis and Applicability Assessment of Korean BOR-60 Irradiation Data

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1. Introduction

In 2022, the U.S. Nuclear Regulatory Commission (NRC) established the Fuel Qualification Assessment Framework (FQAF) for advanced reactor fuels in NUREG-2246. Subsequently, NUREG/CR-7305 applied this framework to U-10Zr metallic fuel and derived a reference fuel design, manufacturing specifications, and safety and operating envelopes [1,2]. These reference criteria define the applicability of the U.S. legacy irradiation database and provide a basis for identifying additional testing and data needs for future fuel qualification.

This study analyzes the U.S. legacy irradiation database and qualification basis presented in NUREG/CR-7305 and quantitatively compares them with Korean BOR-60 irradiation data. The analysis assesses the extent to which the Korean data are applicable to the U.S. reference design and operating envelope and identifies additional PIE data and cladding-specific validation requirements for Korean fuel qualification.

2. Structure of the U.S. Fuel Qualification Basis

2.1 Scale of the Legacy Irradiation Database

According to Crawford et al. [3], more than 130,000 metallic fuel rods were irradiated in EBR-II and FFTF. Most were U-5Fs driver fuel rods (approximately 90,000 Mark-I/IA rods and more than 30,000 Mark-II rods), whereas the U-Zr and U-Pu-Zr database comprises more than 13,600 rods, representing approximately 10% of the total experience. Among these, approximately 1,387 rods were included in irradiation experiments involving U-10Zr/HT9 and related fuel–cladding combinations, accounting for less than 1.07% of the reported metallic fuel rods (Fig. 1).

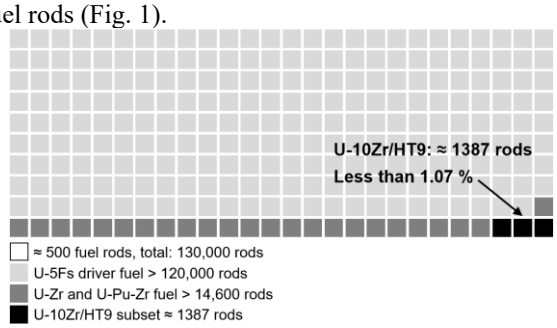


Fig. 1. Relative scale of the U-10Zr/HT9-related irradiation database within the U.S. metallic-fuel experience.

From this broader database, NUREG/CR-7305 identified 18 irradiation experiments comprising 1,262 U-10Zr/HT9 rods as directly relevant to the reference qualification case: 404 rods from 11 EBR-II experiments and 858 rods from 7 FFTF experiments [2]. Only four cladding breaches were reported among these 1,262 rods, which is insufficient to statistically demonstrate the reliability target of no more than one breach per core loading, equivalent to approximately 6×10^{-5} to 8×10^{-5} per rod.

2.2 Scale of the Transient Test Database

The transient test database for U-10Zr fuel with HT9 cladding is sparse. The TREAT M-series examined 15 rods in 6 experiments, but only 1 rod used U-10Zr/HT9 at 2.9 at.% burnup. Similarly, only 1 of the 7 out-of-pile FM-series tests used U-10Zr/HT9 at 3 at.% burnup [2]. The remaining tests used U-5Fs/316SS or U-19Pu-10Zr/D9, which differ from the reference design in fuel composition or cladding material. These tests help identify failure mechanisms common to metallic fuels but do not directly demonstrate the transient behavior of U-10Zr/HT9.

Appendix A of NUREG/CR-7305 identifies the same limitation. The normal-operation basis is supported by irradiation data accumulated up to 10 at.% burnup, whereas the small transient test database may result in unnecessarily large safety margins [2].

2.3 Use of PIE Data in TerraPower Licensing Documents

The size of an irradiation database alone does not show how its individual data are used to support fuel qualification. To examine how irradiation data are incorporated into the fuel qualification basis, the TerraPower Fuel and Control Assembly Qualification Topical Report and the corresponding NRC Safety Evaluation were reviewed [4].

TerraPower selected 38 assemblies from the EBR-II and FFTF irradiation programs: 34 EBR-II assemblies and 4 FFTF MFF-series assemblies. Each assembly was characterized in terms of its operating history, design and manufacturing specifications, available PIE data, and

qualification priority [4]. The PIE measurements were then linked to specific predictions of the fuel-performance models, as summarized in Table I.

Table I: Validation targets of legacy PIE data as identified by TerraPower

PIE measurement	Validation target
Profilometry (contact / laser)	Cladding strain prediction, primarily for HT9 cladding
Neutron radiography (NRAD)	Radial and axial fuel swelling, primarily for U-10Zr fuel
Fission gas release	Fission gas release and plenum pressurization
Metallography	Cladding wastage prediction

The NRC accepted the selected assemblies as a source of qualification data for Natrium Type 1 fuel, while requiring that each pairing of an assembly with a specific model or behavior be justified individually [4].

