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Office of Nuclear Reactor Regulation, Director (Post 870411

SUBJECT: Application for amend to License R-61, revising Tech Specs to reflect fuel-related changes to SAR & revised fuel dimensions.

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WORCESTER
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INSTITUTE

Worcester
Massachusetts 01609
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September 17, 1987

Director of Nuclear Reactor Regulations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attn: Document Control Desk

Re: License No. R-61
Docket No. 50-134

Dear Sirs:

As part of the application to the NRC for authorization to convert the fuel of the WPI Nuclear Reactor Facility to low enriched uranium, enclosed please find five copies of the fuel related changes to the WPI Reactor Safety Analysis Report, as well as a list of the changes and a brief rationale for each of the changes.

Also enclosed is Section 4.4 of the WPI Technical Specification of which the fuel dimensions were changed to those of the LEU elements proposed. Please note that this is the only technical specification change necessary as a direct result of fuel conversion.

Finally, please find enclosed three copies of "Analyses for Conversion of the Worcester Polytechnic Institute Reactor from HEU to LEU Fuel", which is a discussion of the approach and technique used by Argonne National Laboratory in performing the safety analyses for this conversion.

If we can supply further information, please let us know.

Sincerely,

Thomas H. Newton Jr.

Thomas H. Newton, Jr.
Director,
Nuclear Reactor Facility

THN:ama

cc w/enc.: Prof. J. A. Mayer, J. Michaels, WPI Project Manager

A020
1/1

cc wo/enc.: Prof. R. Goloskie, Prof. D. N. Zwiep. Provost R. Gallagher
Mr. Donald Haverkamp, USNRC
Mr. Thomas Elsasser, USNRC
Mr. William J. Raymond, Vermont Yankee Nuclear Power Plant

COMMONWEALTH OF MASSACHUSETTS
WORCESTER, SS. September 18, 1987

Then personally appeared the above-named, THOMAS H. NEWTON, JR., and signed the foregoing instrument in my presence, declaring same to be his free act and deed.

Marcia E. Kuederis
MARCIA E. KUEDERIS, NOTARY PUBLIC

My comm exp. 9-25-88

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WPI SAFETY ANALYSIS REPORT
Changes for Conversion to LEU

Section		Change and Rationale
2.2		Final sentence. Quantities and enrichment of the fuel changed to that of LEU.
2.3	p.10	Fuel enrichment, fissionable material quantity to that of LEU.
2.3	p.10	Thermal Characteristics: Power peaking factor, hot channel factor, specific power, gamma heat. Values changed to reflect ANL thermal-hydraulic calculations. These values do not represent a large change and do not significantly affect safe operation of the reactor.
2.3	p.10	Nuclear characteristics: Thermal flux, fast flux, critical mass, temperature and void coefficients, prompt neutron lifetime. All as a result of ANL calculations. Flux changes reflect slight hardening of LEU fuel flux, critical mass reflects fuel change, temperature and void coefficients have slight change to the conservation side, and the prompt neutron lifetime is decreased by about 13%.
2.3	p.11	Standard Fuel: enrichment and plate dimensions.
4.1.1		Fuel element description changed to LEU fuel elements.
5.1		Calculational description expanded to include ANL calculations.
5.2.1		Critical mass and enrichment changed to LEU fuel.
5.2.2		Neutron fluxes changed to ANL calculations reflecting slight flux hardening in the LEU fuel. Note figures 6 and 7 also changed.
5.2.3		Blade worths recalculated. These values represent no change from HEU, but are part of the amendment request submitted September 10, 1987 as a result of the ANL calculations.
5.2.5		Section rewritten to include temperature, void and Doppler coefficients, eliminating the reference to the Spanish OPR and including the HEU values for comparison. The only changes in the values for the LEU fuel are more conservation. Figures 8 - 10 also changed.
5.2.6		Neutron lifetime changed to LEU fuel value.
5.2.7		Reactivity values for postulated core geometry alternations changed to reflect LEU fuel as calculated by ANL.

