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ABSTRACT

The Neutron Radiography (NRAD) reactor is a 250-kW TRIGA-(Training, Research, Isotope Production, General Atomics)-conversion-type reactor at the Idaho National Laboratory; it is primarily used for neutron radiography analysis of irradiated and unirradiated fuels and materials. The NRAD reactor was converted from HEU to LEU fuel with 60 fuel elements and brought critical on March 31, 2010. This configuration of the NRAD reactor has been evaluated as an acceptable benchmark experiment and is available in the 2011 editions of the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (ICSBEP Handbook) and the *International Handbook of Evaluated Reactor Physics Benchmark Experiments* (IRPhEP Handbook). Significant effort went into precisely characterizing all aspects of the reactor core dimensions and material properties; detailed analyses of reactor parameters minimized experimental uncertainties. The largest contributors to the total benchmark uncertainty were the ^{234}U, ^{236}U, Er, and Hf content in the fuel; the manganese content in the stainless steel cladding; and the unknown level of water saturation in the graphite reflector blocks. A simplified benchmark model of the NRAD reactor was prepared with a keff of 1.0012 ± 0.0029 (1σ). Monte Carlo calculations with MCNP5 and KENO-VI and various neutron cross section libraries were performed and compared with the benchmark eigenvalue for the 60-fuel-element core configuration; all calculated eigenvalues are between 0.3 and 0.8% greater than the benchmark value. Benchmark evaluations of the NRAD reactor are beneficial in understanding biases and uncertainties affecting criticality safety analyses of storage, handling, or transportation applications with LEU-Er-Zr-H fuel.

Key Words: Benchmark, ICSBEP, IRPhEP, NRAD, TRIGA

1 INTRODUCTION

The neutron radiography (NRAD) reactor is a 250 kW TRIGA®-(Training, Research, Isotope-Production, General Atomics)-conversion-type reactor located at the Idaho National Laboratory (INL). It is equipped with two beam tubes with separate radiography stations for the performance of neutron radiography irradiation on small test components. It is primarily used for analysis of both irradiated and unirradiated fuels and materials. Typical applications for examining the internal features of fuel elements and assemblies include fuel pellet separations, fuel central-void formation, pellet cracking, evidence of fuel melting, and material integrity under normal and extreme conditions.

The NRAD reactor was originally located at the Puerto Rico Nuclear Center (PRNC) and later converted to a TRIGA-FLIP-(Fuel Life Improvement Program)-fueled system (70% ^{235}U in UZrH matrix) in 1971. The 2-MW research reactor was closed in 1976 and moved to Idaho in 1977. As part of the Reduced Enrichment for Research and Test Reactor (RERTR) Program, in support of the Global Threat Reduction Initiative (GTRI), the NRAD reactor was again converted, but this time from the highly enriched uranium (HEU) FLIP fuel to low enriched uranium (LEU) fuel (<20% ^{235}U in UErZrH matrix). The FLIP-fueled core was defueled by August 26, 2009, with refueling commenced September 17, 2009, and the initial approach to critical on March 9, 2010. Initial criticality was achieved with 56 fuel elements on March 19, 2010, and the operational core was established with 60 fuel elements on March 31, 2010. Start-up testing for the HEU/LEU fuel conversion was performed over the period of March 9 to June 7, 2010; measurements included the neutron multiplication in the approach to initial criticality, determination of control rod worths, excess reactivity and shutdown margins, and calorimetric power calibrations up to 250 kW. Additional measurements were performed to evaluate the worth of some of the graphite reflector blocks used in the core and a dry tube used for experiment irradiations.

A benchmark of the 60-fuel-element core critical configuration was developed to support start-up testing and subsequent reactor analyses [1]. It has been evaluated as an acceptable benchmark experiment and is available in the 2011 editions of the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (ICSBEP Handbook) [2] and the *International Handbook of Evaluated Reactor Physics Benchmark Experiments* (IRPhEP Handbook) [3]. A benchmark of the annular TRIGA Mark II reactor in Slovenia was previously evaluated [4]; this reactor, however, contains UZrH fuel with no erbium burnable poison. Benchmark evaluations of the NRAD reactor are beneficial in understanding biases and uncertainties affecting criticality safety analyses of storage, handling, or transportation applications with UErZrH fuel.

1.1 Overview of NRAD Reactor Configuration

The NRAD reactor (Fig. 1) is water-moderated, heterogeneous, solid-fuel, tank-type research reactor. It is comprised of fuel in three- and four-element clusters that can be arranged according to reactivity requirements. The grid plate consists of 36 holes, on a 6 × 6 rectangular pattern, that mate with the end fittings of the fuel cluster assemblies. The operational core configuration (Fig. 2) contains 60 fuel elements, two water-followed shim control rods, and one water-followed regulating rod. A water hole is provided as an experimental irradiation position. There are two neutron beam tubes on the East and North sides of the core. Peripheral positions in the grid contain nuclear-grade-graphite reflector block assemblies.

