Physics Experiments in the ZED-2 Reactor using CANFLEX-RU

J.E. Atfield and M.B. Zeller
AECL Chalk River Laboratories, Chalk River, Ontario, Canada

Abstract

This paper describes experiments performed in ZED-2 that provide data appropriate for validating code predictions used for the purpose of safety analyses and qualification of the use of CANFLEX-RU fuel in CANDU®. The experiments involved the substitution of CANFLEX-RU test assemblies into a natural-uranium reference lattice.

The experimental test components are:

1) Substitution of D_2O-cooled CANFLEX-RU fuel at room temperature,
2) Substitution of air-cooled CANFLEX-RU fuel at room temperature, and
3) Effect of heating channels containing CANFLEX-RU bundles from 25 to 300°C.

Experimental results are presented and discussed.

1. Introduction

Fuel enrichment reduces spent fuel volume and increases the core-average discharge burn-up. Enrichment adds to the cost of a fuel bundle, but, because even very small enrichments provide large benefits, there is an opportunity to reduce fuel-cycle costs by utilizing the large quantities of uranium recovered from the reprocessing of Light Water Reactor (LWR) fuel. This Recovered Uranium (RU) is slightly enriched (~0.9 wt% U^{235} in total U) and a fuel cycle based on it with the CANFLEX® fuel carrier is referred to as CANFLEX-RU.

Operationally, a fuel cycle based on RU represents a major departure from Natural Uranium (NU). For example, if the RU feed stock is not blended down the burn-up is at least doubled compared to NU, resulting in a much larger change in properties between fresh and discharged fuel. In addition, bundle radial power variations and end-flux peaking are more pronounced.

Atomic Energy of Canada Ltd. (AECL), British Nuclear Fuels Ltd. (BNFL) and the Korea Atomic Energy Research Institute (KAERI) agreed to undertake a joint program to study the use of RU as a fuel for CANDU reactors.

As part of the program, RU fuel pellets were provided by BNFL. The uranium product used for the pellet manufacture was obtained from the UO_3 stockpile at Sellafield's Thermal Oxide Reprocessing Plant (THORP). The pellets were fabricated at the Research and Technology (R&T) building located at the Springfields site in Lancashire, UK.

The RU pellets were used to fabricate thirty-six 43-element bundles (35 welded bundles plus one demountable) for use in the reactor-physics experimental program. The 43-element CANFLEX bundle design was chosen because it has the capability of improved

---

CANDU® is a registered trademark of AECL.
CANFLEX® is a registered trademark of AECL and the Korea Atomic Energy Research Institute (KAERI).

performance with slightly enriched fuels when compared to the standard 37-element CANDU bundle.

The bundle assembly was performed by members of AECL's Fuel Development Branch in the Advanced CANDU Fuel Development Laboratory (ACFDL) at Chalk River Laboratories (CRL), located in north-eastern Ontario.

The reactor-physics experiments were performed in the ZED-2 reactor [1], also located at CRL. The experimental test components include Coolant Void Reactivity (CVR) measurements, Fuel Temperature Coefficient (FTC) measurements, fine-structure reaction-rate measurements and end-flux peaking measurements. This paper presents results from the CVR and FTC measurements and includes analyses and a discussion of those results.

2. ZED-2 reactor

A significant component of the validation of the CANDU reactor-physics lattice codes is based on comparison of code predictions to measurements performed in ZED-2.

The ZED-2 calandria vessel, as shown in Figure 1, is a cylindrical tank with a sidewall thickness of 0.64 cm and a bottom thickness of 2.2 cm. The calandria has a 3.36 m diameter and 3.30 m depth. It is surrounded by graphite blocks arranged with an average thickness of 60 cm radially and 90 cm below the tank. Fuel assemblies are hung vertically from beams located above the calandria.

The reactor is made critical by pumping heavy water moderator into the calandria and power is controlled by adjusting the moderator level. Typical moderator critical levels range from 150 to 250 cm above the calandria floor. The maximum power is about 200 watts (nominal), corresponding to an average neutron flux of about 10^9 neutrons per centimetre squared per second.

3. Measuring lattice critical size

3.1 Flux map method

Critical buckling measurements are representative of the uniform lattice reactivity of the fuel comprising the lattice and are used to support burn-up predictions and for providing CVR data for fresh fuel. The latter are acquired by performing critical buckling measurements on lattices with the fuel channel both "cooled" with heavy water and voided.

The simplest means of measuring buckling is by the flux-map method. In this method a critical core is assembled and the thermal flux shape is mapped both radially and axially (along the direction of the channels) by irradiating appropriately placed copper foils at cell boundaries. Buckling is determined by fitting the measured distributions to a Bessel function radially and a cosine axially. This method works well for most lattices, providing that the lattice size is not too small (*i.e.*, the buckling is not too large).

3.2 Substitution method

With slightly enriched fuel the lattice size is much smaller so that the asymptotic region is reduced; this in turn decreases the accuracy of measuring buckling using the flux-map method.

The amount of RU fuel used for these measurements was 35 bundles. Non-uniform critical lattices, containing only a few CANFLEX-RU assemblies, were used to study the reactivity effects caused by channel voiding and heating by employing the substitution method. The experimental method has been described in a previous paper [2] and only a brief description is given, as follows.