7.3.1		Fuel heat capacity for LEU fuel about 2% higher than that of HEU (also changed to metric units).
7.4.2	p.44	Loss of coolent with one damaged fuel plate - claculations changed slightly using LEU fuel dimensions.
Figure 22		Fuel element drawing changed to LEU (18 plate) element.

4.0 SITE AND DESIGN FEATURES

4.1 Site

The reactor and associated equipment is housed in the Washburn Laboratories located between West Street and Boynton Street on the campus of Worcester Polytechnic Institute in Worcester, Massachusetts.

4.2 Restricted Area and Exclusion Area

The reactor room shall constitute a restricted area as defined in 10 CFR 20 and shall be controlled by partitions and normally locked doors. In addition, two small areas, one each on the second and third floors of Washburn Laboratories, directly above the reactor control drives, shall become restricted areas whenever the reactor is operating at power levels in excess in 1 kW and radiation levels in any of the rooms exceed 2 mrem/hr. The exclusion areas, as defined in 10 CFR 100, shall consist of the reactor room and the areas above the reactor.

4.3 Reactor Building and Ventilation System

The reactor shall be housed in a closed room that is designed to restrict leakage. The ventilation system shall provide at least two changes of air per hour in the reactor room whenever the reactor is operating.

4.4 Reactor Core

Fuel Elements: Standard fuel elements shall be flat plate type consisting of uranium-aluminum alloy clad with aluminum. The width and depth of each fuel element shall be 3 in. x 3 in. Each element shall have an active length of 24 in. There shall be a maximum of 10 g of U-235 in each fuel plate and not more than 170 g of U-235 in any fuel element. Standard fuel elements have 18 fuel plates, each plate 1.52 mm thick with a clad thickness of 0.381 mm on each side. A maximum of 28 standard fuel elements may be installed in the core.

Not more than two experimental fuel elements with sixteen removable fuel plates similar to standard fuel plates and fitted with removable top end boxes may be installed in the core. These elements may be used as part of the core assembly either as complete elements or as partial assemblies, loaded with from 2 to a total of 18 fuel plates each.

4.5 Reactor Safety and Control Systems

The safety system shall be designed so that no single electrical fault that partially or completely disables the automatic scram function can, in any manner, impair or disable the manual scram function, and vice versa. The safety system shall be fail safe with respect to loss of voltage.

4.5.1 Nuclear Instrumentation

The channels of nucler instrumentation (listed below with their minimum operating ranges) shall during all reactor critical operations be operational and shall be connected to the safety system, except as noted in Table 4.1.

SECTION 2

SUMMARY

2.1 Introduction

The Worcester Polytechnic Institute Pool Training Reactor (Figure 1) is a 10 kw (heat) light water cooled and moderated reactor designed and built by the General Electric Company as a modification of their standard open pool reactor. This type of pool reactor is similar to the Bulk Shielding Reactor at ORNL and the Geneva Aquarium except it is operated at low power and has additional safeguard features. Primary requirements for safe and flexible operation of the reactor as a student training aid have been met by the use of open pool design, low excess reactivity, and large negative temperature and void coefficients.

The reactor is located in an existing building on the Worcester Polytechnic Institute campus about one mile from the center of Worcester, Mass. The primary use of the reactor is as an integral part of the Worcester Polytechnic Institute elective program in Nuclear Science and Nuclear Engineering. The reactor provides undergraduate and graduate students, under close supervision of qualified personnel, with reactor operating experience and experimental experience in the fields of reactor engineering, metallurgy, chemistry and physics. A list of representative experiments is included in Appendix C.

2.2 General Description of Reactor Facility

The Institute Training Reactor, Figure 1, is a light water cooled and moderated heterogeneous reactor, fueled with approximately 4.0 kg of 20% enriched U-235. The core is located in the center of a pool of demineralized water 8 feet square by 15 feet deep. There are 10 feet of water above the core. Experimental facilities converge toward the core and afford opportunity for performing a number of different experiments at the same time.