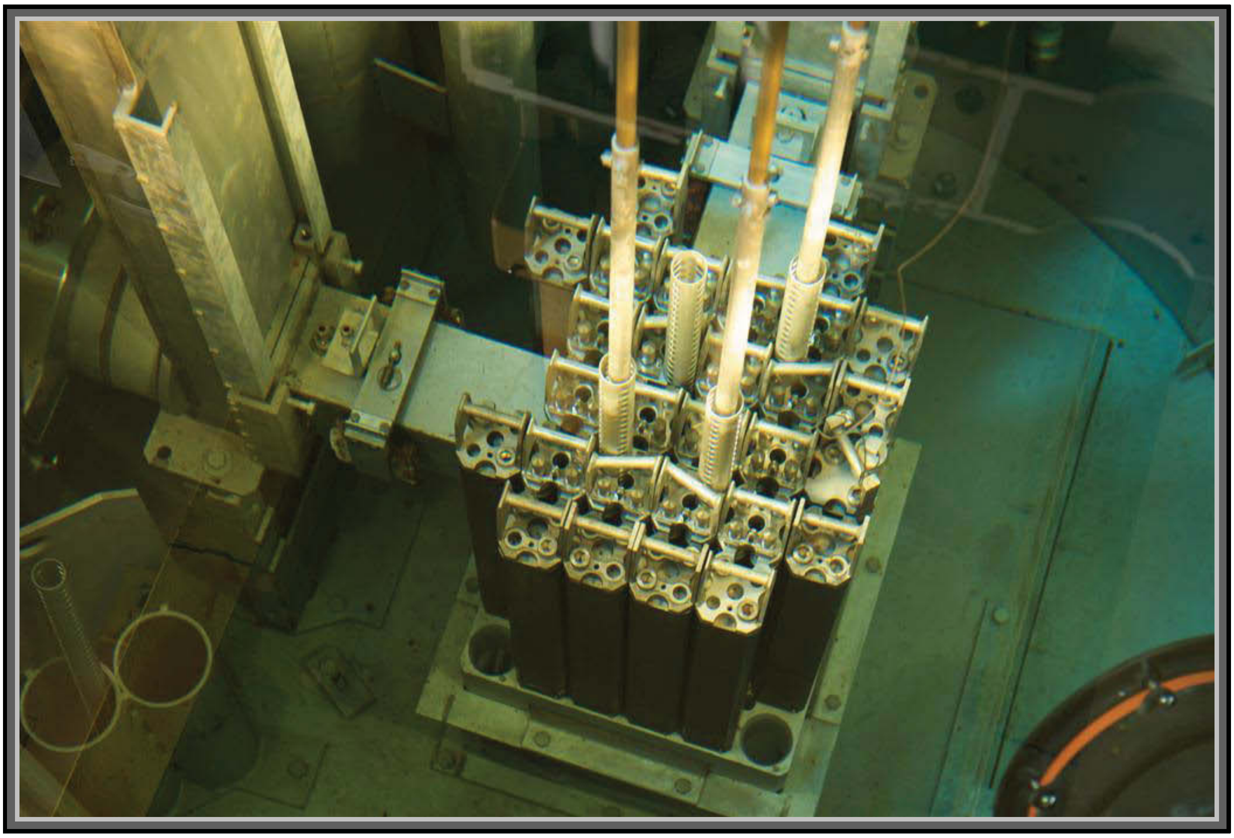

Figure 1. In-Tank View of the NRAD Reactor Core.