The measurements were performed by systematically substituting the CANFLEX-RU assemblies into a NU reference lattice, and observing the resulting change in lattice critical size (*i.e.*, moderator critical height) as the substitutions proceeded. CVR was studied by performing the substitutions using both heavy water and air (void) coolant in the test assemblies. High-temperature reactivity effects were studied by placing the CANFLEX-RU bundles in ZED-2 hot channels [3].

4. Substitution analysis

The MCNP code [4] used at Chalk River has been modified for the specific purpose of performing substitution analyses. The modified version allows for the adjustment of the weight of the fission neutrons born in the fuel in each lattice cell. A Neutron Production Correction Factor (NPCF) (*i.e.*, a scaling factor) is used to adjust the weight of the source neutrons in a given calculation. This allows the analyst to adjust the calculated k-effective value to force it to be unity, *i.e.*, to force the MCNP model of a system known to be critical to agree with experiment.

The procedure is as follows. The reference lattice measurement is modelled initially. Typically, the reference lattice is a uniform lattice that contains only reference fuel. The reference-fuel NPCF is adjusted to achieve neutron balance (*i.e.*, k-effective equal to unity) and this NPCF value is fixed for the subsequent substitution-lattice calculations.

The substitution lattices are modelled using the measured moderator critical heights to define the lattice critical size for each calculation. The reference-fuel neutron yield is fixed using the reference-lattice NPCF, and the NPCF in all cells containing test fuel is adjusted so that neutron balance is achieved for each substitution-lattice calculation.

A separate substitution analysis is performed for each coolant type, one for void coolant and one for D_2O coolant. The difference in the NPCF values then defines the bias in the calculation of CVR in the MCNP model. When the model is then used to calculate CVR for the test-fuel lattice (*i.e.*, for a uniform bare lattice of CANFLEX-RU) the calculated value will be corrected for bias if the NPCF values are used for the calculation [5]. The bias-corrected value can then be compared directly to the prediction of a lattice code, such as WIMS-IST [6], to validate the code and associated nuclear data library for CVR calculations.

A significant improvement in substitution analysis results from using MCNP over previous diffusion-theory analyses [7] in that only one test-fuel NPCF value is required for all of the various substitution-lattice configurations. This is due to the ability of MCNP to accurately calculate the neutronic interaction between adjacent cells in the lattice containing dissimilar fuel types. It is not necessary to use approximations, such as contamination parameter, to extrapolate to the NPCF value corresponding to a uniform lattice of the test fuel.

5. Room-temperature experiments

Figure 2 is a top view of the lattice used for the study. It comprised 55 assemblies arranged hexagonally with a 31-cm centre-to-centre spacing. The seven centre sites comprise the substitution region. Figure 3 shows the various substitution configurations used for the measurements.

Figure 4 and Figure 5 depict the test assemblies used for the room-temperature CVR measurement. The seven test assemblies each contained five CANFLEX-RU bundles (35 bundles total) with Zr-Nb Pressure Tubes (PT) and Zr-2 Calandria Tubes (CT). The radial dimensions of the assemblies (referred to as CANDU-type channel assemblies in this paper) are identical to those of the CANDU-6 PT and CT. The assemblies have bottom openings that allow heavy water to enter the pressure tubes as moderator is pumped into the ZED-2 calandria. Plugs were inserted into the bottom openings of the assemblies for the air-coolant experiments to isolate the test fuel from the moderator.

As mentioned previously, the test-fuel bundle geometry is CANFLEX. However, there are no appendages attached to the fuel sheaths (*i.e.*, no spacers and bearing pads). This is to allow the bundles to fit inside channels with a smaller PT inner diameter and to facilitate the insertion of support rods, thermocouples and heater leads into the sub channels for the high-temperature experiments. The absence of bearing pads necessitated the use of zirconium-wire clips (two clips per bundle) to centre the bundles in the CANDU-type channel assemblies (see also Figure 4 and Figure 5).

5.1 Results

Figure 6 shows the moderator critical heights obtained for the various substitution configurations (see Figure 3). The data show that the moderator level decreases for both coolant types as test fuel is substituted into the lattice. This is consistent with the CANFLEX-RU assemblies being more reactive than the NU reference-fuel assemblies. The critical heights decrease more rapidly with air coolant, compared to D_2O coolant, indicating a positive CVR.

Analyses were performed on the reference lattice measurement and the two seven-rod substitution measurements to derive NPCF values for the reference fuel and the test fuel (D_2O and air coolant). These NPCF values were applied to the MCNP ZED-2 model and k-effective values were derived for all of the substitution configurations using the measured critical heights as inputs. The results are summarized in Figure 7. Note that the k-effective values are within 0.2 milli-k of unity for all configurations.

The NPCF value for the D_2O-cooled CANFLEX-RU assembly was then applied to a bare-lattice model (*i.e.*, no reflector) comprising only CANFLEX-RU assemblies; the lattice was hexagonal at a 31-cm spacing with cylindrical boundaries. The size of the lattice was adjusted both radially and axially until the k-effective output value approached unity. It was found that an axial length of 157 cm and radius of 129.5 cm resulted in a k-effective value of 1.00017 and this length and radius were used for the remaining analysis.

The lattice size was fixed, D_2O coolant was replaced with air coolant, and the air-coolant NPCF value was applied to the model. K-effective was recalculated and this value was used to derive CVR for the lattice. The results are summarized in Table 1.

6. High-temperature experiments

Figure 8 and Figure 9 depict the hot-channel assemblies used for the high-temperature study. The seven assemblies each housed a five-bundle string of CANFLEX-RU fuel contained inside a concentric Zr-Nb PT and Zr-2 CT. Both tubes are capped at the bottom with a hemispheric dome and welded at the top to a machined hub. A top closure provides a high-temperature, high-pressure seal.

