Comparison of Calculated and Measured Neutron Fluence in Fuel/Cladding Irradiation Experiments in HFIR

Ronald J. Ellis

Oak Ridge National Laboratory, P.O. Box 2008, Oak Ridge, TN 37831-6172; ellisrj@ornl.gov

INTRODUCTION

A recently-designed thermal neutron irradiation facility has been used for a first series of irradiations of PWR fuel pellets in the high flux isotope reactor (HFIR) at Oak Ridge National Laboratory. Since June 2010, irradiations of PWR fuel pellets made of UN or UO_2, clad in SiC, have been ongoing [1-3] in the outer small VXF sites in the beryllium reflector region of the HFIR, as seen in Fig. 1.

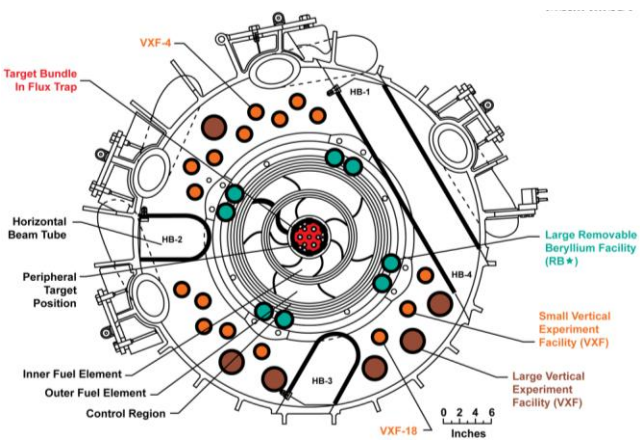

Fig. 1. Schematic view of HFIR and its irradiation sites.

HFIR [4] is a versatile, 85 MW isotope production and test reactor with the capability and facilities for performing a wide variety of irradiation experiments. HFIR is a beryllium-reflected, light-water-cooled and -moderated, flux-trap type reactor that uses highly enriched (in ^{235}U) uranium (HEU) as the fuel. The reactor core consists of a series of concentric annular regions, each about 2 ft (0.61 m) high. A 5-in. (12.70-cm)-diam hole, referred to as the flux trap, forms the center of the core.

The fuel region is composed of two concentric fuel elements made up of many involute-shaped fuel plates: an inner element that contains 171 fuel plates, and an outer element that contains 369 fuel plates. The fuel plates are curved in the shape of an involute, which provides constant coolant channel width between plates. The fuel (U_3O_8-Al cermet) is nonuniformly distributed along the arc of the involute to minimize the radial peak-to-average power density ratio [2]. A burnable poison (B_4C) is included in the inner fuel element primarily to reduce the negative reactivity requirements of the reactor control plates. A typical HEU core loading in HFIR is 9.4 kg of ^{235}U and 2.8 g of ^{10}B. The thermal neutron flux in the flux trap region can exceed 2.5×10^{15} n/cm^2·s while the fast flux in this region exceeds 1×10^{15} n/cm^2·s.

The inner and outer fuel elements are in turn surrounded by a concentric ring of beryllium reflector approximately 1 ft (0.30 m) thick. The beryllium reflector consists of three regions: the removable reflector, the semi-permanent reflector, and the permanent reflector. It is surrounded by a water reflector of effectively infinite thickness. In the axial direction, the reactor is reflected by water above and below the reactor.

The irradiation facilities (see Fig. 2), one for UN and the other for UO_2 pellets, utilize a thin cylindrical hafnium shield approximately 4 cm in diameter surrounding the facility basket to reduce the thermal neutron flux sufficiently such that the linear power rating in the irradiated fuel pins will be similar to PWR operating conditions. The facilities each contain nine fuel pins, each comprising 10 fuel pellets, arranged as if three fuel rods.

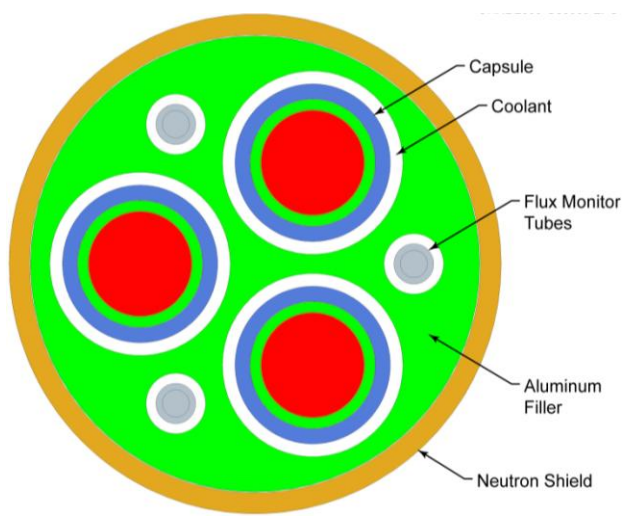

Fig. 2. Horizontal cross-sectional view of the thermal neutron irradiation facility.