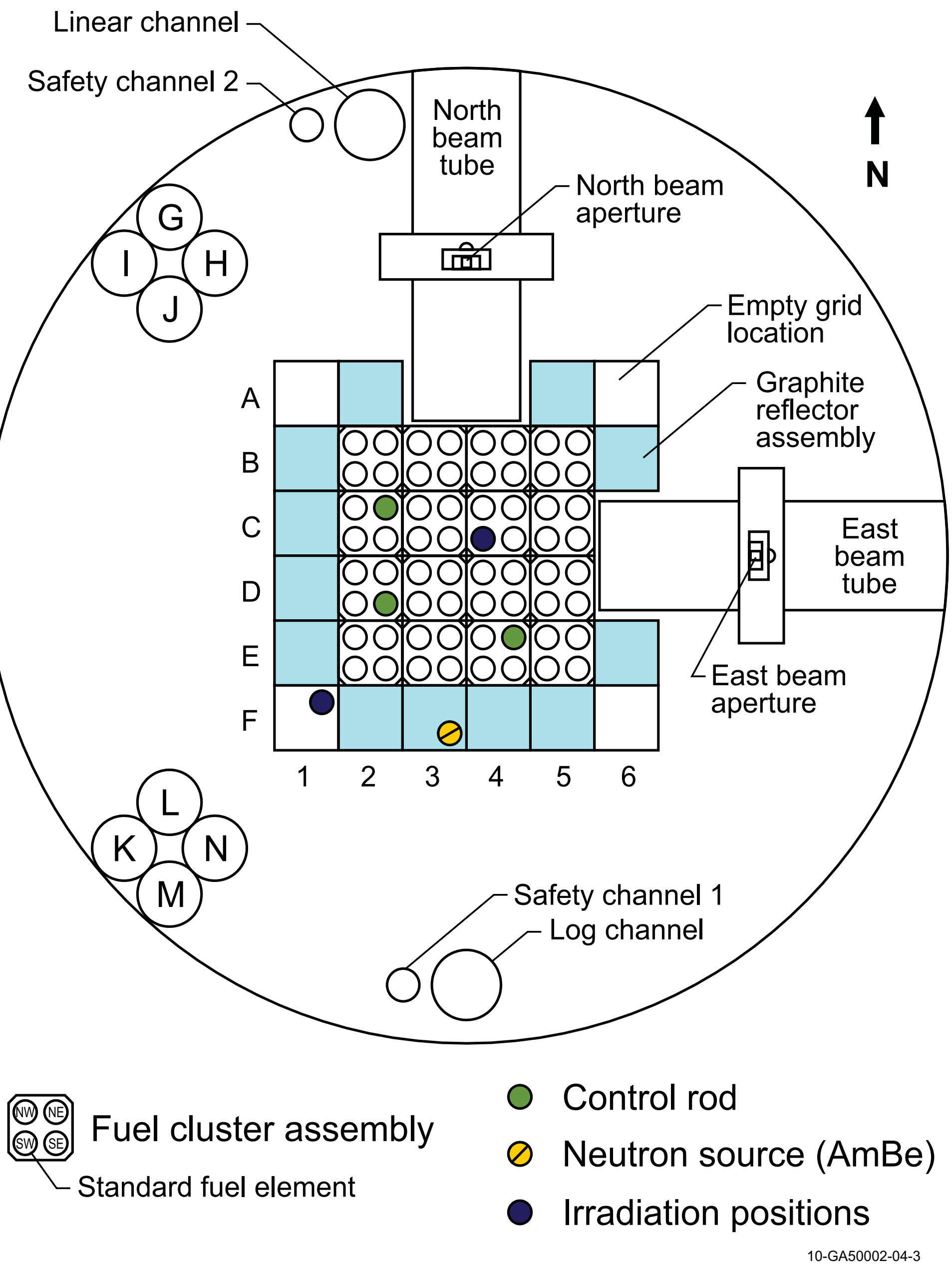

Figure 2. NRAD LEU Core Configuration.

The NRAD LEU fuel is a mixture of uranium, erbium, and zirconium hydride. The uranium is enriched to approximately 19.75% in ^{235}U and is approximately 30 wt.% of the fuel. There is a uniform dispersion of 0.9 wt.% natural erbium that is used as a burnable poison to offset initial reactivity of the fresh fuel and contribute to the prompt negative temperature coefficient. The H/Zr ratio is ~1.6. The 15 in. of fuel pellets have a zirconium spine and are clad with two 3.43 in. graphite axial reflectors and a thin molybdenum poison disc in stainless steel (see Fig. 3). A summary of the as-built fuel data is provided in Table I.

The B_4C control rods are also 15 in. in length and contained within aluminum tubes; typically the two shim rods (clusters C-2 and D-2) are fully withdrawn and reactivity is adjusted with the regulating rod (cluster E-4). The graphite blocks are unclad and are 25.9 in. long with the sides each 2.9 in. in length. Both shim rods were fully withdrawn and the regulating rod was withdrawn approximately 8 in. to achieve criticality with 60 fuel elements at a power of 50 W. Core and assembly components were aluminum.