3. Position of the Korean Irradiation Data and Remaining Gaps

The Korean BOR-60 irradiation inventory consists of 13 metallic fuel pins (12 irradiated and 1 unirradiated HT9) and 357 cladding specimens. The claddings of the fuel pins are HT9 for 3 pins, FC92N for 8 pins, and Cr-coated FC92N for 1 pin, with maximum burnups of 10 at.% for the FC92 series and 7 at.% for HT9. The cladding specimens have been shipped to Korea, whereas the fuel pins remain in interim storage at RIAR [5].

3.1 Fuel Performance Envelope and Korean Irradiation Conditions

The fuel performance envelope given in NUREG/CR-7305 §2.3 comprises a peak fuel rod burnup of 10 at.%, a peak BOL linear heat generation rate of 40–55 kW/m, a peak fuel-cladding-interface temperature of 650 °C, and a total radial strain and deformation of 2 % [2]. Korean BOR-60 irradiation data reached the envelope limits for interface temperature. However, the burnup limit of 10 at.% was reached only by the FC92-series rods, whereas the HT9-clad rods corresponding to the reference fuel system reached 7 at.%. Thus, no Korean rod both matches the reference U-10Zr/HT9 combination and reaches the envelope burnup limit. The linear heat generation rate fell below the 40–55 kW/m band, and total radial strain at 10 at.% burnup has not been measured (Fig. 2).

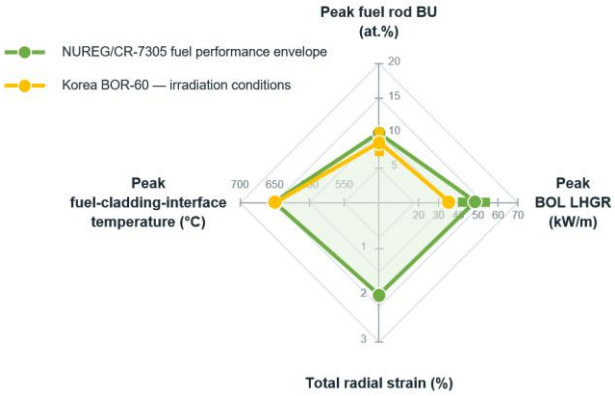


Fig. 2. Korean BOR-60 irradiation conditions compared with the NUREG/CR-7305 fuel performance envelope.

The fuel-cladding interface temperature was 650 ± 20 °C. The 650 °C limit is intended to restrict thermal creep of HT9 and FCCI and to prevent fuel-cladding liquefaction; observations from the EBR-II X447 experiment, in which cladding breaches occurred at 660 °C, contributed to the basis for this limit [2]. The Korean irradiation temperature therefore overlaps this limiting region.

The peak BOL linear heat generation rate was 35 ± 3 kW/m, below the lower envelope value of 40 kW/m. Total radial strain has not been measured at the limiting condition: profilometry of 3 rods withdrawn at 2.82 at.% yielded diametral strains of 0.050–0.153 %, and for rods irradiated to 7 at.% only LIFE-METAL predictions of 0.170–0.508 % are reported [5]. Because the envelope covers operation up to 10 at.%, no strain measurement exists at that condition.

3.2 Fuel Design and Manufacturing Specification

The manufacturing specification of the reference fuel design in NUREG/CR-7305 is compared with the as-designed dimensions of the Korean BOR-60 fuel rods [6] in Fig. 3. The reference values are a cladding outer diameter of 6.9 mm, a cladding thickness of 0.6 mm, a fuel outer diameter of 5.0 mm, a fuel height of 91 cm, a plenum-to-fuel volume ratio of 1.4, and a fuel slug smeared density of 75 % [2]; the corresponding values for the Korean rods are 7.4 mm, 0.50 mm, 5.54 mm, 18.5 cm, 2.0, and 75 %. These are the values for the BOR-60 test rods, and their representativeness with respect to the Korean prototype design must be assessed separately.

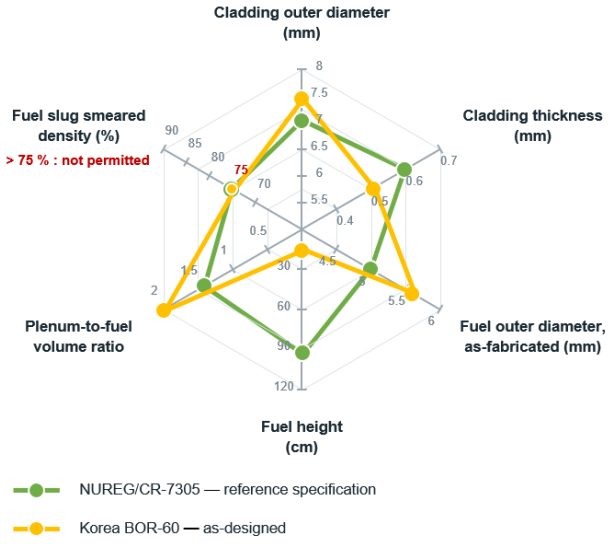


Fig. 3. As-designed dimensions of the Korean BOR-60 fuel rods compared with the NUREG/CR-7305 reference fuel design.

