NEUTRONIC CALCULATIONS FOR THE CONVERSION OF THE UNIVERSITY OF FLORIDA TRAINING REACTOR FROM HEU TO LEU FUEL

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1.0 Introduction

The University of Florida Training Reactor (UFTR) is located on the University of Florida campus in Gainesville, Florida. The reactor is the Argonaut type, heterogeneous in design and currently fueled with 93% enriched, uranium-aluminum alloy MTR plate-type fuel. Investigations are being performed to examine the feasibility of replacing the highly-enriched fuel of the current UFTR with 4.8% enriched, cylindrical pin SPERT fuel. The SPERT fuel is stainless steel clad and contains uranium dioxide (UO_2) pellets.

On a broad spectrum, training reactor conversion from high enrichment uranium (HEU) to low enrichment uranium (LEU) fueled facilities has been a continuing concern in the International Atomic Energy Agency (IAEA)[1,2] and significant work has been done in this area by the Argonne RERTR Program. The International Atomic Energy Agency cites three reasons for reactor conversion to low-enriched uranium. The main reason is the desire to reduce the proliferation potential of research reactor fuels. The second is to increase the assurance of continued fuel availability in the face of probable restrictions on the supply of highly-enriched uranium. The third reason is the possible reduction in requirements for physical security measures during fabrication, transportation, storage and use[1]. This same IAEA report points out that the three reasons stated for the conversion of the fuel of research reactors are interrelated and cannot be considered individually.

The concerns of the Nuclear Engineering Sciences Department at the University of Florida relating to the HEU fuel of the UFTR coincide with those of the International Atomic Energy Agency. The primary reason for going to low-enriched pin-type fuel is the concern with proliferation provoked by the highly-enriched plate fuel which has led to tighter security of nuclear facilities such as the UFTR. A second reason for changing to the pin-type fuel is because of difficulties that are being encountered in the supply of the plate-type fuel. A final reason for converting to the SPERT fuel is related to a desire to upgrade the UFTR from a 100 to a 500 kw facility. Such a change would be difficult with the current aluminum-uranium alloy plate fuel since

warping of the plates has been experienced at the 100 kw power level. A motivating force in the proposed UFTR conversion is the existence of the SPERT fuel in-house at the University of Florida (under a DOE agreement) and the DOE's encouragement to pursue the described conversion.

The purpose of this paper is to present the results of detailed neutronic investigations performed as part of the feasibility study for the UFTR conversion from HEU to LEU fuel. These neutronics studies included two phases. In the first phase, the efforts were directed primarily towards finding a satisfactory metal-to-water ratio for the SPERT fuel. An acceptable metal-to-water ratio has to satisfy the following neutronics criteria:

1. The neutron multiplication factor, or k_{eff}, for the unrodded UFTR must lie in the range of 1.03 to 1.09 [3] both for the current power limit of 100 kw and also for the proposed upgraded rating of 500 kw.

2. The system must be in an undermoderated state.

3. Reactivity coeffients and other safety-related neutronics parameters (e.g. effective delayed neutron fraction and prompt neutron lifetime) must be such that current UFTR safety levels are either maintained or enhanced.

For the Phase 1 studies, the only permanent UFTR structural modification allowed is the insertion of new grid assemblies into the existing fuel boxes to hold the the cylindrical SPERT fuel in place.

The Phase 2 neutronic studies examined the effects of certain additional UFTR structural changes using the metal-to-water ratio lattice determined in the Phase 1 studies. These changes included systematic alterations in certain region thicknesses and compositions. The primary intention of the Phase 2 studies was to optimize the peak thermal neutron flux level in the UFTR while maintaining an acceptable k_{eff} value. Optimal configurations discovered during these investigations provide a potential means of maintaining, or even enhancing, the current value of the UFTR peak thermal neutron flux, for a given power, despite the change from HEU to LEU fuel.

2.0 University of Florida Training Reactor Description

2.1 Physical Description

The UFTR is a light water-and graphite-moderated, water-cooled reactor and is a variation of the Argonaut type reactor which originally was designed by Argonne National Labortory. The reactor is heterogeneous in design and consists of six fuel boxes surrounded by graphite. The graphite is further surrounded by a biological shield made of removable concrete blocks. A shield tank, containing demineralized water is located at the west end of the reactor. The overall dimensions of the reactor, placement of the fuel boxes relative to other regions, along with the various material regions associated with the reactor are indicated on the horizontal cross-sectional view of the UFTR presented in Figure 1. Two fuel regions are indicated in this figure; each region consists of three fuel boxes separated by control blade shrouds and a void region where the control blades are located.

The UFTR currently employs a highly-enriched (93 wt.%) aluminum-uranium alloy flat plate MTR-type fuel. A cutaway view of the placement of the plate-type fuel in a fuel box is shown in Figure 2. Eleven plates make up an assembly and four assemblies bolted together make up a full fuel box.

The fuel which is proposed to replace the plate-type fuel of the current UFTR is in the form of cylindrical pins and is referred to as SPERT (F-1) type fuel which contains uranium-dioxide fuel pellets enclosed by stainless steel cladding. Figure 3 includes a sectional, isometric view of the SPERT fuel rod as well as a list of physical characteristics. A key feature of the SPERT fuel is its 4.8% enrichment compared to the 93% enrichment of the MTR plate-type fuel.

The number of SPERT fuel pins in a fuel box is dependent upon the selected metal-to-water ratio. For example, the 1.0 M/W ratio configuration has 8 pins per box in the north-south direction and 10 pins per box in the east-west direction for a total of 80 pins per box and 480 pins for the reactor. A cutaway view showing the proposed placement of the SPERT fuel pins inside the existing fuel boxes and restrained by spacer grids is presented in Figure 4.

2.2 Description of Regions Used in Neutronic Calculations

The regions of the UFTR specified to facilitate neutronic calculations are illustrated by two figures: a horizontal cross-sectional view of the reactor is presented in Figure 5 and a vertical view at the location of a fuel box is shown in Figure 6. The centerlines indicated on Figure 5 represent the dividing lines between the north, south, east and west directions. The centerline in Figure 6 divides the core into upper and lower sections. A schematic of the fuel regions in Figure 7 shows that each fuel region includes three fuel boxes. The fuel boxes consist of unit fuel cells, water jacket and aluminum. Magnesium control blade shrouds and a void region which separate the fuel boxes are considered to be part of the fuel region. The material surrounding the fuel region is reactor-grade graphite.

The unit fuel cells consist of four regions: a) uranium dioxide, (fuel), b) gap, c) cladding (304 stainless steel), and d) moderator (water). Figure 8 includes a schematic of a unit fuel cell as well as values for the radii of the various regions of the "equivalent" cylindrical unit fuel cell for the 1.0 metal-to-water ratio lattice.

3.0 Phase 1 Neutronic Calculations

3.1 Introduction

The Phase 1 neutronic studies were directed towards determining an acceptable metal-to-water ratio for the UFTR using the LEU SPERT fuel. Acceptability is based upon satisfying the neutronic design criteria under the design constraint set forth in Section 1.0.

The initial step in the neutronic calculations was to obtain group constants for the various fuel and non-fuel regions in the UFTR. A different set of group constants was obtained for each examined metal-to-water ratio and for each studied power level. Four broad group (3 fast groups and 1 thermal

group) macroscopic neutron cross section data were generated using computer codes based upon approximate solutions to the neutron transport equation. The UFTR thermal group constants were generated using BRT (Battelle-Revised THERMOS)[4] which solves the integral form of the neutron transport equation in one-dimension. The UFTR fast group constants were generated using PHROG[5] which solves a B-1 or P-1 approximation to the energy-dependent transport equation.

Four group diffusion theory calculations were then performed on the UFTR to determine k_{eff}'s and flux and power distributions for various metal-to-water ratio configurations and power levels. One-, two- or three-dimensional diffusion theory calculations were performed using CORA[6] EXTERMINATOR-2[7] and CITATION[8]. CORA is a one-dimensional few group diffusion theory code while EXTERMINATOR-2 is a multigroup two-dimensional diffusion theory code. CITATION is a one-, two- or three-dimensional multigroup diffusion theory code capable of solving fuel depletion problems. For the UFTR analysis, only standard eigenvalue calculations were performed with CITATION; the fuel depletion capabilities were bypassed.

With regard to the one-and two-dimensional four-group diffusion theory calculations on the UFTR, there are two important considerations that should be mentioned. The first is the representation of the perpendicular leakage(s) from all material regions associated with a calculation. This is accomplished by using perpendicular bucklings. Perpendicular bucklings for the various regions were determined by performing detailed buckling iteration calculations. The second consideration is identifying neutronically the most important direction(s) of the system. The north-south direction is neutronically the most important direction of the system since 1) it includes explicitly the important graphite I region located between the two parallel fuel regions and 2) the leakage of neutrons from the north-south direction is larger than from either the east-west or height directions. Neutronically, the east-west direction is the second most important direction due to its superior reflecting abilities compared to the height direction. Consequently, the X-Y plane (north-south/east-west plane) is neutronically the most important plane for the UFTR calculations.

The three-dimensional diffusion theory calculations eliminate the need to identify the most important direction(s) and use perpendicular bucklings to account for perpendicular leakage. However, due to the great expense associated with such a calculation, preliminary one-and two-dimensional scoping calculations were always performed prior to any three-dimensional run.