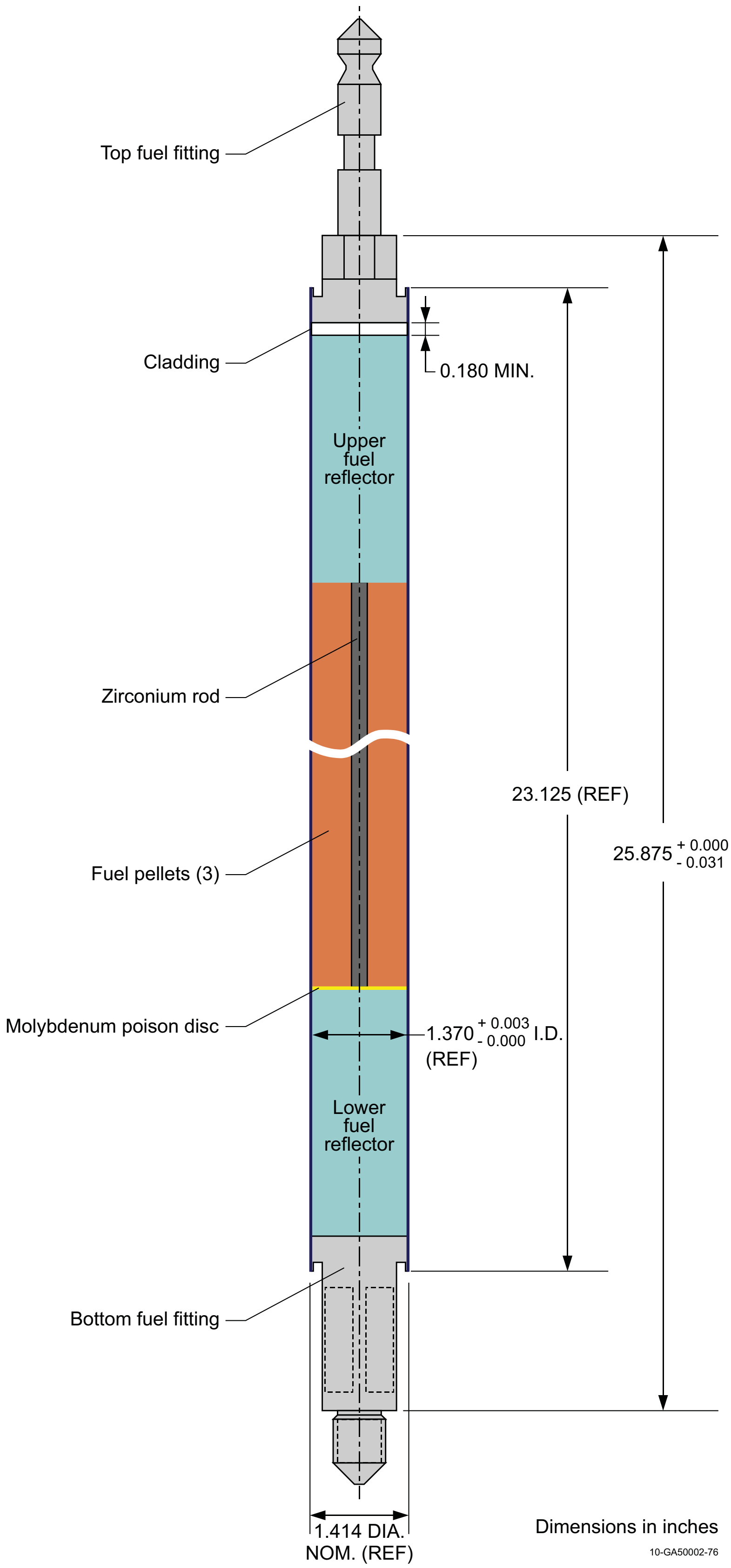

Figure 3. Typical TRIGA Fuel Element.

Table I. Summary of As-Built Fuel Data per Average Fuel Element (Appendix D).[(a)]

Core Configuration	Operational
Number of Fuel Elements	60
Total Mass (g)	2506.5 ± 3.4
Uranium Mass (g)	749.9 ± 2.7
^{235}U Mass (g)	148.0 ± 0.6
^{235}U Enrichment (%)	19.74 ± 0.02
U Mass Content (wt.%)	29.92 ± 0.09
H/Zr Ratio	1.58 ± 0.01
Er Content (wt.%)	0.90 ± 0.02
C Content (wt.%)	0.30 ± 0.02
Fuel Element Length (mm)	380.2 ± 0.4
Fuel Element Diameter (mm)	34.805 ± 0.003
Cladding Inner Diameter (mm)	34.894 ± 0.005
Fuel-Clad Difference (mm)	0.089 ± 0.005

(a) The uncertainty in these values is 1σ of the average population and not the average mean.

The beam lines each consist of a rectangular beam-filter tube adjacent to the core (typically filled with helium), an aperture mechanism containing boron nitride apertures to adjust the definition of the neutron radiograph, and a beam tube attached to the tank wall. Additional beam tubes extend from the external tank surface to the radiography stations. A 5 Ci AmBe source is located in the graphite block in the F-3 core position and the dry tube used for experiments is located in the F-1 core position. An empty control rod guide tube is placed in the C-4 cluster to facilitate in-core irradiations. The tank water temperature was 27.5 ºC. Typical water temperatures are between 26 and 39 ºC, where the upper limit is maintained during full-power operations with a coolant water heat exchanger.

2 BENCHMARK EVALUATION

Monte Carlo n-Particle (MCNP) version 5.1.51 calculations were utilized to estimate the biases and uncertainties associated with the experimental results in this evaluation. MCNP is a general-purpose, continuous-energy, generalized-geometry, time-dependent, coupled n-particle Monte Carlo transport code [5]. The Evaluated Neutron Data File library, ENDF/B-VII.0, cross section data was also used in this evaluation [6]. The statistical uncertainty in k_{eff} and Δk_{eff} is 0.00007 and 0.00010, respectively.

2.1 Uncertainty Analysis

Significant effort went into precisely characterizing all aspects of the reactor core dimensions and material properties. Detailed analyses of reactor parameters minimized experimental uncertainties. The purpose of the additional work in evaluating the experimental uncertainties was to reduce their worth when assessing the total computational bias for models of the NRAD reactor. A compilation of the total evaluated uncertainty in the NRAD reactor benchmark is shown in Table II. Systematic and random components of the uncertainties were evaluated. All uncertainty values correspond to 1σ; uncertainties less than or equal to 0.00010 are treated as negligible (neg) and those with no evaluated random component of the total uncertainty are noted with “--”. All uncertainties providing at least 0.04 %Δk_{eff} are highlighted in gray. The largest contributors to the total benchmark uncertainty were the ^{234}U, ^{236}U, Er, and Hf content in the fuel; the manganese content in the stainless steel cladding; and the unknown quantity of water saturation in the graphite reflector blocks.

2.2 Benchmark Model Development

A detailed model of the NRAD reactor 60-fuel-element core configuration was prepared to evaluate biases and uncertainties in the benchmark model. Because many of the uncertainties and effective biases in model simplifications were small or otherwise negligible, a detailed benchmark model was unnecessary. A more thorough discussion of the simplification process, benchmark model details, as well as example detailed and simple model input decks for MCNP5, are publicly available [1]. The mid-plane cross section view of the NRAD reactor core is shown in Fig. 4; a vertical cross section view providing the reference placement of core components is shown in Fig. 5.