A lip on the PT dome supports a Zr-4 fuel standoff and a Zr-Nb fuel support plate. For high-temperature work the space (annulus) between the PT and CT is evacuated to reduce convection heat loss into the moderator. To reduce radiative heat loss, the annulus contains two polished aluminium heat shields. The heat shields are held in place by ceramic spacer collars positioned axially on even intervals inside the annulus.

The temperature of the channel contents was raised and maintained by the use of small high-power electrical heaters located under the test-fuel strings. The heater leads (not shown) run up the outermost sub channel and through a pressure seal in the top closure. The fuel string is instrumented with three thermocouples (not shown) located at the mid-planes of the top, middle, and lowest bundles.

Natural convection distributes the heat throughout each hot channel. Pressurized CO_2 gas is the convective medium for the void-coolant measurements. A pressurized helium cover gas was used to suppress boiling with D_2O coolant in the channels. Pressure is monitored using transducers positioned inside the high-pressure boundary.

6.1 Results

Figure 10 shows the moderator critical heights obtained from heating the channels using the seven-rod substitution configuration. The two critical-height curves imply that the reactivity worth of the CANFLEX-RU fuel decreases with temperature for both coolant conditions. However, the water-coolant curve increases more rapidly than the gas-coolant curve, demonstrating that CVR increases with channel temperature.

With D_2O coolant the critical-height curve increases linearly with temperature up to about 150^{o}C. Above 150^{o}C, the water density decrease is sufficiently rapid that the positive CVR causes the critical-height curve to begin turning over.

With CO_2 gas coolant there is minimal coolant-density effect with heating. These data are a direct measure of the fuel-temperature reactivity effect. The critical-height curve is approximately linear between room temperature and 300^{o}C.

NPCF values were derived for the reference lattice and the two room-temperature substitution lattices (D_2O and air coolant). These NPCF values were applied to the MCNP model and k-effective values were calculated at the various channel temperatures, using the measured critical heights as inputs. The results are summarized in Figure 11.

The D_2O-water coolant results show a definite trend, with the k-effective values increasing smoothly by about a milli-k between room temperature and 300^{o}C. This shows a discrepancy between the model and experiment that is not understood at this time. Further investigation is required.

However, the CO_2-gas coolant results do not show any trend; the k-effective values appear to vary randomly between room temperature and 300^{o}C. This implies that the model successfully calculates the reactivity effect of heating the fuel.

The NPCF value for the air-cooled CANFLEX-RU CANDU-type channel assembly was applied to the MCNP bare-lattice model comprising voided CANFLEX-RU at room temperature. It was found that an axial length of 155 cm and radius of 128 cm resulted in a k-effective value of 0.99949.

The lattice size was then fixed, and the fuel temperature was increased to 300^{o}C. The k-effective value was recalculated and this value was used to derive the FTC for the lattice. The results are summarized in Table 2.

7. Comparison to WIMS-IST calculations

As discussed in Section 4, the above analyses used measured NPCF values and the derived CVR value and FTC value are bias corrected (i.e., they are truly representative of a CANFLEX-RU lattice). These values can be compared directly to lattice-code predictions for the purpose of code validation.

Calculations were performed using the lattice code WIMS-IST for comparison with the experimental results. Both CVR at room temperature and FTC between room temperature and 300^{o}C were calculated. The results are summarized in Table 3 (CVR comparison) and Table 4 (FTC comparison).

The calculations over predict CVR with a bias of $+0.8 \pm 0.4$ milli-k. The FTC comparison shows bias of -0.001 ± 0.001 milli-k Kelvin^{-1} between room-temperature and 300^{o}C.

These comparisons show that there is good general agreement on trends between calculation and experiment. However, the uncertainties listed in this paper are currently based on “engineering judgement” and a more rigorous assessment of uncertainties will be the subject of a future study.

8. References

[1] “ZED-2”, AECL Report, AECL-2132, 1964.

[2] M.B. Zeller et al., “Void-Reactivity Measurements in the ZED-2 Reactor using the Substitution Method”, Fifteenth Annual Conference of the Canadian Nuclear Society, Montreal, PQ, Canada. 1994 June.

[3] M.B. Zeller et al., “High Temperature Physics experiments using UO_2 and Simulated Irradiated CANDU-Type Fuel in the ZED-2 Reactor”, Twenty-First Annual Conference of the Canadian Nuclear Society, Toronto, Ontario, Canada. 2000 June.

[4] “MCNP – A General Monte Carlo N-Particle Transport Code”, Version 5 Los Alamos National Laboratory Report, LA-UR-03-1987, 2003 April.

[5] Y. Dweiri et al., "Comparison of Bare Lattice Calculations using MCNP Against Measurements with CANFLEX LEU in ZED-2", Twenty-Ninth Annual Conference of the Canadian Nuclear Society, Toronto, Ontario, Canada. 2008 June.

[6] J.D. Irish and S.R. Douglas, "Validation of WIMS-IST", Twenty-Third Annual Conference of the Canadian Nuclear Society, Toronto, Ontario, Canada. 2002 June.

[7] R.S. Davis et al., "Validation of the Substitution Method for Measurement of Void Reactivity", Twenty-First Annual Conference of the Canadian Nuclear Society, Toronto, Ontario, Canada. 2000 June.