3.2 Results From Phase 1 k_{eff} Calculations

A number of preliminary calculations at various metal-to-water ratios indicated that the 1.0 M/W ratio configuration for the UFTR would satisfy the design criteria specified in Section 1.0. Three different power levels at 1,100 and 500 kw were examined for the 1.0 M/W ratio lattice; corresponding fuel and moderator temperatures are listed in Table 1. One of the reasons for studying different power levels is because this allows an overall temperature coefficient of reactivity to be determined for the UFTR system. The temperature coefficient of reactivity for the SPERT-fueled UFTR can then be compared with the value for the current UFTR configuration. Although the current UFTR is limited to a power level of 100 kw, the calculation performed at 500 kw is intended to support the

Table 1

Fuel and Moderator Temperatures As A Function of Power for the 1.0 M/W Ratio Lattice

Power Level (kw)	Average Fuel Temperature (°K)	Average Moderator Temperature (°K)
1	303.9	303.3
100	359.7	309.4
500	473.9	333.7

UFTR Conditions: Coolant Flow Rate of 4100 cm^3/sec (65 gals/min) and Inlet Temperature of 303°K.

Table 2

Average Flux Data for the UFTR System taken from CORA North-South run for the 1.0 M/W Ratio at a Power Level of 100 KW

Energy Group	FUEL REGION Average Flux Per Watt (n/cm**2-sec-watt)	Peak-to-Average Flux Ratio
1	4.523E+06	1.070
2	6.311E+06	1.104
3	5.457E+06	1.122
4	2.804E+06	2.093

Peak-to-Average Power Density is 1.909.

$k_{eff} = 1.192$

$\phi_{th}^{max} = 1.07 \times 10^7$ n/cm^2sec watt

feasibility of operating the UFTR system at a higher power level.

As indicated in Section 3.1, the one direction which best represents the UFTR system when performing a one-dimensional neutron multiplication factor calculation is the north-south direction. The CORA neutron multiplication factor and the peak thermal flux obtained for a north-south direction calculation at a power level of 100 kw are 1.192 and 1.07 X 10^7 n/cm^2 sec watt, respectively. The average group fluxes and peak-to-average flux ratios in the fuel region as obtained from this calculation are presented in Table 2.

The one-dimensional CORA calculations provide an initial estimate for the k_{eff} for a specific power level of the UFTR. To further pinpoint the k_{eff} for the system, two-dimensional neutron diffusion theory calculations were performed at a power level of 100 kw using the computer code CITATION. As indicated in Section 3.1, the plane which best represents the UFTR system when performing a two-dimensional k_{eff} calculation is the X-Y plane. The CITATION k_{eff} and peak thermal neutron flux for an X-Y plane calculation at a power level of 100 kw are 1.143 and 9.04 X 10^6 n/cm^2 sec watt, respectively. The average group fluxes and peak-to-average flux ratios for the fuel region for this calculation are presented in Table 3.

Comparison of Tables 2 and 3 shows some differences in results for the one- and-two-dimensional UFTR calculations. For example, the one-dimensional k_{eff} is 4.3% higher than the two-dimensional value while average fluxes in the fuel region as predicted by the one-dimensional calculation are as much as 20% greater than corresponding values from the two-dimensional calculation. These differences can be attributed to limitations associated with the technique of using perpendicular bucklings to account for perpendicular leakage. The method for using perpendicular bucklings to account for the perpendicular leakage has no theoretical justification in the general case [6]. It is nearly valid for some idealized systems such as large, bare homogeneous reactors of certain regular shapes but the configuration of the UFTR does not fit into this category.

The above-cited differences between the one and two dimensional neutron multiplication factors indicates the need for a three-dimensional calculation. Such a calculation was performed on the UFTR system at a power level of 100 kw using the computer code CITATION. The neutron multiplication factor obtained from this three-dimensional calculation is 1.069; the peak thermal neutron flux level is 9.83 X 10^6 n/cm^2 sec watt. Average group fluxes and peak-to-average flux ratios for the fuel region for this three-dimensional calculation appear in Table 4.

An important conclusion that can be obtained from the results of the three-dimensional calculation is that fairly significant errors arise from the representation of perpendicular leakage by means of perpendicular bucklings for the UFTR. For example, comparison of data in Tables 2, 3 and 4 shows that one- and two-dimensional k_{eff} values differ from the three-dimensional result by 12% $\Delta k/k$ and 7% $\Delta k/k$, respectively. For preliminary scoping or parametric comparison studies this loss of accuracy for the one-dimensional or two-dimensional calculations as compared to the three-dimensional analysis can be justified by the significant cost savings for the former as compared to the latter. Final assessment of a design for the UFTR, however, requires results from the expensive three-dimensional analysis.

Table 3

Average Flux Data for the UFTR System taken from the CITATION X-Y Plane Run for the 1.0 M/W Ratio at a Power Level of 100 KW

Energy Group	FUEL REGION Average Flux Per Watt (n/cm**2-sec-watt)	Peak-to-Average Flux Ratio
1	3.867E+06	1.171
2	5.326E+06	1.176
3	4.597E+06	1.187
4	2.331E+06	1.656

Peak-to-Average Power Density is 1.671.

$k_{eff} = 1.143$

$\phi_{th}^{max} = 9.04 \times 10^{6}$ n/cm^2 sec watt

Table 4

Average Flux Data for the UFTR System taken from the Three-Dimensional CITATION run for the 1.0 M/W Ratio at a Power Level of 100 KW

Energy Group	FUEL REGION Average Flux Per Watt (n/cm**2-sec-watt)	Peak-to-Average Flux Ratio
1	4.183E+06	1.601
2	5.660E+06	1.572
3	4.794E+06	1.586
4	2.306E+06	1.874

Peak-to-Average Power Density is 1.817.

$k_{eff} = 1.069$

$\phi_{th}^{max} = 9.83 \times 10^{6}$ n/cm^2 sec watt

The three-dimensional neutron diffusion theory calculation for the 1.0 M/W ratio yields a k_{eff} of 1.069 at 100 kw; the k_{eff} at 500 kw is found to be 1.064. Thus, this metal-to-water ratio satisfies the first design criterion, even at the proposed increased power rating. In addition, this metal-to-water ratio results in a safe undermoderated system[9] to satisfy the second design criterion as well. The suitability of the 1.0 M/W ratio with regard to the third design criterion is examined in the next two sections.

3.3 Temperature Coefficient of Reactivity

The temperature coefficient of reactivity is calculated for the UFTR using the equation given below so that

$$\alpha_T = \frac{\left(\frac{\Delta k}{k}\right)_{\substack{\text{from moderator} \\ \text{temperature change}}} + \left(\frac{\Delta k}{k}\right)_{\substack{\text{from fuel} \\ \text{temperature change}}}}{\Delta T_{\text{moderator}}}$$

comparisons can be made with the experimental value of the current UFTR. Using the difference in moderator temperature between 1 kw and 100 kw along with the corresponding k_{eff}'s obtained from the one-dimensional CORA runs for the 1.0 M/W ratio configuration yields α_T = -3.24E-04/°K = -1.80E-04/°F. The experimental temperature coefficient of reactivity for the current UFTR for the same power range is[10]: α_T = -3.0E-05/°F.

Since operation of the UFTR system is planned for 500 kw the temperature coefficient of reactivity for the power range of 1 kw to 500 kw, using the above equation, is also of interest. The result of this calculation is: α_T = -2.49E-04/°K =-1.38E-04/°F.

3.4 Effective Delayed Neutron Fraction and Prompt Neutron Lifetime

Standard first-order perturbation theory calculations were performed on the UFTR for the 1.0 M/W ratio using the two-dimensional computer code EXTERMINATOR-2 in the X-Y plane. The purpose of the perturbation calculations was to obtain the effective delayed neutron fraction and effective prompt neutron lifetimes for the UFTR. These values are necessary not only for comparison with the current UFTR values but also for neutron dynamic and reactor safety studies that are currently being performed on the proposed SPERT-fueled UFTR.

The effective delayed neutron fractions for each delayed group for the UFTR system as obtained from the perturbation calculations are printed in Table 5. The total effective delayed neutron fraction is 6.78×10^{-3}. The delayed neutron effectiveness factor, which is the ratio of the effective delayed neutron fraction to the actual delayed neutron fraction, is 1.025 for the SPERT-fueled UFTR. The first order perturbation theory approximation to the effective prompt neutron lifetimes for the various material regions of the X-Y plane are presented in Table 6. The total effective prompt neutron lifetime, obtained by summing up the values for each region, is 2.8×10^{-4} sec.

Table 5

Effective Delayed Neutron Fractions for the UFTR System Using a 1.0 M/W Ratio*

Group	Effective Delayed Neutron Fraction
1	2.85E-04
2	7.77E-04
3	2.68E-03
4	1.33E-03
5	1.48E-03
6	2.20E-04
Total	6.78E-03

*Calculated from the Computer Code EXTERMINATOR.

Current UFTR value: 7.05E-03

Table 6

Effective Prompt Neutron Lifetimes for the UFTR System Using a 1.0 M/W Ratio*

Material Region	Effective Prompt Neutron Lifetime (sec)
Graphite I	5.61E-05
Graphite II	9.15E-05
Graphite III	5.09E-05
Graphite IV	6.51E-05
Fuel	1.58E-05
Shield Tank	3.76E-07
Total	2.80E-04

*Calculated from the Computer Code EXTERMINATOR.
Current UFTR value: 2.42E-04 sec

The current UFTR has a calculated total effective delayed neutron fraction of 7.05E-03 [11] and a delayed neutron effectiveness factor of 1.085. The calculated effective prompt neutron lifetime is 2.42E-04 sec [11].