NUREG/CR-7305 identifies the maximum smeared density of 75 % as a parameter that must be adhered to, since exceeding it significantly influences cladding strain and fuel column growth. The dimensions, in contrast, are described not as rigid design requirements but as an example of a fuel system, and different tolerance values on key dimensions may be justified by a sensitivity analysis or new experience [2]. The Korean rods have a smeared density of 75 %, which meets this requirement. The differences in cladding outer diameter, cladding thickness, fuel outer diameter, and fuel height therefore require justification by sensitivity analysis or experience.

The plenum-to-fuel volume ratio of the Korean rods is 2.0, larger than the reference value of 1.4. NUREG/CR-7305 does not prescribe a fixed value for this ratio, stating instead that the plenum must accommodate the released fission gas and keep cladding creep strain within acceptable limits, and that acceptable ratios therefore vary with the intended terminal burnup and cladding temperature [2]. The larger value provides greater margin for gas accommodation, but justification against the intended burnup and temperature conditions is still required.

Fabrication tolerances and inspection criteria for these rods are reported in the open literature, including a fuel slug diameter tolerance within 0.1 mm and a total O·C·N·Si impurity limit of 2,000 ppm set on the basis of EBR-II fuel [6]. A numerical limit of 2,000 ppm is also specified in NUREG/CR-7305, although the listed impurity elements differ [2]. Tolerances on the cladding dimensions were not identified in the sources reviewed.

3.3 Availability of PIE Data by Validation Target

The Korean data are organized in Table II according to the validation targets identified by TerraPower (Table I). In Korea, diametral cladding strain was measured by profilometry, and axial fuel swelling was evaluated from the position of the Cs peak in gamma scanning [5]. The latter differs in method from the neutron radiography used in the TerraPower TR for fuel swelling and redistribution, but it addresses part of the same validation target in that it establishes the axial dimensional change of the fuel slug. Both measurements, however, are limited to the 3 rods withdrawn at 2.82 at.%, and are not data at the envelope limit of 10 at.%.

Table II: Availability of PIE Data by Validation Target

Validation target	Korean data
Cladding strain	Measured on 3 rods withdrawn at 2.82 at.% [5]
Axial fuel swelling	Measured on 3 rods withdrawn at 2.82 at.% [5]
Radial fuel swelling and constituent redistribution	Not available (dedicated facility required)
Fission gas behavior	Not available (destructive PIE)
FCCI and cladding wastage	Not available (destructive PIE)

No measurements are available for the remaining three targets: radial fuel swelling and constituent redistribution, fission gas behavior, and cladding wastage due to FCCI. Of these, fission gas data are used to evaluate plenum pressure and the resulting cladding deformation, and metallography data are used to predict wastage and to evaluate the allowable limit on remaining cladding thickness. Both require destructive post-irradiation examination and are therefore the priority items to be obtained.

4. Applicability of the U.S. Legacy Database to Different Cladding Materials

Korea completed the irradiation of 12 U-10Zr fuel rods in BOR-60 between 2014 and 2020. Because irradiation testing is the most time-consuming element of the qualification process, these data form the basis for qualification in Korea. The scale is limited, however, and the data cannot by themselves constitute a qualification basis; a link to the U.S. legacy data is presupposed.

Using the U.S. legacy data in Korean fuel qualification requires that the manufacturing specification of the Korean fuel be consistent with the NUREG/CR-7305 reference design, and that a domestic fuel performance code be available to reproduce it. The Korean BOR-60 rods adopt U-10Zr fuel, a smeared density of 75 %, and sodium bonding.

The 3 U-10Zr/HT9 rods irradiated in Korea share the same fuel and cladding as the U.S. reference combination and can therefore serve as the baseline for code validation linking the U.S. legacy data to the Korean irradiation data. However, this baseline covers burnup only up to 7 at.%, below the 10 at.% limit of the reference envelope. The basic predictive capability of the code would first be validated against the U.S. data, then compared with the PIE results of the Korean HT9 rods, and the validation range subsequently extended to the

nine FC92-series rods irradiated in the same rig, forming a staged verification and validation program.

5. Conclusions

The Korean BOR-60 irradiation data provide a basis on which the U.S. legacy data can be brought into fuel qualification in Korea. The manufacturing specification should first be shown to be consistent with the reference design, and the domestic fuel performance code validated using the U.S. U-10Zr/HT9 data together with the PIE results of the Korean HT9 rods. The validation range can then be extended to the domestically developed cladding through destructive PIE of the FC92-series rods, and the data gaps thereby identified used to define additional irradiation testing and the domestic design and operating envelope.

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