9. Acknowledgements

The authors acknowledge the contribution of our technologist Greg Morin in planning and performing the experiments.

Thanks to Yousif Dweiri and Benoit Arsenault, former members of our section, for their contributions in developing and validating the MCNP-based substitution method.

Special thanks to Bruce Wilkin, senior ZED-2 physicist, who modified the MCNP source code for the purpose of performing substitution analyses.

Table 1: Bare-lattice derivation of room-temperature CVR for CANFLEX-RU in CANDU-Type Channel Assemblies

Coolant	Height (cm)	Radius (cm)	NPCF	k-effective	CVR Worth (milli-k)
D_2O	157.0	129.5	1.00797	1.00013 ± 0.00030	+4.8 ± 0.4
Air	157.0	129.5	1.00719	1.00496 ± 0.00030	

Table 2: Bare-lattice derivation of fuel-temperature coefficient for CANFLEX-RU in CANDU-Type Channel Assemblies

Fuel Temperature	Height (cm)	Radius (cm)	NPCF	k-effective	Worth (milli-k)	FTC (milli-k/^oC)
25^oC	155.0	128.0	1.00719	0.99949 ± 0.0003	-6.1 ± 0.4	-0.022 ± 0.001
300^oC	155.0	128.0	1.00719	0.99345 ± 0.0003		

Table 3: WIMS-IST calculated CVR and Bias for CANFLEX-RU in CANDU-Type Channel Assemblies

Coolant	Buckling (m^{-2})	k-effective	CVR Worth (milli-k)	CVR Bias (milli-k)
D_2O	7.011	1.00000	5.6	+0.8 ± 0.4
Air	7.011	1.00568		

Table 4: WIMS-IST calculated FTC and Bias for CANFLEX-RU in CANDU-Type Channel Assemblies

Fuel Temperature	Coolant	Buckling (m^{-2})	k-effective	Worth (milli-k)	FTC (milli-k/^oC)	FTC Bias (milli-k/^oC)
25^oC	Void	7.190	1.00000	-6.3	-0.023	-0.001 ± 0.001
300^oC	Void	7.190	0.99377			

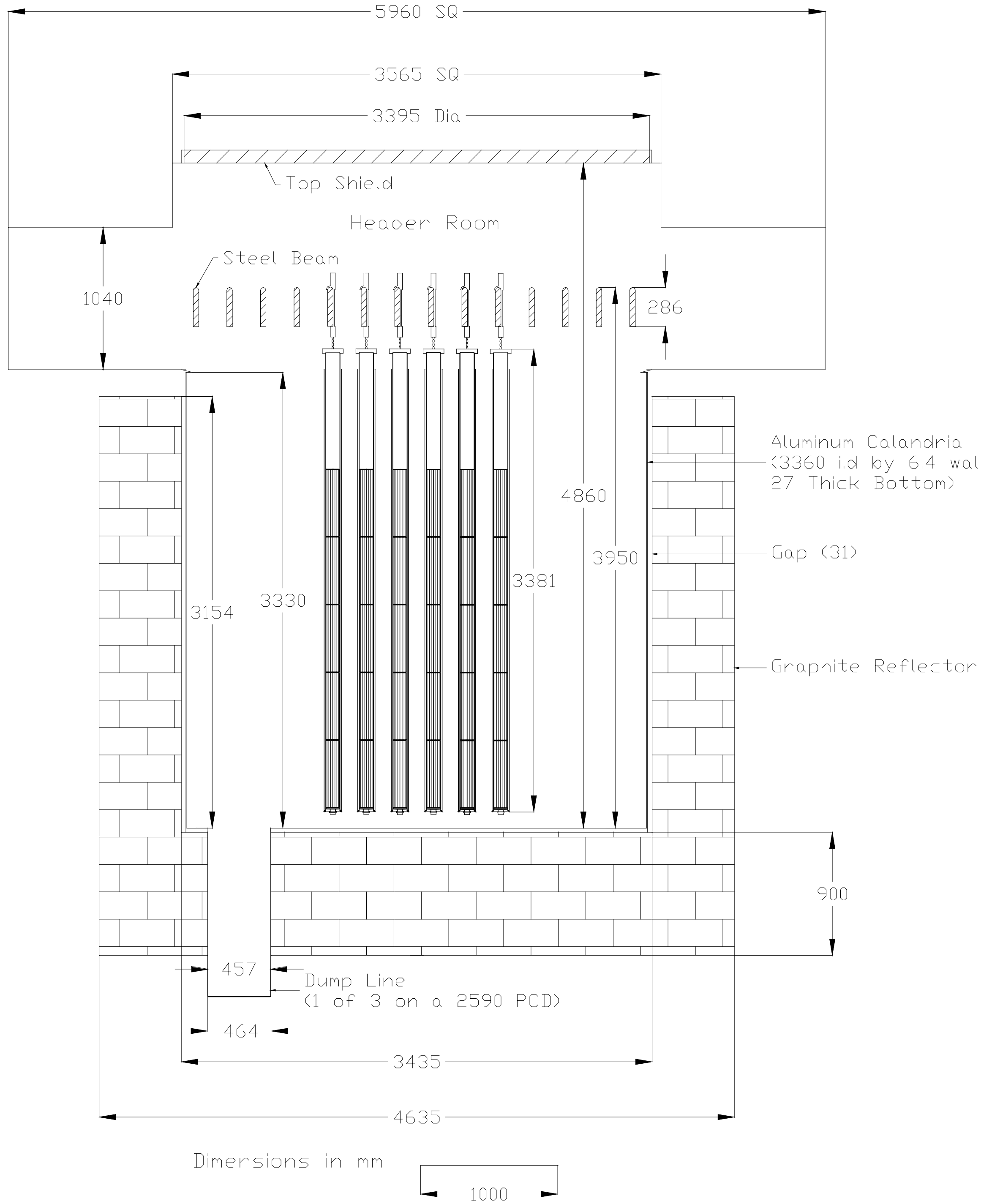
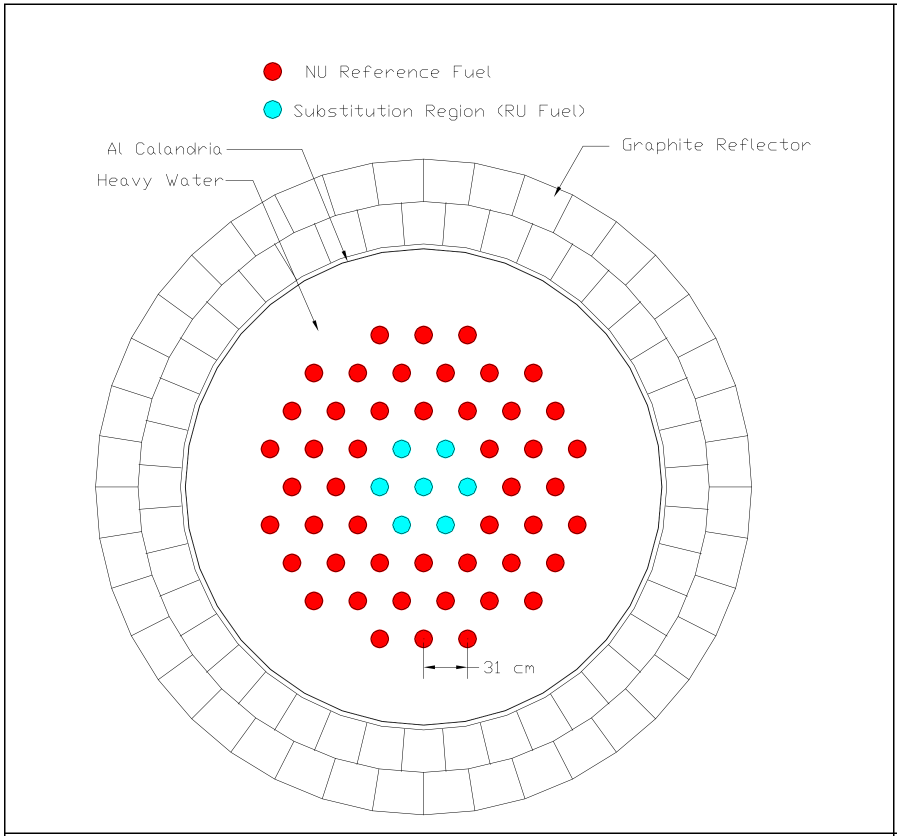