3.5 Comparison of Flux Profiles

The use of low-enriched fuel in training reactors has brought concern that the peak thermal flux would be severely reduced from the value for the same reactor using high-enriched fuel. The purpose of this section is to make some comparisons between the flux profiles of the SPERT-fueled UFTR system with those of the current UFTR.

The flux profiles in the north-south direction for the SPERT-fueled UFTR with the 1.0 M/W ratio lattice are presented in Figure 9. These flux profiles were obtained from one-dimensional CORA calculations at a power level of 100 kw. The group-dependent fluxes along the north-south direction for the current UFTR are presented in Figure 10 [12]. These profiles were also calculated from the one-dimensional computer code CORA at a power level of 100 kw.

From Figures 9 and 10, the peak thermal neutron flux for the SPERT-fueled UFTR system using the 1.0 M/W ratio is 1.07 X 10^7 n/cm^2 sec watt while the corresponding value for the current UFTR is 1.22 X 10^7 n/cm^2 sec watt. Thus, at 100 kw the reduction in peak thermal neutron flux that occurs due to a replacement of the HEU MTR fuel with the LEU SPERT is only 12%.

3.6 Phase 1 Conclusions

In Section 3.2 it was shown that the SPERT-fueled 1.0 M/W ratio UFTR system satisfies the first two design criteria, namely that the system k_{eff} be in the range of 1.03 to 1.09 when unrodded and that the system be in an undermoderated state. These criteria were satisfied not only at the current power rating of 100 kw but also at the proposed limit of 500 kw.

The third criterion requires that reactivity coefficients and other safety-related neutronic parameters be such that current UFTR safety levels be maintained or enhanced. In Section 3.3 it was shown that at 100 kw, the overall temperature coefficient of reactivity for the SPERT-fueled UFTR is six times larger (more negative) than for the current UFTR. Even at 500 kw, the SPERT-fueled UFTR has a temperature coefficient of reactivity that is over four times larger (more negative) than for the current UFTR at 100 kw. In Section 3.4 it was observed that the effective delayed neutron fraction for the SPERT-fueled UFTR is slightly smaller, by about 16%, than for the current UFTR. The slight decrease in effective delayed neutron fraction represents no significant safety problem and in fact is more than compensated for by the much larger (more negative) overall temperature coefficient of reactivity and larger effective prompt neutron lifetime.

In going from the current HEU MTR-fueled system to the LEU SPERT-fueled 1.0 M/W ratio configuration, there is about a 12% decrease in the peak thermal neutron flux (units of n/cm^2 sec watt) in the UFTR. However, since upgrading to a 500 kw power level could easily be accomplished with the SPERT-fueled system, the peak thermal neutron flux in the UFTR could actually be increased by about 340% relative to the current value.

The above conclusions are based on an analysis in which the only allowed structural modification to the current UFTR is the insertion of new grid assemblies into the existing fuel boxes to hold the cylindrical SPERT fuel. The Phase 2 neutronic studies, covered in the next section, examined the effect that certain additional structural changes have on neutron fluxes and k_{eff} values for the UFTR using the SPERT fuel 1.0 M/W ratio lattice.

4.0 Phase 2 Neutronic Calculations

4.1 Introduction

The primary intention of the Phase 2 studies was to optimize the peak thermal neutron flux level in the UFTR while maintaining an acceptable k_{eff} value. A variety of modifications involving the fuel and non-fuel regions of the UFTR were examined. The fuel region composition, however, was maintained constant by fixing the SPERT fuel lattice to the 1.0 M/W ratio configuration. Consequently, reactivity coefficients and other safety-related neutronics parameters are essentially unchanged from the values reported in Section 3.0. Optimal configurations discovered in these investigations thus provide a potential means of maintaining, or even enhancing, the current value of the UFTR peak thermal neutron flux, for a given power level, despite the change from HEU to LEU fuel.

The calculation of k_{eff} and peak thermal neutron flux for the various modifications to the UFTR were determined by using the one-dimensional CORA code. Justification for the use of the one-dimensional code in this type of comparative scoping calculation is based upon a cost versus accuracy consideration. Desired information on relative effects of various modifications can be determined with sufficient accuracy, for preliminary studies, from one-dimensional calculations.

4.2 Phase 2 Results

Although many modifications were investigated, results are presented only for two of the more interesting cases. In the first modification, the half-thickness of the graphite I region is varied while the fuel region width is maintained fixed at the standard value of 13.34 cm (see Figure 5, the north-south direction). The thickness of the graphite III region is varied so as to compensate for changes in the graphite I region thickness and thus keep the overall UFTR dimension constant in the north-south direction.

The results of analysis of Modification 1 on the 1.0 M/W ratio configuration are presented in Figure 11. For graphite I regions of less than 15.24 cm half-thickness, a reduction in k_{eff} and the peak thermal neutron flux relative to the standard values occurs. Graphite I region half-widths larger than 15.24 cm cause an increase in k_{eff} and the peak thermal neutron flux from the standard values thus indicating one possible way of simultaneously raising the peak thermal neutron flux and k_{eff}. Although not shown in Figure 11, the curves tracing the variation of k_{eff} and maximum thermal neutron flux eventually peak and then decrease for a continuing increase in the thickness of the graphite I region. The decrease is caused by the decoupling of the two fuel regions that are separated by the central graphite I region.

For Modification 2, a water region is added on the inner side of the fuel region (see Figure 5, north-south direction). The graphite I and graphite III region thicknesses are fixed at the existing 30.48 cm and 47.63 cm, respectively. The water region thickness is increased from 0 cm while the fuel region thickness is decreased from 13.34 cm in such a manner so as to maintain the overall system dimension constant in the north-south direction. Results from analysis of Modification 2 (Figure 12) indicate that both k_{eff} and peak thermal neutron flux reach a maximum and then decrease with increasing water region thickness. The maximum increase in k_{eff} relative to the standard configuration value is about 1% $\Delta k/k$ and this occurs at a water thickness of 0.6 cm. The maximum increase in peak thermal neutron flux relative to the standard configuration value is about 210% at a water thickness of around 4 cm.

Practically speaking, the water region on the inner side of the fuel region can be achieved by removing the innermost (east-west) row(s) of SPERT fuel. For the 1.0 M/W ratio configuration, the removal of one row of fuel would yield a water region thickness of about 1.5 cm (see Figure 8). From Figure 12, this implies a 120% increase in peak thermal neutron flux and a 0.5% decrease in k_{eff}. The three-dimensional k_{eff} prediction for one row of fuel removed is thus projected to be 1.064 at 100 kw and 1.059 at 500 kw. Both of these values are within the range specified by the first design criterion. The removal of two rows of fuel yields a water region thickness of about 3 cm and an increase in the peak thermal neutron flux by about 195% compared to the standard configuration value. However, there is about a 7.8% decrease in k_{eff}. This implies a three-dimensional k_{eff} prediction of about 0.985 at 100 kw which is well below the range specified by the first design criterion.

5.0 Summary

The results of the Phase I studies indicate that a 1.0 M/W ratio satisfies the neutronic design criteria set forth for the SPERT-fueled UFTR system both at the current 100 kw rating and also at the proposed 500 kw rating. These results also show that there would be changes in the flux profiles as compared to those of the current UFTR. For a given power level, the calculations show that with the SPERT fuel there would be a small reduction (12%) in the peak thermal neutron flux level as compared to the current UFTR, if the existing fuel boxes are completely filled with fuel rods.

The results of the Phase 2 studies show that either Modification 1 or 2 could be used to compensate for the 12% decrease in peak thermal neutron flux incurred by going from the HEU MTR fuel to the LEU SPERT fuel. In fact, either one of these modifications could lead to a net increase in the peak thermal neutron flux for the UFTR without adversely affecting k_{eff} or reactivity coefficients and other safety-related neutronics parameters. For example, for the SPERT-fueled 1.0 M/W ratio lattice at 500 kw, the addition of a 1.5 cm water region to the inner side of the fuel region (Modification 2) could lead to a peak thermal neutron flux that is about 460% higher than for the current UFTR. The addition of the water region could be achieved by removing the inner (east-west) row of SPERT fuel from each fuel box.

Finally, calculations have been performed to address the proliferation concern with regard to plutonium production for the SPERT-fueled UFTR. For the current UFTR, the total plutonium production rate is calculated to be around 0.2

mg/full power day at 100 kw. For the SPERT-fueled 1.0 M/W ratio configuration, the calculated total plutonium production rate is around 50 mg/full power day at 100 kw and around 250 mg/full power day at 500 kw. Assuming a constant plutonium generation rate at the 500 kw value leads to a requirement of 11 full power years of operation in order to achieve a total plutonium production of 1 kg. At a 25% duty cycle, it would thus take 44 years to achieve a total plutonium production of 1 kg. Because of plutonium consumption, the minimum time to achieve a net production of 1 kg of plutonium would be significantly longer than 44 years. Consequently, it is concluded that plutonium production poses no significant proliferation problem for the LEU SPERT-fueled UFTR.