Figure 3 is a vertical view of an irradiation facility that corresponds to the horizontal view in Fig. 2. The axial placement of the three fuel pins in each of the three rod locations is seen. The fluence monitor wires are positioned axially at the midpoint of each of the fuel pins.

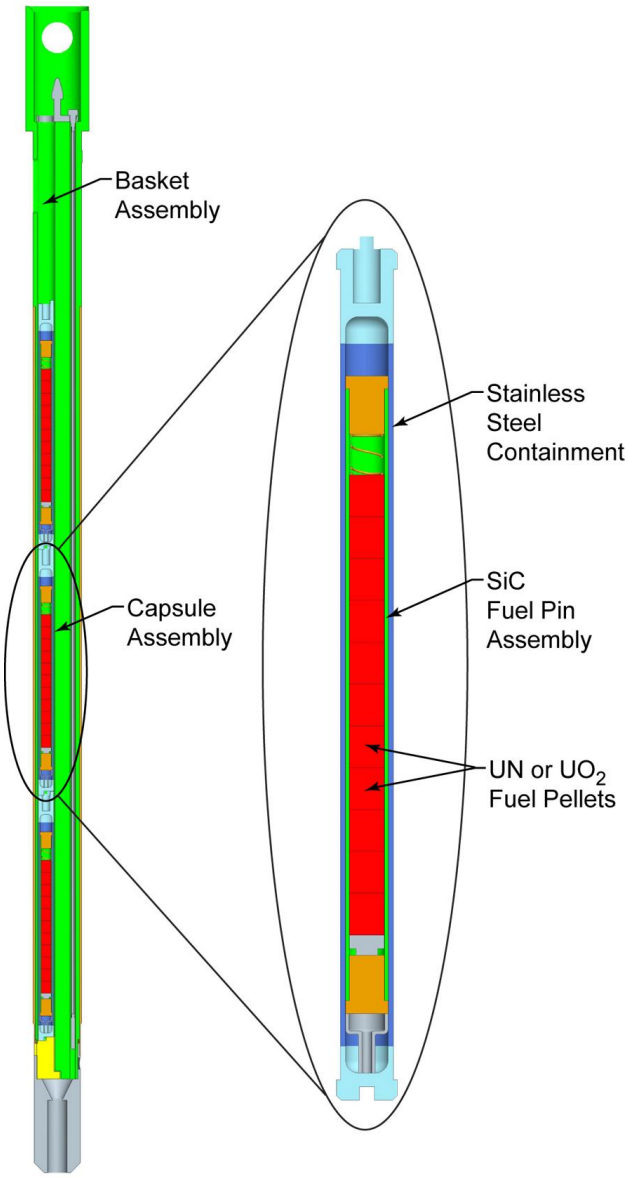

Fig. 3. Vertical view of the thermal neutron irradiation facility with a close-up of one of the nine capsule assemblies.

DESCRIPTION OF THE ACTUAL WORK

The irradiation facilities are equipped with three flux monitor tubes. In each of these tubes, thin wires of aluminum with about 0.1 wt % ^{59}Co are inserted. Axially, there are three monitor locations: top, middle, and bottom. Experimentally, a measure of the total neutron fluence experienced at the nine locations is the amount (activity) of ^{60}Co that is produced under the neutron irradiation. A requirement of the continued irradiation of the facilities is that neutron flux and fluence levels (as represented by ^{60}Co activities) as measured experimentally and as calculated through physics models compare well (within about 20%) at key times during the multicycle HFIR irradiation campaigns, including after the first and sixth irradiation cycles, for each fuel type..

In this paper, the assessment of the ^{60}Co activities is reported for the UN fuel pellet irradiation facility after its first cycle in HFIR.

The experimental ^{60}Co activities were determined [5] for the nine fluence monitor wires by careful gamma spectrometric counting of the two main gammas from ^{60}Co, normalized to a known standard ^{60}Co source.

The ^{60}Co activities were calculated through physics models using MCNP [6] and SCALE [7]. This work uses an input model based on a detailed HFIR MCNP reference model [8] (Fig. 4) to calculate flux levels at the appropriate locations. These flux levels are used in SCALE/TRITON with the NEWT-based t-depl depletion mode, and in SCALE applications of the COUPLE and ORIGEN-S modules to calculate the levels of ^{60}Co generated in the monitor wires.