A typical core configuration is based on 24 fuel elements in rectangular array as shown in Figure 13. A 1 curie Pu-Be source occupies one module adjacent to the active core. Three safety blades and a manually actuated regulating blade control the reactivity. The blades move vertically in two shrouds, extending the length of the core.

The core is moderated, reflected, and cooled by light water which is circulated by natural convection. The thermal column side of the core is also reflected by graphite. Core elements are contained in a grid box enclosed on four sides to confine the flow of cooling water to the channels between and surrounding the elements. The grid box and contents, as well as the blade drive mechanisms, are supported by a suspension frame from a reactor bridge. The cold, clean core with control blades removed has less than half percent excess reactivity. The safety blades, because of their location and large surface area, have a total shutdown worth of approximately 12% Δk eff.

2.3 Summary of Calculated Reactor Data (Furnished by Argonne National Laboratory (1)

Tabulated below are the significant design parameters which are used in this reactor.

Reactor Materials:

Fuel	Uranium aluminum alloy, 19.75% enriched
Moderator	High purity light water
Reflector	High purity light water and graphite
Coolant	High purity light water
Control	Boral and stainless steel
Structural material	Aluminum
Shield	Water and aluminum lined concrete

Structural Dimensions:

Pool	8 x 8 by 15 ft. deep
Core (active portion)	15 x 15 by 24 inches high
Grid box	9 x 6 array of 3 inch modules
Beam port	One, 6 inch diameter
Thermal column	One, 40 x 40 inches in cross-section

Strategic Materials:

Fissionable material	4.2 Kg U-235
Burn-up	Approximately 1% U-235
Fuel life	Limited by factors other than burn-up

Thermal Characteristics: (Calculated)

Operating power	10 kw (maximum)
Temperature, water	130 deg. F (maximum)
Power peaking factor	2.17
Maximum hot channel factor	1.51
Maximum heat flux	400 Btu/hr.-sq. ft.
Specific power (clean, cold)	2.5 watt/gm U-235
Maximum gamma heat in core	11 watt/liter

Nuclear Characteristics: (Calculated)

Average thermal flux	7.2×10^{10} nv
Average fast flux	25×10^{10} nv
Maximum operating excess reactivity	0.5% Δkeff.
Critical mass	4.0
Temperature coefficient	-0.0163% Δkeff. per degree C
Void coefficient	-.25% Δkeff. per 1% void
Prompt neutron lifetime	61.2 μseconds

Control:

Safety Elements

Number	3 vertical blades
Dimensions	10.5 inches wide by 40.5 inches long by 0.375 inches thick
Material	Boral
Reactivity control	3.5% each, minimum
Total worth, 3 blades	12 Δk_{eff}
Maximum withdrawal rate	7.5 inch/min., one blade at a time

Regulating Element

Number	1 vertical blade
Dimensions	10.65 inches wide by 40.5 inches long by 0.125 inches thick
Material	Stainless steel
Reactivity control	0.7% $\Delta k_{eff}/k$
Maximum withdrawal rate	3.8 inch/min.

Standard Fuel

Type	Flat plate
Number of elements	24 for minimum critical loading
Fuel alloy	Uranium-alumnimum
Clad material	Aluminum
Fuel enrichment	U-235 19.75% enriched
Number of plates per element	18
Plate thickness	1.52 mm (0.060 in.)
Clad thickness	0.381 mm (0.015 in.)

Cooling System:

Coolant	Pool water
Type cooling	Natural convection
Temperature	130 deg.F (maximum)
Purification	Recirculating demineralizer
Purity required	1 ppm - 5 x 10^5 Ohm-cm

2.4 Experimental Facilities

The experimental facilities provided with the reactor are a thermal column, a 6 inch beam tube and 2 fuel elements comparable to above elements but with 16 removable plates.