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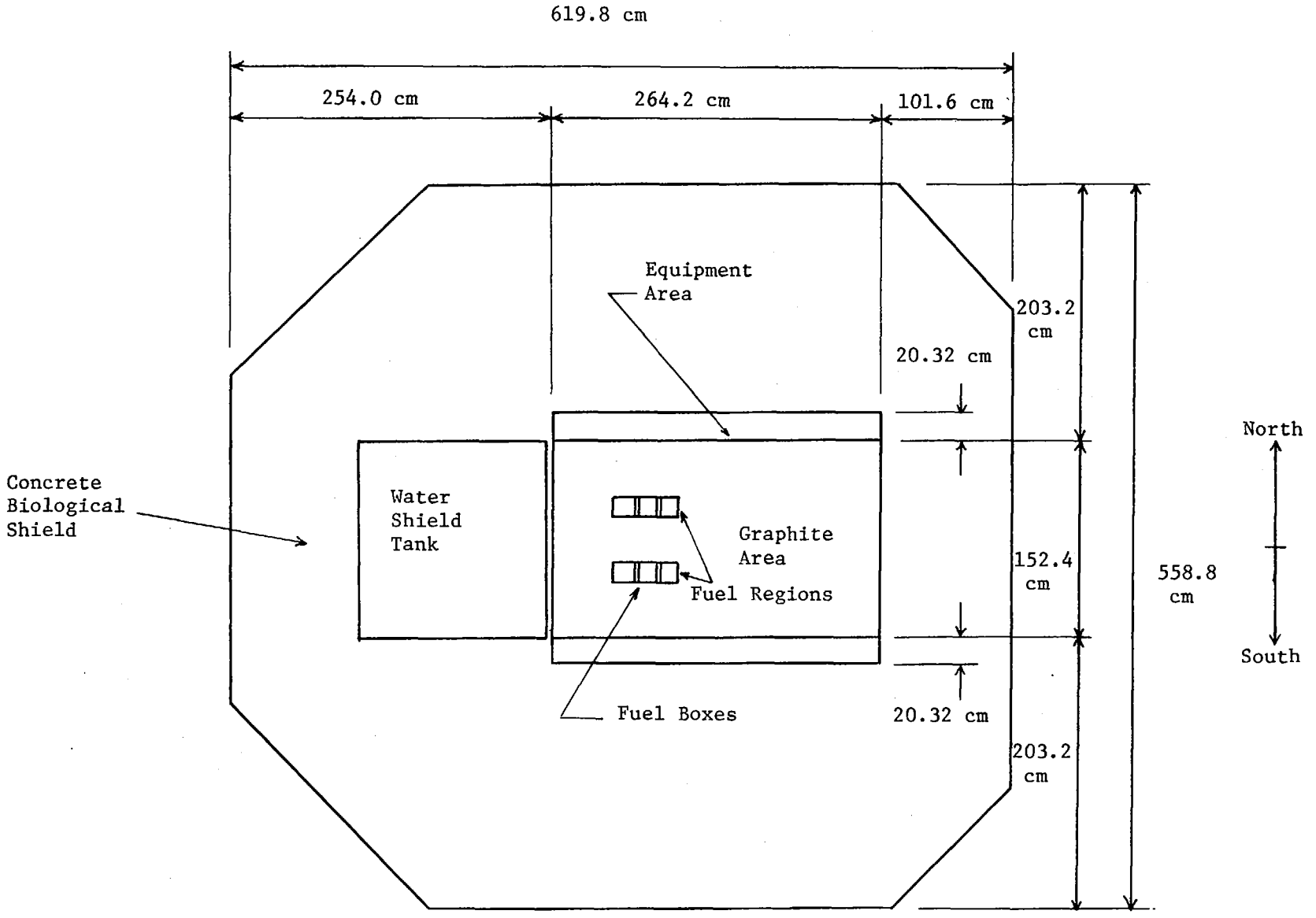


Figure 1 University of Florida Training Reactor Horizontal Cross-Sectional View

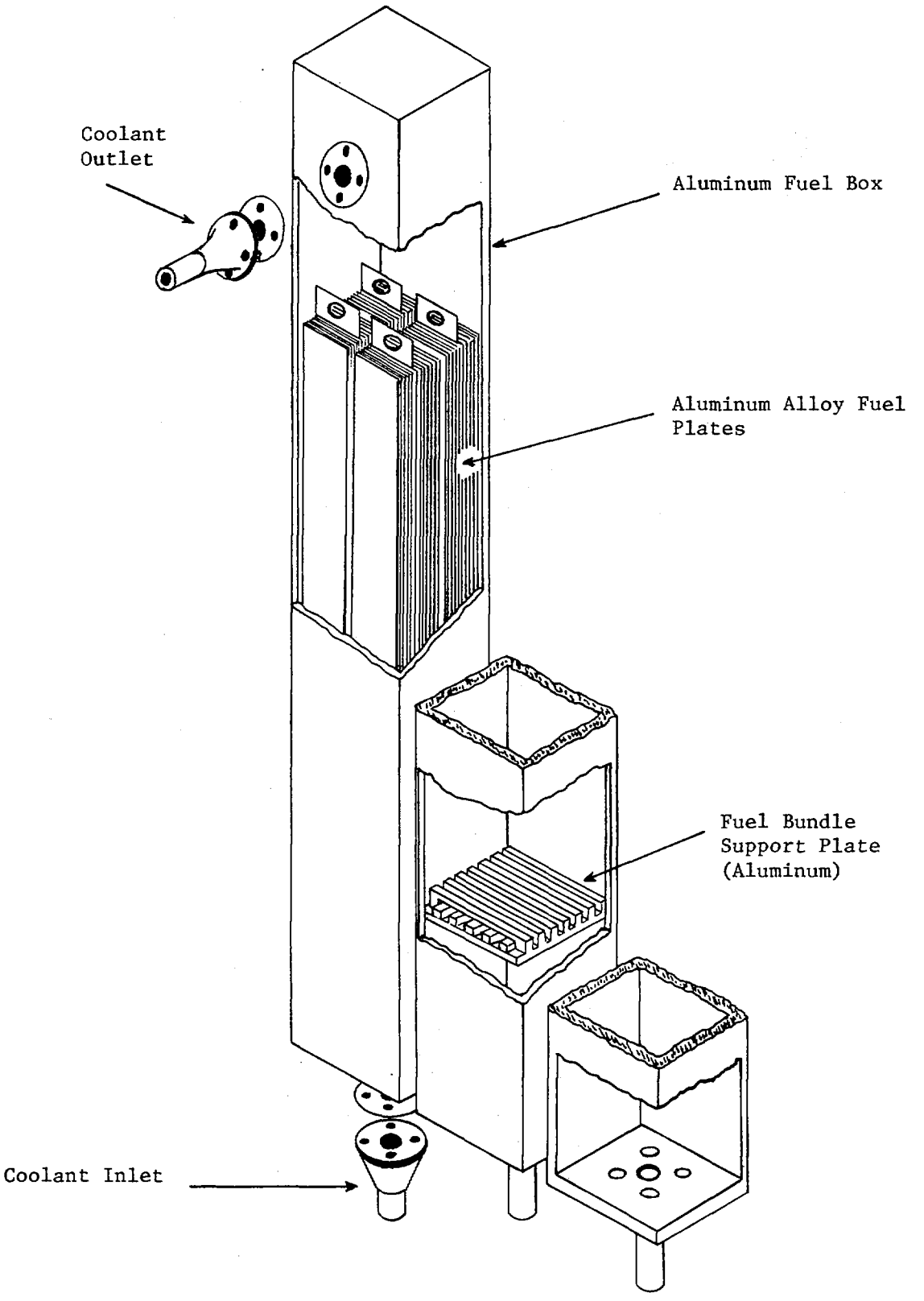
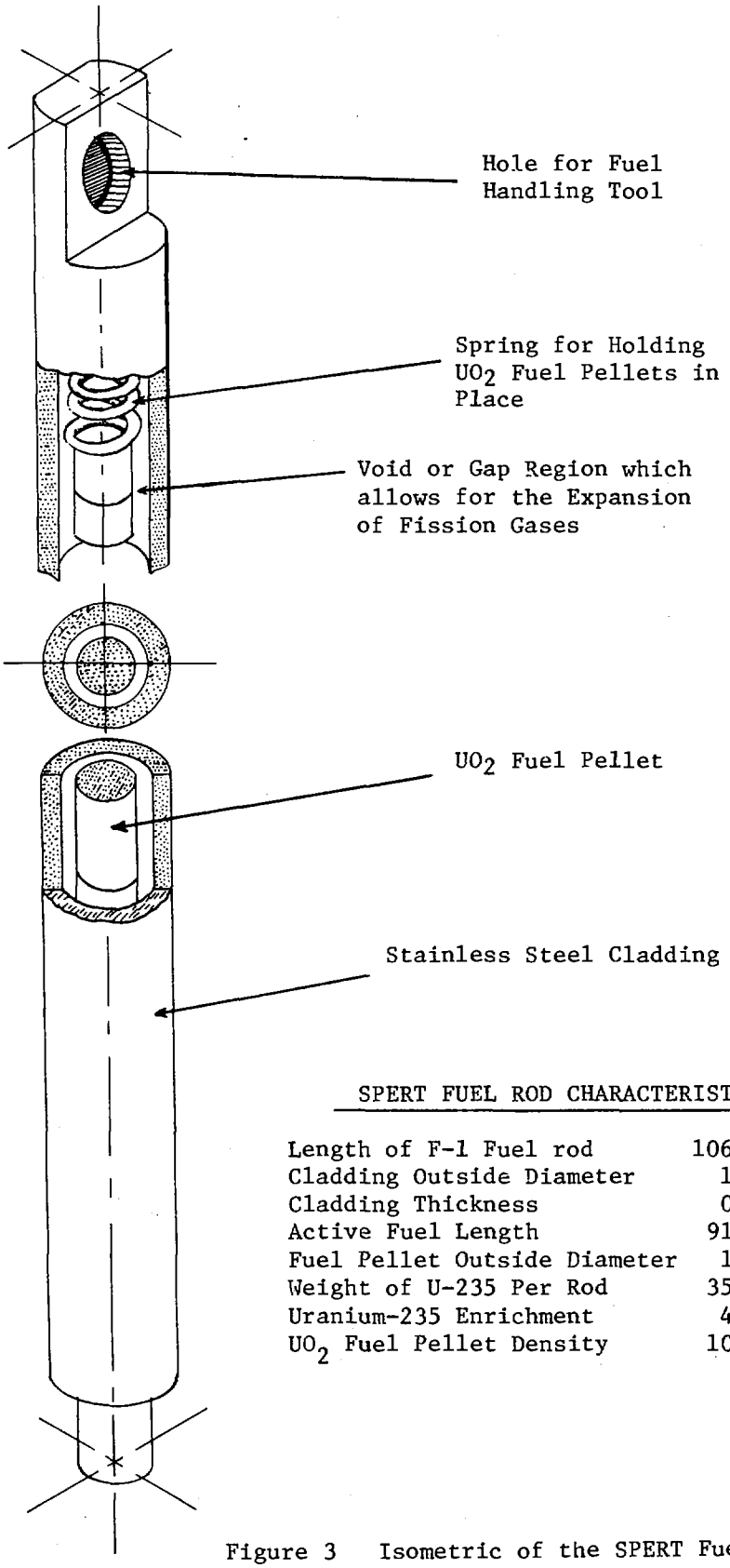