Figure 1 ZED-2 Reactor—Vertical Cross Section



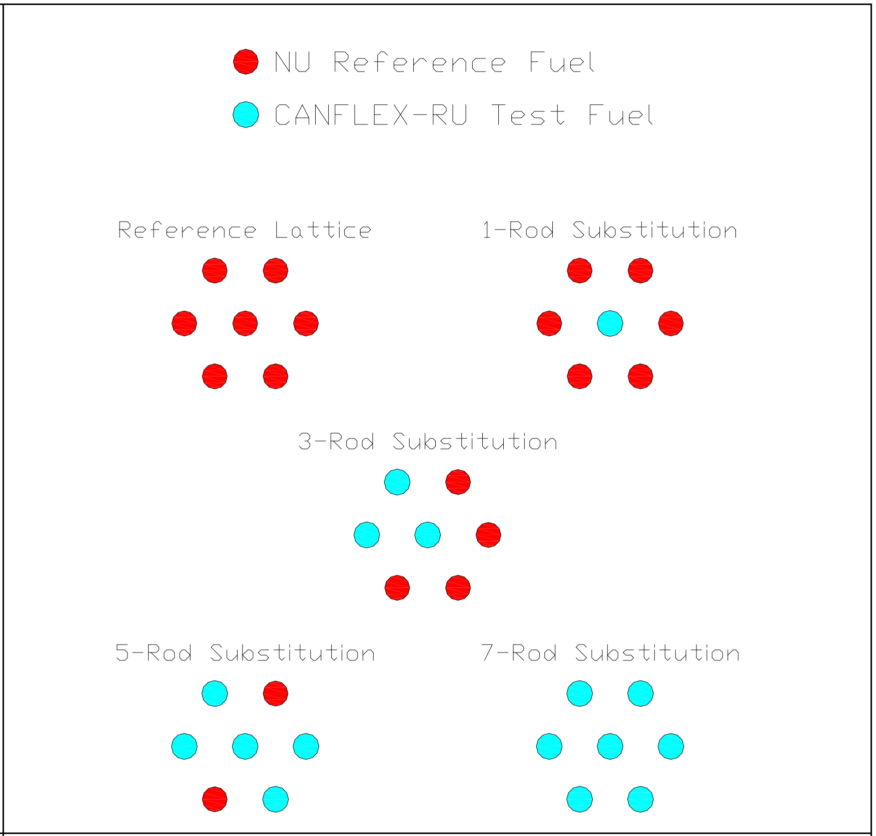


Figure 2 Top View Substitution Lattice	Figure 3 Sequence of Substitutions

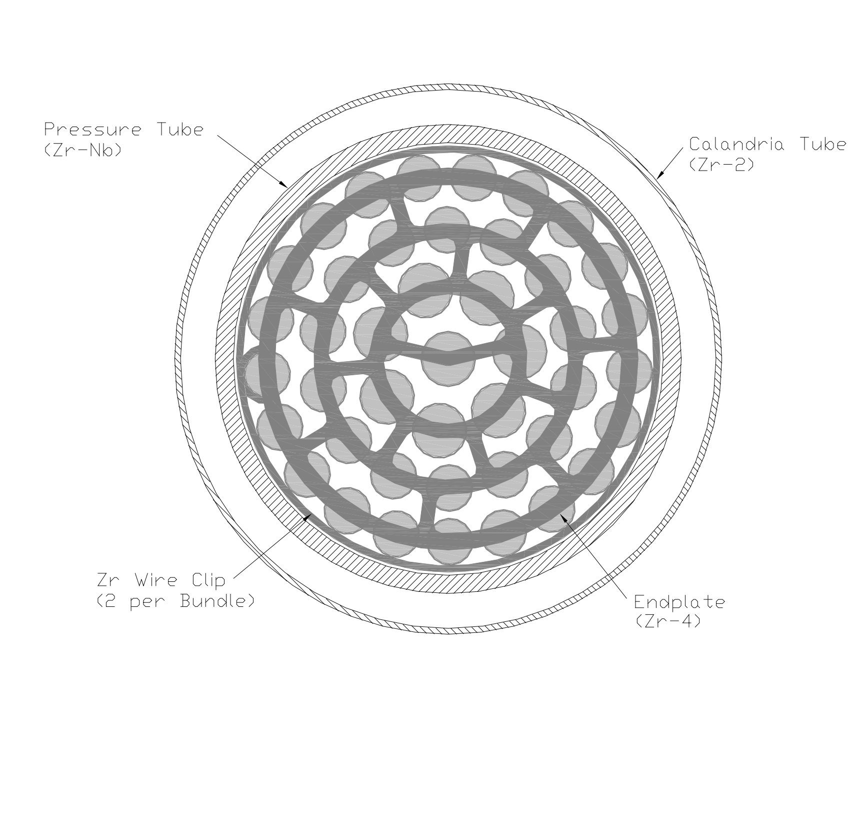


Figure 4 Top view of CANFLEX-RU in a CANDU-type Channel

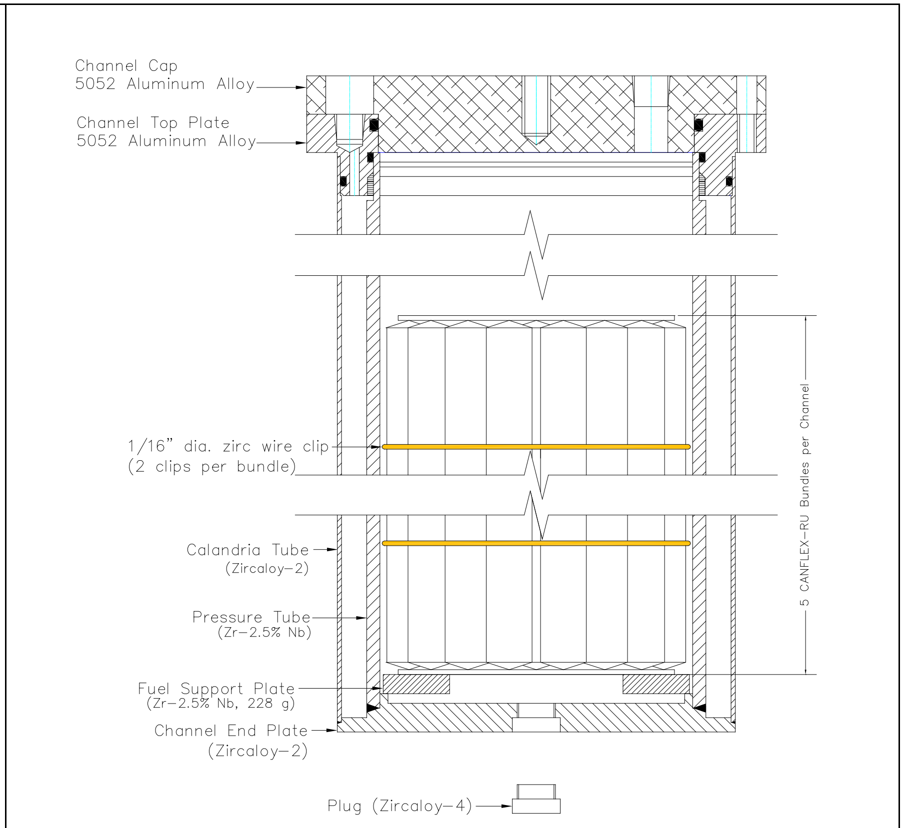