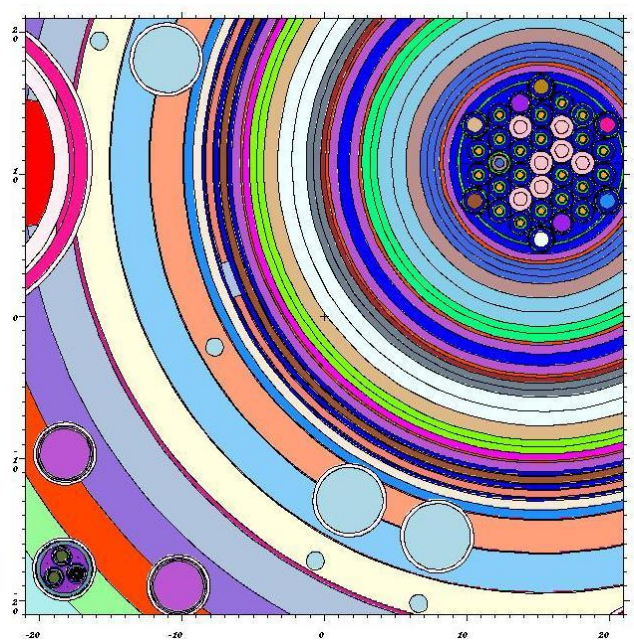

Fig. 4. Part of the MCNP model of HFIR.

The irradiation facility is seen in the lower left of Fig. 4; two of the fuel pins are in the preferred orientation, equidistant from, and facing towards, the center of HFIR. Consequently, the flux monitor wires at each axial location are arranged such that one is facing forwards, and two are at a slightly greater radial distance from the center of HFIR. The flux monitor wires that are closer to the center will experience a greater flux level and thus will have a higher level of ^{60}Co generated. This is evident in Table 1 which shows a comparison of the measured and calculated ^{60}Co activity as the monitor wires of series 1 (B, M, T) have higher levels of experimentally measured ^{60}Co than the other two series of monitor wires.

Table I. Cobalt-60 Activities and Neutron Flux Calculations for the Irradiation Facility with UN fuel

Monitor Wire (ID)	^{60}Co Activity (mCi/g)		(M - C)/M (%)	Calculated EOC Neutron Flux/10^{13} (n/cm^2/s)	
	Measured, M	Calculated, C		Thermal Flux	Fast Flux
1B	2.40	2.45±0.22	-2	3.32±0.25	2.97±0.22
1M	4.18	3.66±0.33	12	4.97±0.37	5.00±0.37
1T	2.58	2.32±0.22	10	3.33±0.25	3.24±0.24
2B	1.63	1.71±0.15	-5	2.33±0.17	2.34±0.18
2M	2.72	2.58±0.23	5	3.24±0.24	3.75±0.28
2T	1.70	1.48±0.13	13	2.21±0.17	2.38±0.18
3B	1.50	1.66±0.15	-11	2.25±0.17	2.38±0.17
3M	2.42	2.63±0.23	-9	3.45±0.26	3.94±0.30
3T	1.48	1.66±0.13	-12	2.10±0.16	2.32±0.16

RESULTS AND DISCUSSIONS

Table I lists in column 2 the measured ^{60}Co activities for the nine flux monitor wire segments (as identified in the first column). The monitor wire locations are identified by position 1 (facing core center), position 2 which is to the left, and position 3 which is to the right. Three axial positions are indicated for each wire location, the midpoint of the three fuel axial positions: top, middle, and bottom. Column 3 lists the ^{60}Co activities calculated with the physics models for SCALE using MCNP flux determinations. Column 4 tabulates the percentage differences between the experimentally measured and the calculated ^{60}Co activities.

For information, the calculated thermal and fast neutron flux levels at the end of cycle (EOC) at the various flux monitor wire locations are listed.

The agreement between experimental measurements and calculated levels of ^{60}Co activities, as summarized in Table I, is good, with all comparisons within 13% and with the average of the absolute values of the differences being about 8.5%. Agreement within 20% is normally expected.

ACKNOWLEDGMENTS

These irradiation experiments began as research sponsored by the Department of Energy Office of Nuclear Energy and the Laboratory Directed Research and Development Program of Oak Ridge National Laboratory, managed by UT-Battelle, LLC for the U.S. Department of Energy. The current work was performed under the DOE-NE Light Water Reactor Sustainability program.

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