Figure 2 Isometric of University of Florida Training Reactor Fuel Boxes Showing the Position of the Aluminum-Uranium Alloy Fuel Plates



SPERT FUEL ROD CHARACTERISTICS

Length of F-1 Fuel rod	106.05	cm
Cladding Outside Diameter	1.184	cm
Cladding Thickness	0.051	cm
Active Fuel Length	91.44	cm
Fuel Pellet Outside Diameter	1.067	cm
Weight of U-235 Per Rod	35.2	gm
Uranium-235 Enrichment	4.81	wt%
UO_2 Fuel Pellet Density	10.08	g/cc

Figure 3 Isometric of the SPERT Fuel Rod Containing Uranium-Dioxide Fuel Pellets

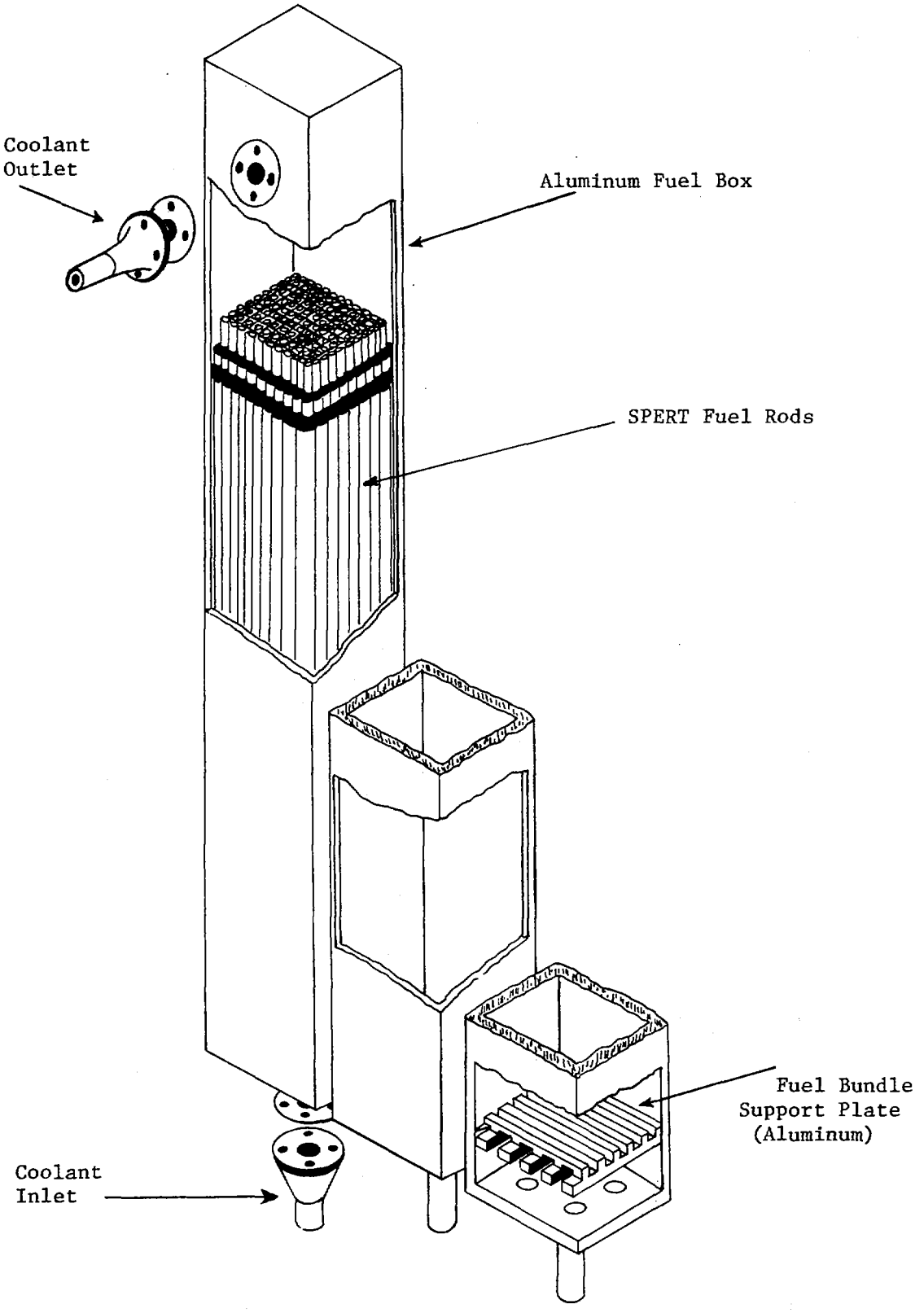


Figure 4 Isometric of University of Florida Training Reactor Fuel Boxes Showing the Position of the Stainless Steel Clad UO_2 Pellet SPERT Fuel Rods