REFERENCES

1. J.E. Matos and K.E. Freese, "Analyses for Conversion of the Worcester Polytechnic Institute Reactor from HEU to LEU Fuel," RERTR Program, Argonne National Laboratory, Argonne, IL, August 1987.

SECTION 4

REACTOR FACILITY DESCRIPTION

4.1 Reactor Description

4.1.0 General

The reactor assembly comprises the core, control system, instrumentation system and supporting structure. A description of the major core components is presented in the following sections.

4.1.1 Fuel Elements

Each fuel element consists of equi-spaced flat uranium-aluminum alloy fuel plates held vertically between two aluminum side plates. Fuel plates are of the sandwich construction similar to those of the ETR and the General Electric 3 MW Open Pool Reactor. Fuel meat is a uranium-aluminum alloy, 20% enriched. It is clad in aluminum by the picture frame technique. The active length of the fuel plates is 24 inches and the overall dimensions of each element including end boxes are 3 inches square by nearly [redacted] inches long. The end boxes position the fuel elements in the grid and provide handles for fuel positioning. The elements may be inverted or rotated 180 degrees to achieve more efficient utilization of fuel. There are twenty five (25) elements, each of which has 18 plates 25 inches long by 2.79 inches wide. The plates are each 1.52 mm (0.060 in) thick including 0.381 mm (0.015 in) of aluminum cladding on each side. A space of 3 mm (0.12 in) between fuel plates provides a passage for the flow of cooling water by natural convection. Each element contains less than 170 grams of ^{235}U per element. Fixed plate elements are designated F1 through F26 and two removable plate elements, similar in dimensions and loading to the fixed plate elements, are designated R27 and R28.

4.1.2 Control Elements

Reactor control for startup and shutdown is accomplished by three blade-type control elements (Figure 15) with a total shutdown worth of approximately 12% Δk_{eff}. The poison section, of boron carbide and aluminum, approximately 0.380 inches thick, is sandwiched between aluminum side plates. It is 40.5 inches long, 25 inches providing active control of the core. The remaining 15.5 inches connect the poison section to the drive tube.

Each safety blade is guided throughout its travel by a shroud shown in Figure 14. The shroud consists of two thin aluminum plates 38 inches high, separated by aluminum spacers to provide a 1/8-inch water annulus around the blade. The shroud is latched to the sides of the grid box, and can be removed, if necessary, by use of the grapple hook. Small flow holes at the bottom of the shroud minimize the effect of viscous damping on the scram line.

SECTION 5

REACTOR OPERATING CHARACTERISTICS

5.1 Introduction

This section summarizes reactor operating characteristics as calculated by the General Electric Company in support of the initial reactor designs and by Argonne National Laboratory in support of the fuel change to low enriched uranium (LEU) (1). The G.E. calculations were "checked, where possible against data from the Bulk Shielding Facility, the Geneva Swimming Pool Reactor, and critical experiments at ORNL" and were confirmed by "low-power tests and actual operation of the Spanish Pool Reactor," modified where necessary to reflect the increase from the original 1 kw to 10 kw in 1967.

5.2 Nuclear Characteristics

5.2.1 Critical Mass and Loading

The critical mass obtained from ANL diffusion calculations is 4.0 kg of U-235 in 20% enriched fuel for the 5-by-5 element core surrounded by water. The initial core loading consisted of 3.50 kg of U-235 in highly enriched fuel with an excess reactivity of 0.23% $\Delta k/k$.

5.2.2 Neutron and Gamma Flux

Average neutron fluxes for the reactor core at 10 kw are as follows:

thermal (<.625 ev): 8.3×10^{10} nv

epi-thermal ($\geq$.625 ev): 5.5×10^{10} nv

fast (>5.53 Kev): 11.3×10^{10} nv

The distribution of the midplane fluxes in the core is shown in figure 6.