Most impurities were removed from materials in the NRAD reactor benchmark model. Boron, carbon, erbium, and hafnium were retained in the fuel matrix. The detailed reactor infrastructure, such as grid plate, support structure, assembly components, etc., was constructed from aluminum and was removed from the model with a small impact on k_{eff}. A simplification bias of 0.0012 ± 0.0009 Δk_{eff} was determined for simplification of the NRAD benchmark model. The experimental k_{eff} of 1.0000 ± 0.0027 (1σ) was corrected to a benchmark k_{eff} of 1.0012 ± 0.0029 (1σ).

Table II. Total Experimental Uncertainty of the 60-Fuel-Element NRAD Core.

Perturbed Parameter	Parameter Value	1σ Uncertainty	±Δk_{eff} (1σ) Systematic	±Δk_{eff} (1σ) Random
Temperature (ºC)	26	±2	neg	--
Control Rod Positions (cm)	38.1, 38.1, 20.4216	±0.0508	neg	--
Total Fuel Element Mass (g)	2506.5	±0.1	neg	neg
Uranium Mass (g)	749.9	±0.1	neg	neg
^{235}U Mass (g)	148.0	±0.43	0.00014	0.00012
^{232}U Content (wt.%)	~0	--	neg	--
^{234}U Content (wt.%)	0.2	±0.03	0.00050	--
^{236}U Content (wt.%)	0.2	±0.2	0.00104	--
Fuel Element Length (cm)	38.02	±0.04	neg	neg
Fuel Element ID (cm)	0.635	+0, -0.0254/√3	neg	neg
Fuel Element OD (cm)	3.4805	±0.0003	neg	neg
Hydrogen/Zirconium Ratio	1.58	±0.01	0.00015	0.00011
Hydride Homogeneity	1.58	±0.03	neg	neg
Erbium Content in Fuel (wt.%)	0.90	±0.02	0.00040	0.00029
Carbon Content in Fuel (wt.%)	0.30	±0.02	neg	neg
Hafnium Content in Fuel (wt.%)	0.008	±0.008/√3	0.00083	--
EBC in Fuel (wt.%)	0.000104	0.000015	neg	--
Zirconium Density (g/cm^3)	6.51	±0.01	neg	--
Zirconium Rod Diameter (cm)	0.5715	+0, -0.0254/√3	neg	neg
Zirconium Impurities	Table 2.18	-100%	neg	--
Molybdenum Density (g/cm^3)	10.22	±0.02	neg	--
Molybdenum Volume	Figure 1.27	--	neg	neg
Molybdenum Impurities	Table 2.21	-100%	neg	--
Axial Graphite Density (g/cm^3)	1.73	±0.01	neg	--
Axial Graphite Volume	Figure 1.28	--	neg	neg
Axial Graphite Impurities	Table 2.24	-100%	neg	--
Stainless Steel Carbon Content	Table 2.27	Table 2.27	neg	--
Stainless Steel Chromium Content	Table 2.27	Table 2.27	neg	--
Stainless Steel Manganese Content	Table 2.27	Table 2.27	0.00066	--
Stainless Steel Nickel Content	Table 2.27	Table 2.27	0.00014	--
Stainless Steel Density (g/cm^3)	8.00	±0.01	neg	--
Steel Cladding Length (cm)	58.7375	--	neg	neg
Steel Cladding ID (cm)	3.4894	±0.0005/√3	neg	neg
Steel Cladding OD (cm)	3.59156	±0.0005/√3	neg	neg
Steel End Fitting Volume	Figures 1.29 and 1.30	±100%	neg	neg

Table II (cont.). Total Experimental Uncertainty of the 60-Fuel-Element NRAD Core.

Perturbed Parameter	Parameter Value	1σ Uncertainty	±Δk_{eff} (1σ) Systematic	±Δk_{eff} (1σ) Random
Stainless Steel Impurities	Table 2.28	±100%	0.00035	--
Air Composition and Density	Tables 2.35 and 2.36	--	neg	--
Boron Carbide Density (g/cm^3)	2.48	+0.04/√3	neg	--
B_4C Boron Content (wt.%)	78.3	±1.0/√3	neg	neg
^{10}B Abundance (wt.%)	18.43	±0.2/√3	neg	--
Absorber Length (cm)	38.1	--	neg	neg
Absorber Diameter (cm)	3.01498	+0, -0.0762/√3	neg	--
B_4C Impurities	none	Table 2.41	0.00010	neg
Control Rod Burnup	--	--	neg	--
Aluminum 6061 Composition	Table 2.43	--	neg	--
Aluminum 6061 Density (g/cm^3)	2.70	±0.01	neg	--
Aluminum Clad Length (cm)	60.96	--	neg	neg
Aluminum Clad Thickness (cm)	0.07112	±100%	neg	--
Guide Tube Volume (vol.%)	Figure 1.35	±30%	neg	neg
Guide Tube End Fitting Volume	Figure 1.35	±100%	neg	neg
Auxiliary Control Rod Parts	Figures 1.31 and 1.33	-100%	neg	--
Aluminum 6061 Impurities	Table 2.43	-100%	neg	--
Fuel Element Pitch in Assembly (cm)	3.8862	+0.01524/√3 -0.01016/√3	neg	neg
Fuel Cluster Assembly Parts	Figures 1.14 through 1.18	-100%	neg	neg
Graphite Block Density (g/cm^3)	1.570	±0.009	0.00020	--
Graphite Block Length (cm)	65.7225	±0.3175/√3	neg	neg
Graphite Block Cross Section (cm^2)	Figure 1.37	Figure 1.37	neg	0.00014
Graphite Block Impurities	Table 2.48	-100%	neg	neg
Water Saturation of Graphite (vol.%)	0	+30%/√3	0.00207	--
Aluminum 2011 Components	Figure 1.36 and Table 2.54	-100%	neg	--
Graphite Assembly Parts	Figures 1.14 and 1.38	-100%	neg	--
AmBe Source Properties	Figure 1.51 and Table 2.56	-100%	neg	--
Source Tube and Cap Properties	Figures 1.52 and 1.53 and Table 2.43	-100%	neg	--

Table II (cont.). Total Experimental Uncertainty of the 60-Fuel-Element NRAD Core.