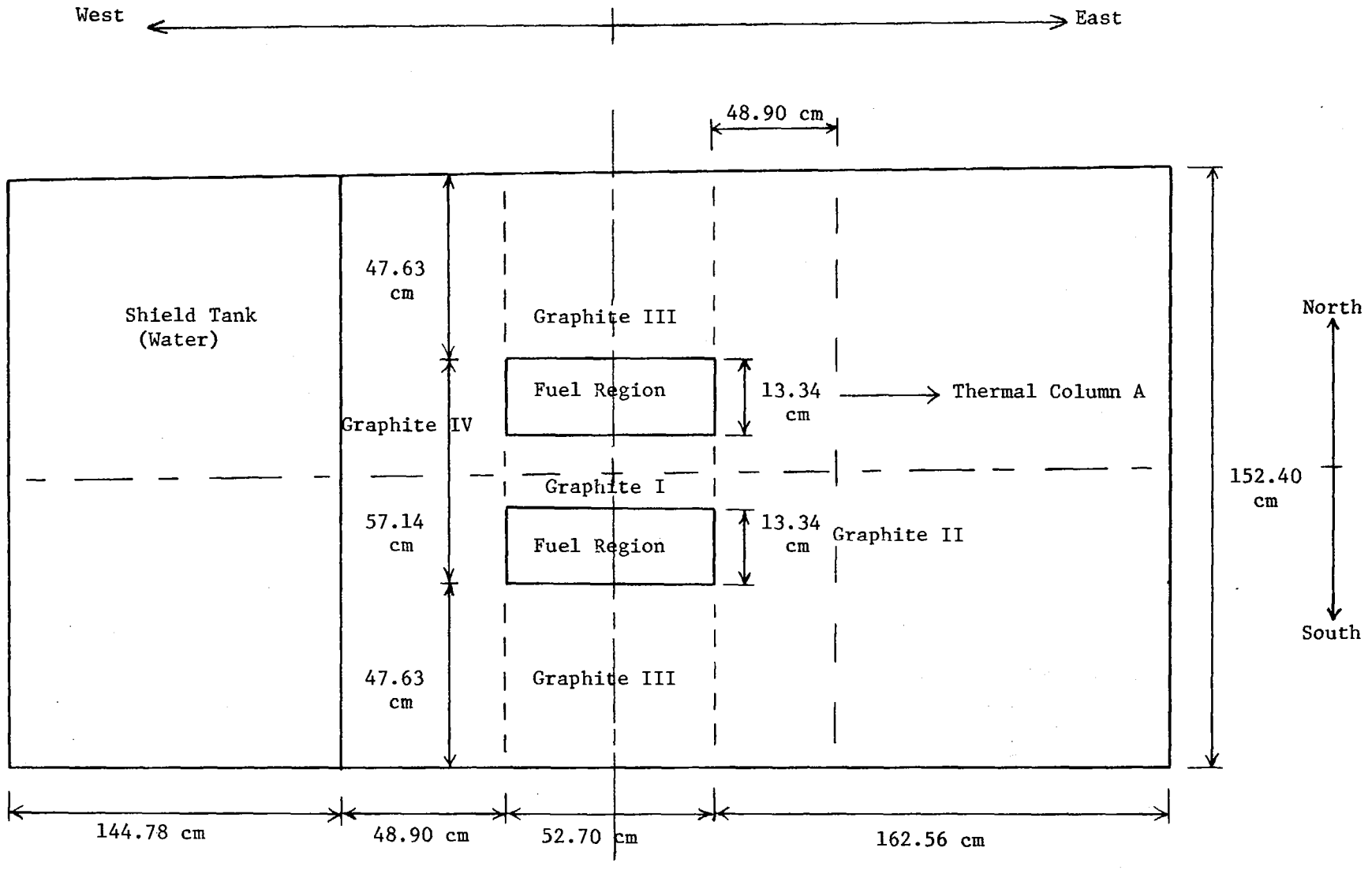


Figure 5 Horizontal Cross-Sectional View of UFTR Used in Neutronic Calculations

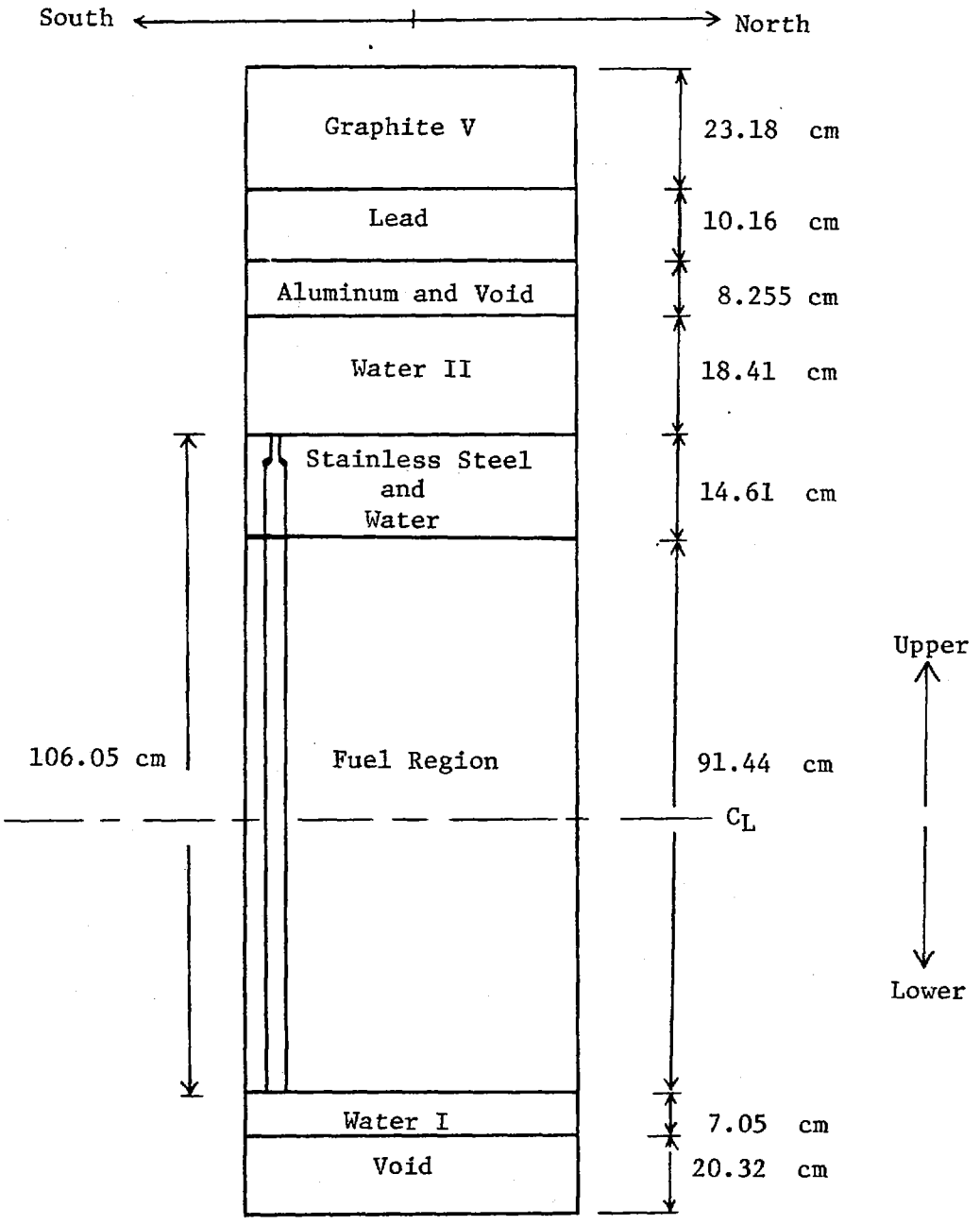


Figure 6 Cross-Sectional View of the Height Direction at the Location of a Fuel Box

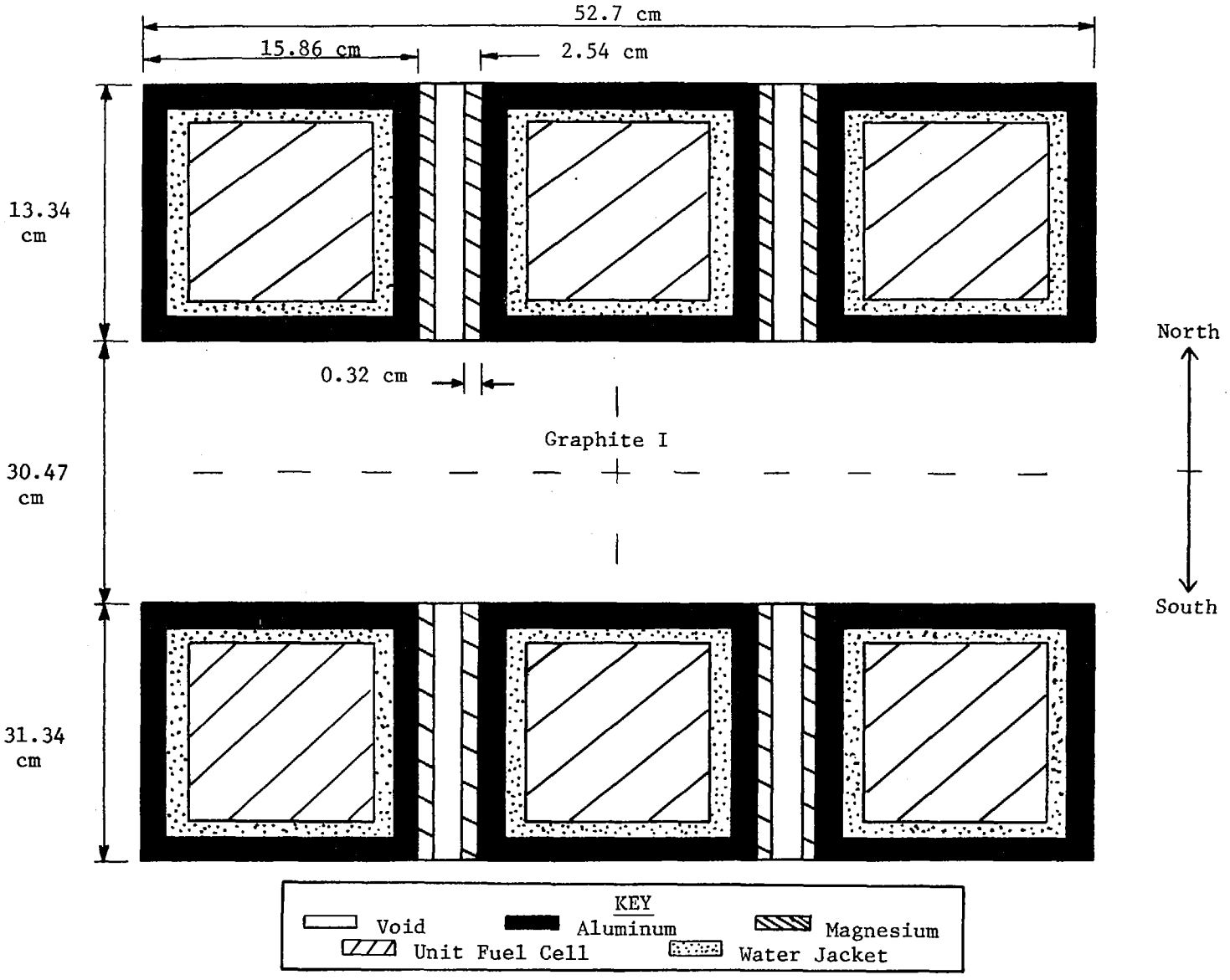
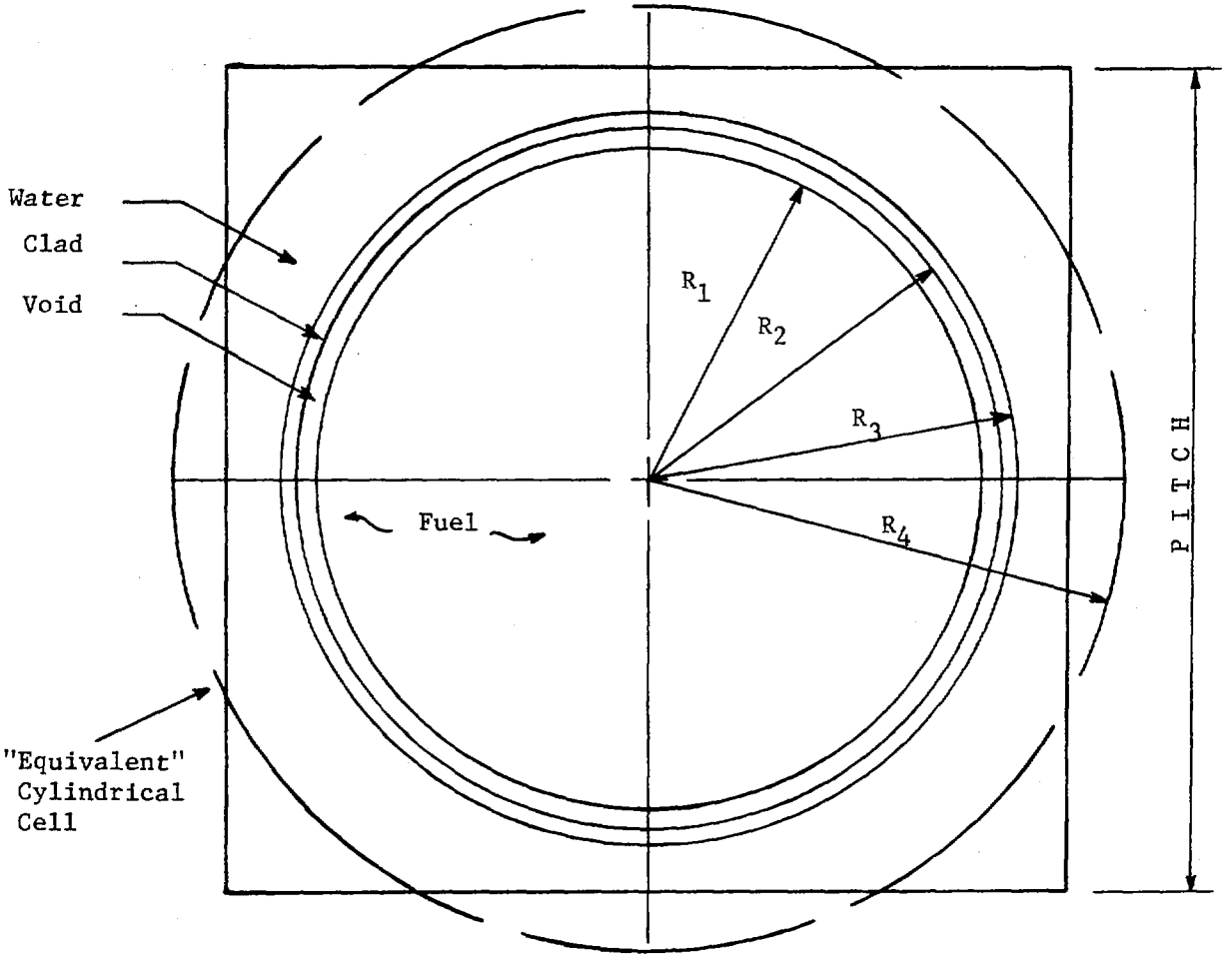