Figure 5 Side view of CANFLEX-RU in a CANDU-type Channel

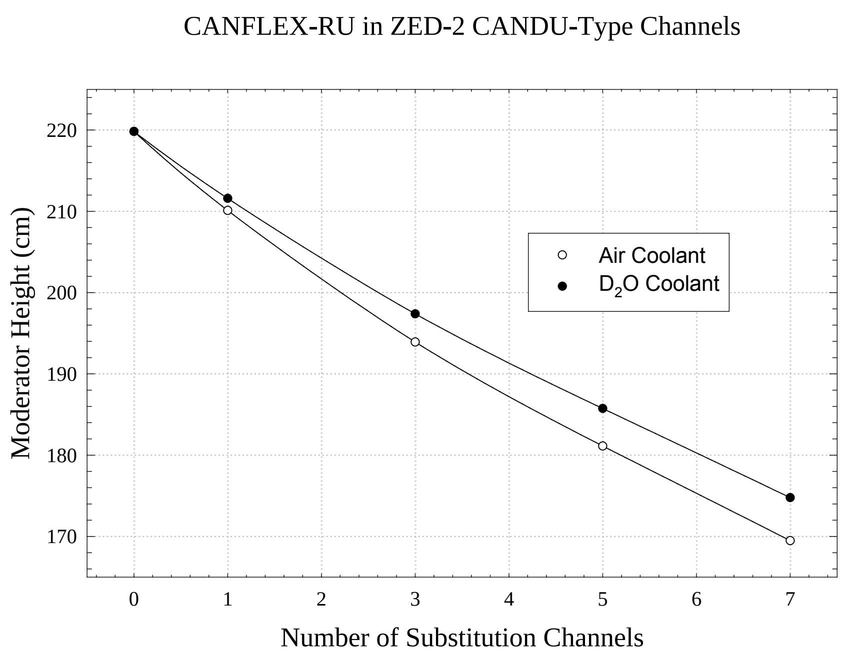


Figure 6 Room-Temperature Critical Height Data

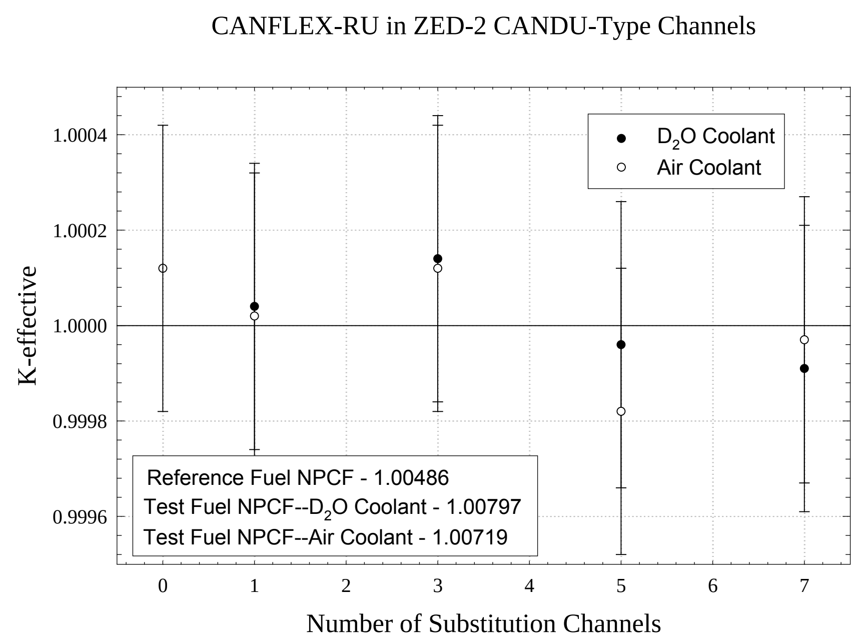


Figure 7 Room-Temperature Substitution Analysis

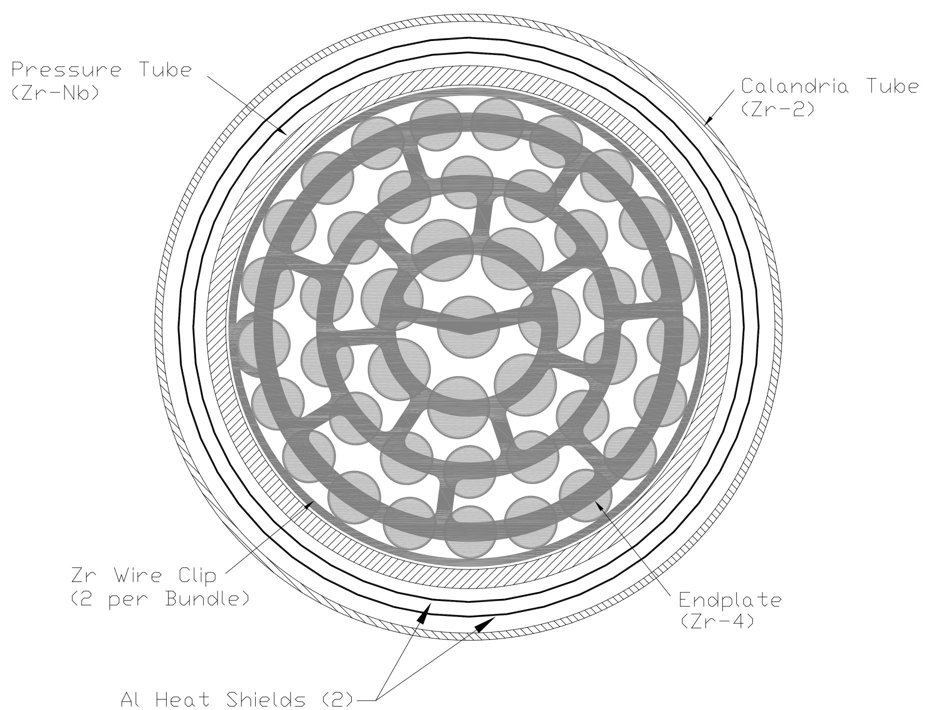