Perturbed Parameter	Parameter Value	1σ Uncertainty	±Δk_{eff} (1σ) Systematic	±Δk_{eff} (1σ) Random
Aluminum 1100 Grid Plate	Figure 1.9 and Table 2.57	-100%	neg	--
Assembly Pitch in Grid Plate (cm)	8.10006 E-W 7.7089 N-S	±0.0533/√3	0.00012	0.00013
Location of Holes in Grid Plate (cm)	Figure 1.10	±0.0127/√3	neg	neg
Assembly Hole Diameter (cm)	6.1722	+0.01778/√3 -0.00508/√3	0.00014	--
Bottom Assembly Diameter (cm)	6.0706	±0.0254/√3	neg	neg
Grid Plate Support Structure	Figure 1.8	-100%	neg	--
Mounting Pad	Figure 1.5	-100%	neg	--
Reactor Tank	Figure 1.5	-100%	neg	--
Water Impurities	none	Table 2.59	0.00010	--
Beam Tube Aluminum Properties	Table 2.43	--	neg	--
Beam Filter Tube Dimensions (cm)	Figure 1.42	±0.0762/√4/√3	neg	neg
Other Beam Tube Aluminum Parts	Figures 1.41	--	neg	--
Beam Tube Aluminum Impurities	Table 2.43	--	neg	--
Gas Content in Beam Tubes	--	-100%	neg	--
Boron Nitride Apertures	Figures 1.43 and 1.44 and Table 2.62	-100%	neg	--
Placement of Beam Lines (cm)	2.032	±0.12/√3	neg	0.00014
Instrumentation and Detectors	Figures 1.55 and 1.56	--	neg	neg
Uncertainty by Type	**--**	**--**	**0.00268**	**0.00041**
Total Experimental Uncertainty	**--**	**--**	**0.00271**	**--**