Figure 7 General Schematic (Vertical View) of the UFTR System Fuel Regions



R_1-Fuel Radius = 0.5334 cm
R_2-Void Radius = 0.5410 cm
R_3-Clad Radius = 0.5918 cm
R_4-Water Radius = 0.8369 cm (for 1.0 M/W ratio lattice)
Pitch = 1.4834 cm (for 1.0 M/W ratio lattice)

Figure 8 UFTR Unit Cell Geometry For SPERT fuel

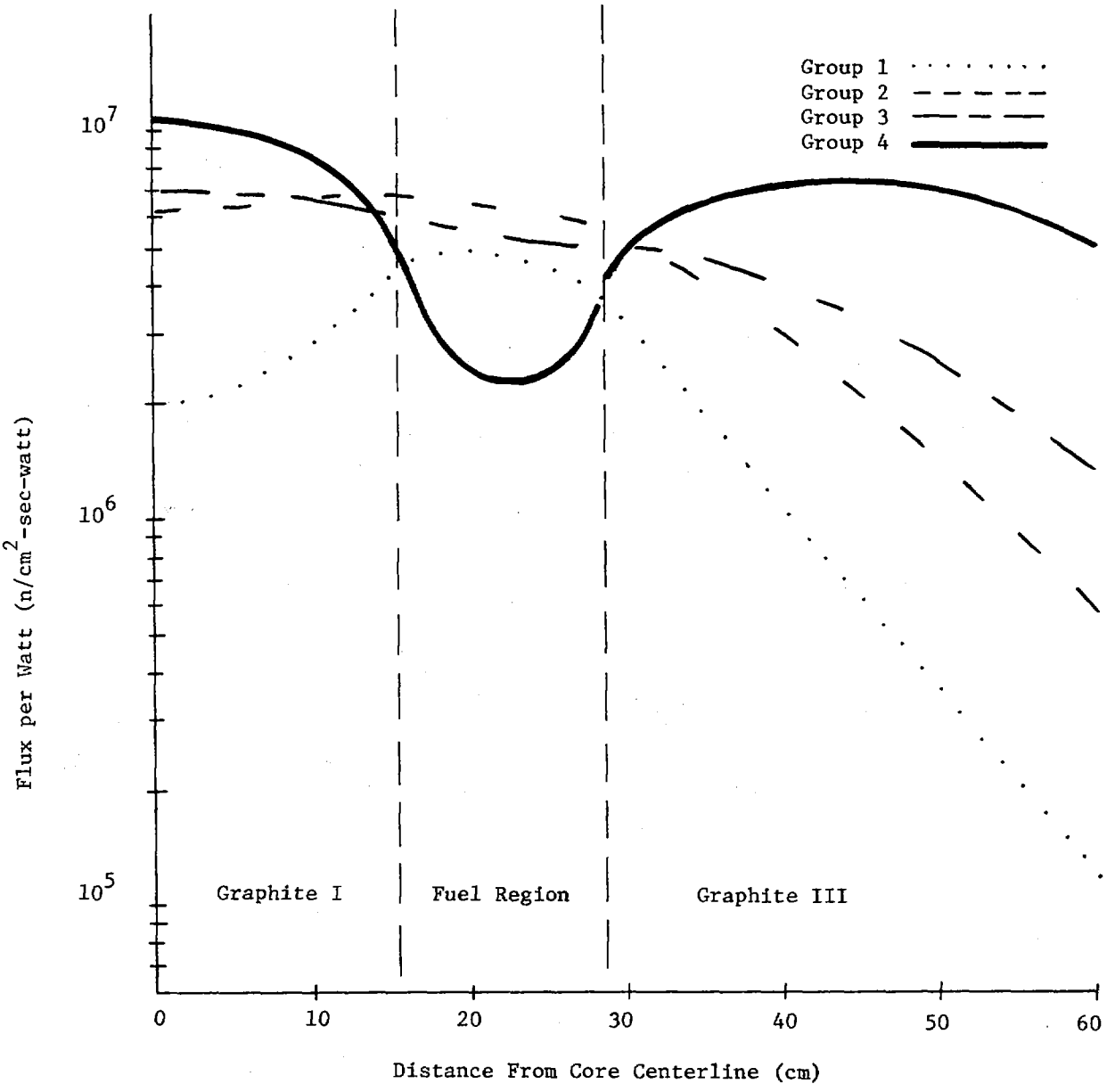


Figure 9 CORA Group-Dependent Fluxes Along the North-South Direction at 100 kw for the SPERT fuel 1.0 M/W Ratio Configuration.