Figure 8 Top view of CANFLEX-RU in a Hot Channel

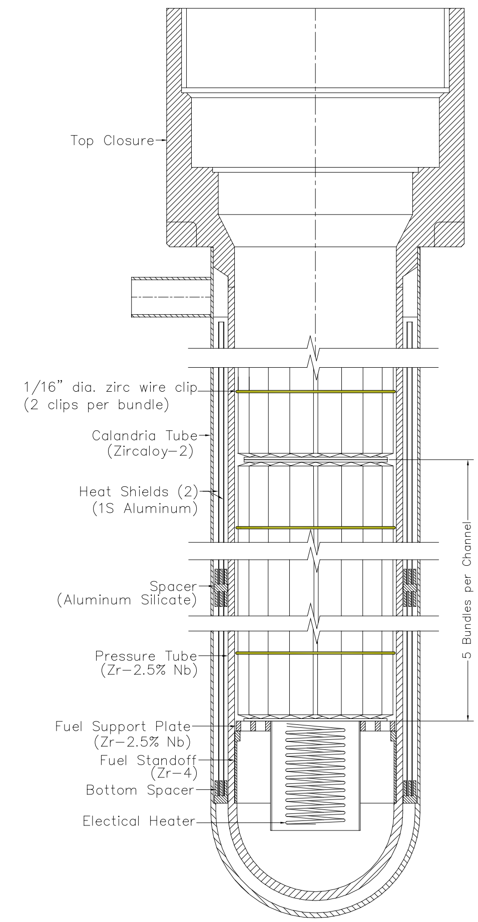


Figure 9 Side view of CANFLEX-RU in a Hot Channel

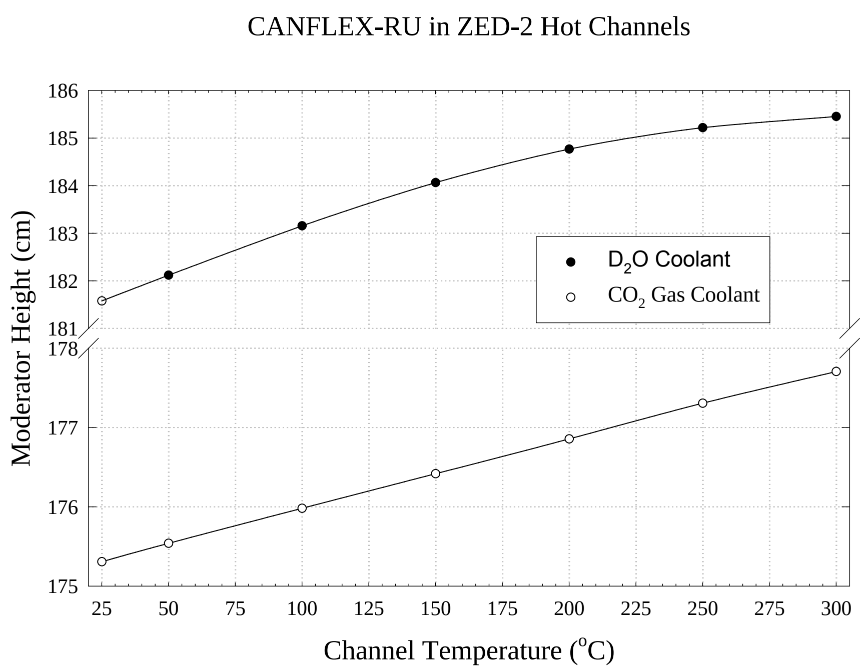


Figure 10 High-Temperature Critical Height Data

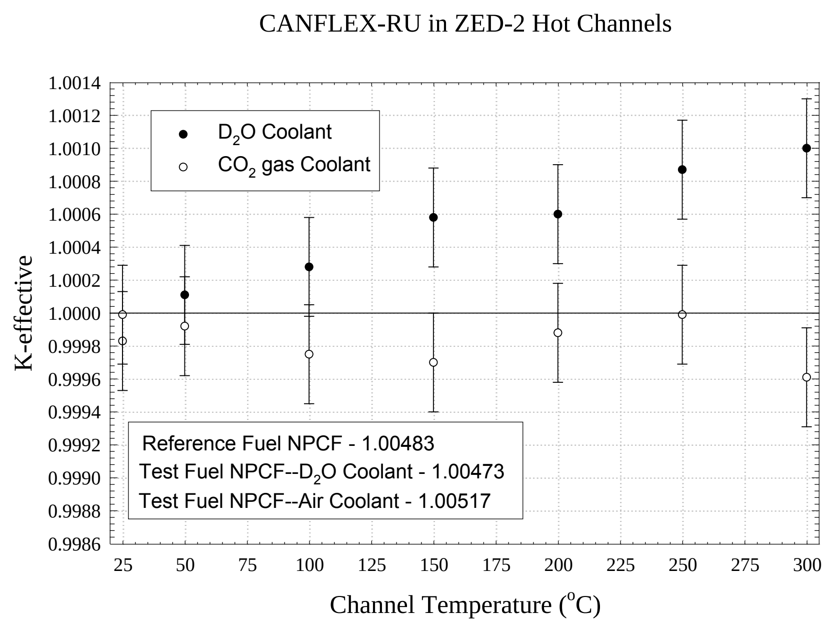


Figure 11 High-Temperature Substitution Analysis