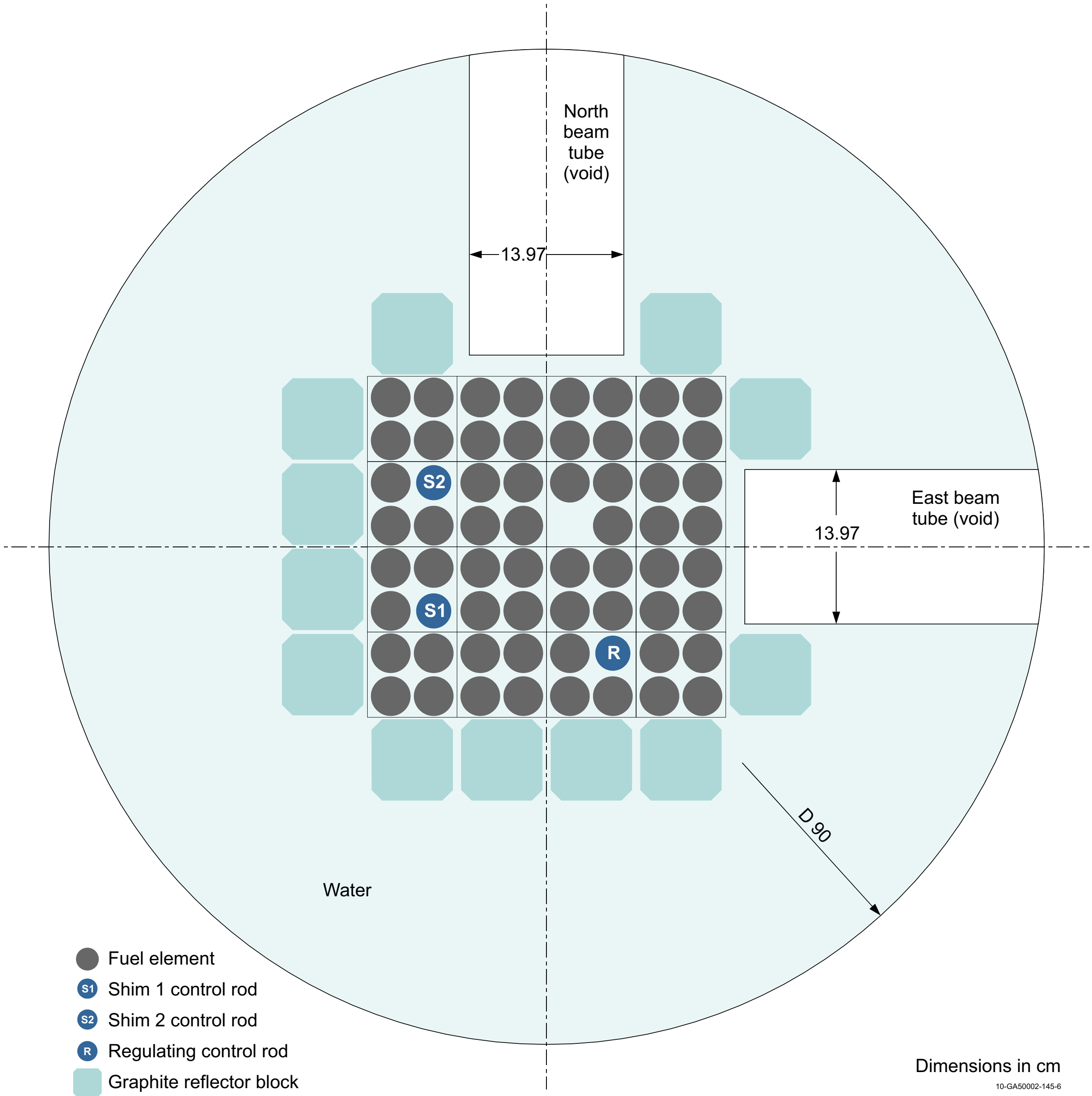

Figure 4. Mid-plane Cross Section View of NRAD Reactor Core.

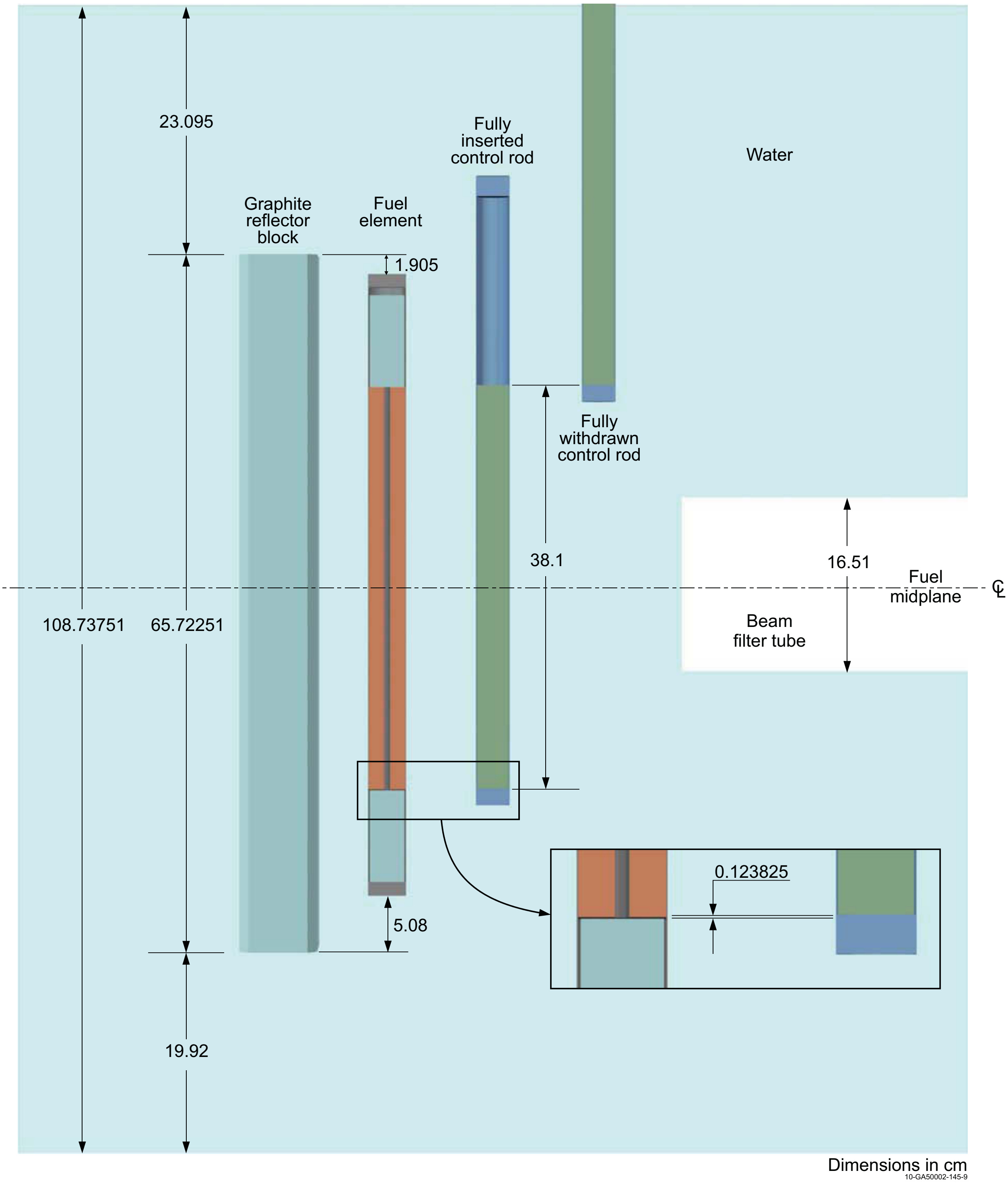

Figure 5. Vertical Cross Section View Showing the Vertical Placement of NRAD Reactor Core Components.