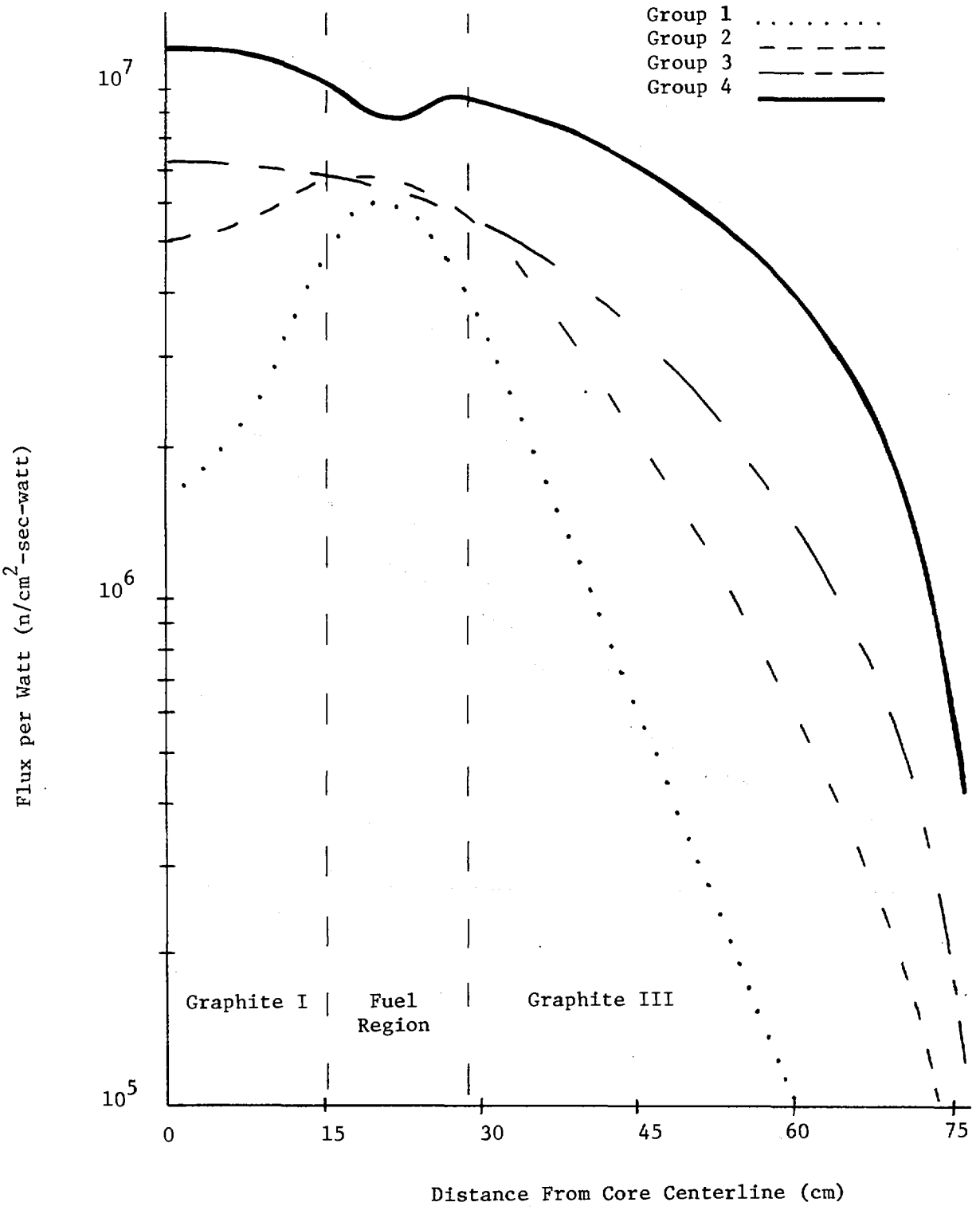


Figure 10 Group-Dependent Fluxes Along the North-South Direction for the Current UFTR System at 100 kw as calculated by CORA

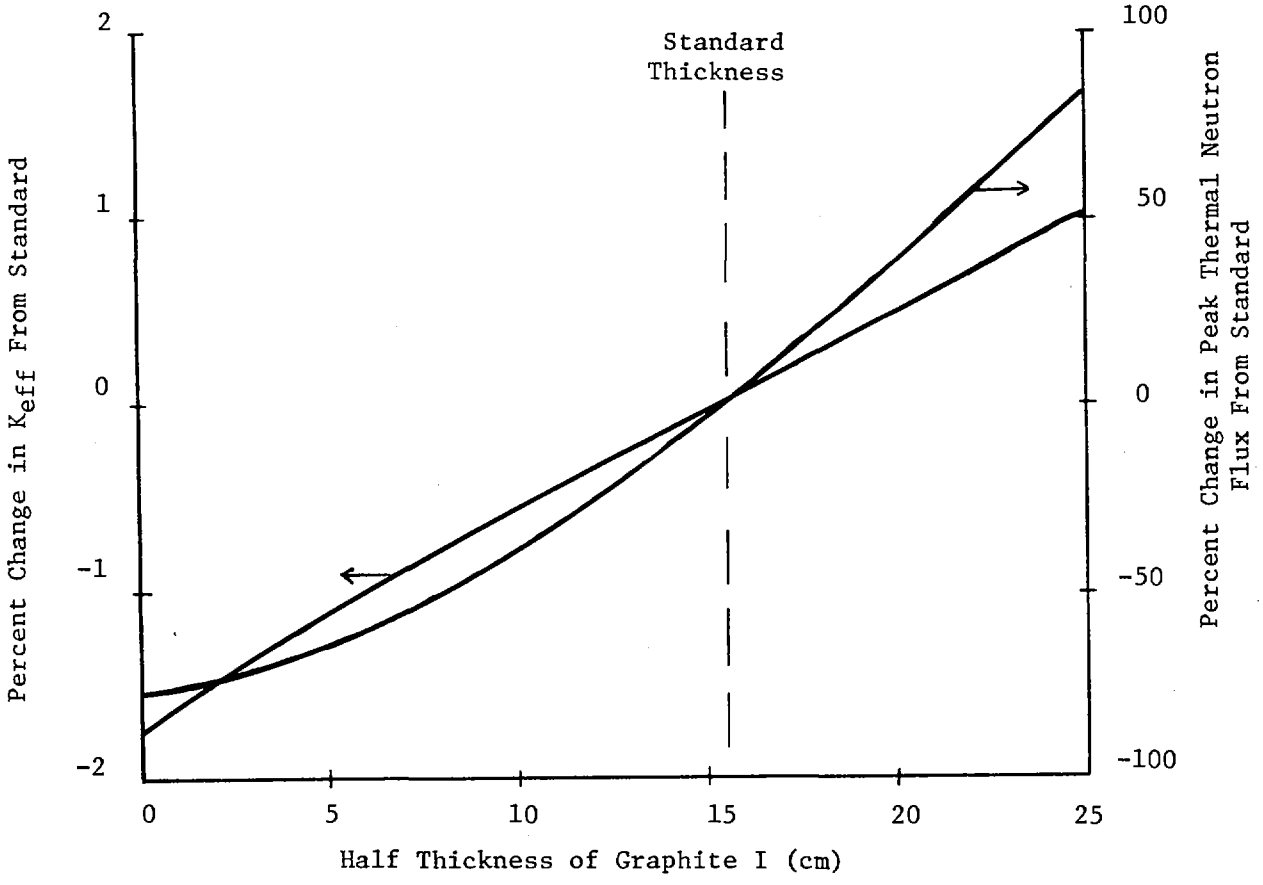


Figure 11 Graphical Representation of Results From Analysis of Modification 1 for 1.0 M/W Ratio

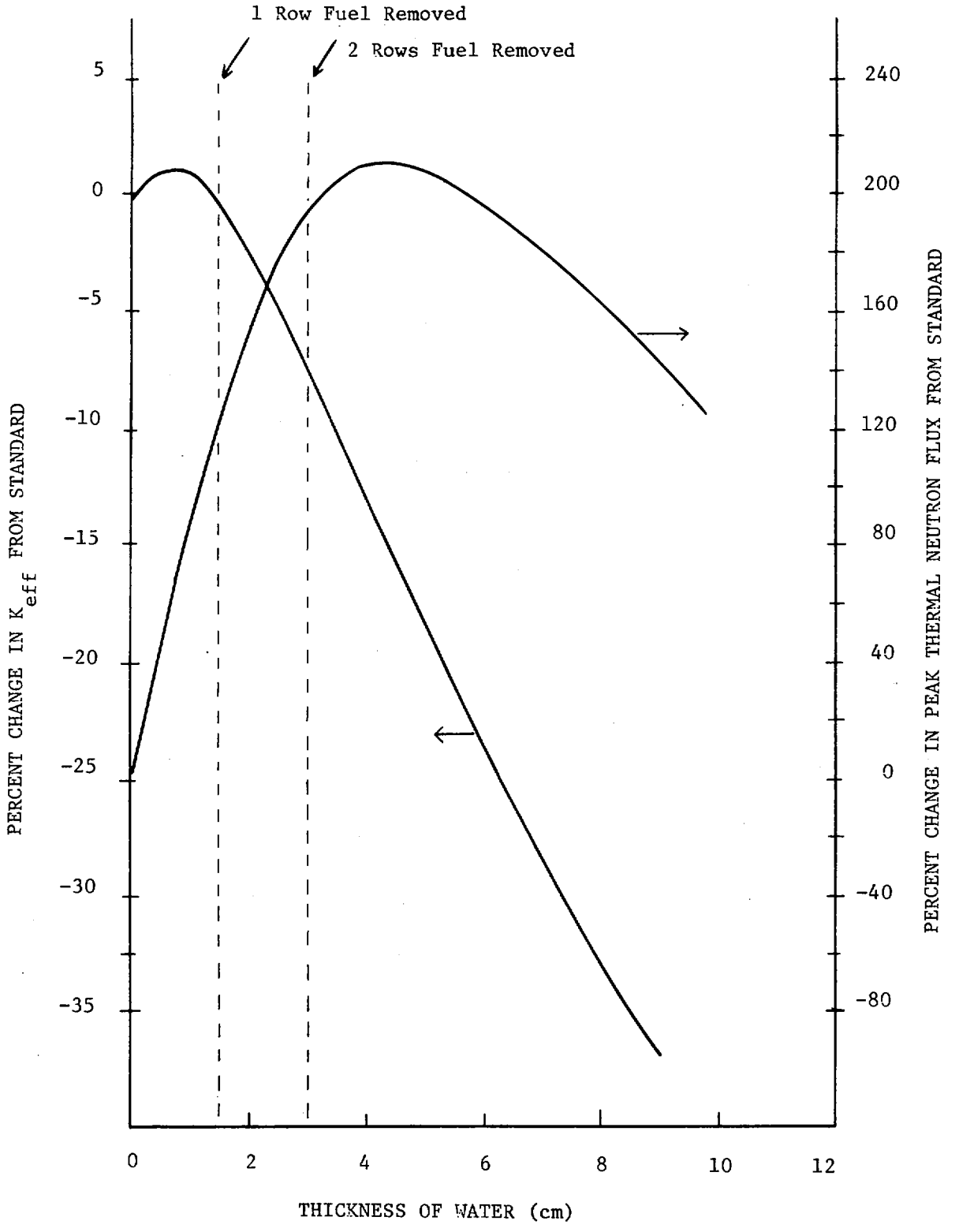


Figure 12 Graphical Representation of Results From Analysis of Modification 2 for the 1.0 M/W Ratio