3 CONCLUSIONS

3.1 Results

The benchmark model of the NRAD reactor was evaluated using MCNP5 and ENDF/B-VII.0. Eigenvalues were also computed using JEFF-3.1 [7] and JENDL-3.3 [8] cross section libraries and KENO-VI [9] with ENDF/B-VII.0 for comparison with the MCNP5 results. All calculated results are between 0.3 and 0.8% greater than the benchmark eigenvalue but within the 3σ uncertainty (Table III). The bias using MCNP5 and ENDF/B-VII.0 is worth approximately \$1.23 (using a β_{eff} of 0.00745), whereas the bias for model simplifications is only \$0.16.

The reason for the difference between the calculated k_{eff} results using MCNP5 and KENO-VI ENDF/B-VII.0 (both continuous energy) is currently unknown but is being investigated.

The calculation bias using MCNP5 with ENDF/B-VII.0 and JEFF-3.1 cross section data is comparable[1] to more recent calculations performed using the benchmark of the Slovenian TRIGA reactor [4]. The two reactor core designs are fundamentally different and this may have some impact on the calculated results. Monte Carlo analysis of the TRIGA Mark II reactor at the Musashi Institute of Technology in Japan using MCNP4A and ENDF/B-V data indicated a computational bias in eigenvalue calculations of about 1.0% Δk/k; worth calculations, however, were in good agreement with experimental results [10]. Worth calculations (not currently available) for measurements in the NRAD reactor are also in good agreement with their experimental values. MCNP modeling of other LEU converted TRIGA reactors also exhibit biases of approximately 1%.[2]

Models of UErZrH- and UZrH-fueled systems over predict the absolute worth of a TRIGA reactor, which impacts prediction of core excess reactivity and rod positions at criticality. It would be expected that over prediction of k_{eff} in criticality safety applications would be similarly impacted. The effect of uncertainties and model simplifications were smaller than the computational bias, indicating a need to reevaluate the cross sections used when modeling UErZrH fuel.

Table III. Comparison of Benchmark Eigenvalues.

Analysis Code	Neutron Cross Section Library	Calculated k_{eff}	±	σ	Benchmark k_{eff}	±	σ	$\frac{C-E}{E}(\%)$
MCNP5	ENDF/B-VII.0	1.00925	±	0.00007	1.0012	±	0.0029	0.80
	JEFF-3.1[(a)]	1.00719	±	0.00007				0.60
	JENDL-3.3[(b)]	1.00633	±	0.00007				0.51
	ENDF/B-VI.8[(c)]	1.00458	±	0.00007				0.34
KENO-VI	ENDF/B-VII.0 (238-group)	1.008741	±	0.000066				0.75
	ENDF/B-VII.0[(d)] (continuous energy)	1.004496	±	0.000076				0.33

(a) JEFF-3.1 results provided by Luka Snoj at the Jozef Stefan Institute.
(b) S(α,β) data from the ENDF/B-VII.0 library was used with the JENDL-3.3 cross section data because S(α,β) data for JENDL-3.3 was unavailable.
(c) Using ENDF/B-VII.0 cross section data for erbium isotopes.
(d) Continuous energy results for KENO-VI provided by Steve Bowman at Oak Ridge National Laboratory.

[1] Personal communications with Luka Snoj at the Jozef Stefan Institute (October 20, 2010).

[2] Personal communications with Chris Ellis at General Atomics and Eric C. Woolstenhulme, Ken Schreck, Randy Damiana, and Ann Marie Philips at INL (August 4, 2010).

3.2 Current Activities

Current activities include revision of the NRAD benchmark model for the 60-fuel-element core to include worth measurements for control rods, excess reactivity, shutdown margin, graphite reflector blocks, and the experiment dry tube. There is interest in benchmarking the initial critical configuration with 56 fuel elements to provide an additional data point for UErZrH-fueled systems. Subsequent discussions with General Atomics have identified more appropriate values for the ^{234}U, ^{236}U, and Hf content in the LEU fuel. Correction of these values in the benchmark model would have an expected reduction in the computational bias of less than 0.06% $\Delta k/k$ and reduce their respective uncertainties. There are plans to weigh some of the graphite reflector blocks to assess their water content, effectively reducing the most significant uncertainty in the benchmark experiment. The model currently simulates the graphite blocks with no water content. An increase in water content is expected to further increase the computational bias up to a maximum of an additional ~$0.50.

Efforts are underway to address potential discrepancies in the Er, Zr, and ZrH (thermal scattering) cross sections. Researchers in Slovenia have identified biases in the Zr and ZrH cross sections [11]. An inconsistency in how Er cross sections are treated in the MCNP5 and KENO-VI (continuous energy) analyses has also been identified as a potential source for the computational bias between the two codes; additional work is needed to further investigate this discrepancy.

Four additional fuel elements and four graphite rods are to be added to the NRAD reactor to increase the total core excess reactivity. Development of a benchmark evaluation of the 62- (intermediate step) and 64-fuel-element core configurations with additional worth and flux measurements is a possibility.

4 ACKNOWLEDGMENTS

The ICSBEP and IRPhEP are collaborative efforts that involve numerous scientists, engineers, and administrative support personnel from 20 different countries. The authors would like to acknowledge the efforts of all of these dedicated individuals without whom these projects would not be possible.

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