The gamma heating in the core has a peak value of about 10 watt/liter. Figure 7 shows the distribution of the midplane thermal, epithermal, and fast fluxes in the thermal column. Dose rates at 10 kw in the thermal column are as follows:

gamma dose at shield face: 3 rem/hr

gamma dose at outside of thermal column door: <10 mrem/hr

maximum neutron dose at outside of thermal column door from streaming: <10 mrem/hr

5.2.3 Control System

The three safety blades, Figure 13, cut the core into three sections. The control blade with maximum reactivity is blade no. 3. When this blade

is fully withdrawn, control blade no. 1 and 2 have a combined reactivity worth of 6.5% ΔK/K. The control blades span almost the full width of the core. As a result, they keep the reactor subcritical even if a loading error were made and more fuel included than planned. To go critical with the entire 6 x 9 grid filled with fuel elements would require at least 300 gm of U-235 per element, about twice the loading of the fuel elements intended to be used.

5.2.4 Burn-up

The burn-up is estimated to be approximately 1% U-235. The fuel life will be limited by factors other than burn-up.

5.2.5 Temperature Void Doppler and Void Coefficient

Negative temperature and void coefficients aid stable operation of the reactor. The coefficients have been computed separately as functions of temperature or void fraction for each of three physical effects: (1) hardening of the neutron spectrum due to an increase in the water temperature only, (2) the increase in neutron leakage due a decrease in the water density only, and (3) the increase in absorption of the ^{238}U epithermal resonances due to an increase of the temperature of the fuel meat only.

Slope of the core average reactivity feedback coefficients between 25°C and 30°C, along with the whole core void coefficient for a 1% change in water density only are compared with those of the WPI core with 93% enriched fuel below:

Core Average Temperature, Void, and Doppler
Coefficients With All Blades Withdrawn

Effect		HEU Core	LEU Core
Water Temperature Only		-1.7	-1.6
Water Density Only,		-0.5	-0.6
$\Delta k/k \times 10^{-4}$ per °C	SUM	-2.2	-2.2
Fuel Temperature (Doppler) Only, $\Delta k/k \times 10^{-5}$ per °C		0.0	-1.8
Void Coefficient (0-1% Void), $\Delta k/k \times 10^{-3}$ per % void		-2.0	-2.4

The effect of water temperature change on reactivity is shown in figure 8. With the water density held constant, the temperature at which the maximum loaded excess reactivity would be offset is about 50°C. Figure 9 shows the effect of water density change on reactivity of the core. In an excursion, the water outside the core and in any control blade gaps would not be heated to the same temperature and density as the water flowing through the fuel elements. Therefore, the values of the reactivity coefficients shown in Section 2.3 are reasonably conservative.

The effect of fuel temperature change on reactivity due to Doppler broadening is shown in figure 10. Reactivity effects of fuel temperature, water density, and water temperature are all additive so that any credible postulated excursion would be easily offset before temperatures could approach that of fuel damage.

5.2.6 Neutron Lifetime

The neutron lifetime has been calculated as 6.1×10^{-5} seconds. This value is comparable to the measured value of 6.5×10^{-5} on the Borax I reactor.

5.2.7 Alteration of Core Geometry

Alterations of core geometry will affect the available reactivity. If the reactor is loaded to criticality with a square array of 25 fuel elements, arranged symmetricaly with respect to the control blades (figure 13), the changes in reactivity are estimated to be as follows:

1. An extra fuel element is added	plus 1% Δk_{eff}
2. A center fuel element is removed	minus 4.6% Δk_{eff}
3. The 5 elements in Row B are moved to positions (8, D8, E8, C2, and E2)	minus 0.8% Δk_{eff}
4. The 4 elements in row F are moved to positions A4 - A7	-1.0% Δk_{eff}

Adding an extra fuel element will not make the reactor critical provided that at least one safety blade is in the core.

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1. J.E. Matos and K.E. Freese, "Analyses for Conversion of the Worcester Polytechnic Institute Reactor from HEU to LEU Fuel," RERTR Program, Argonne Nat. Lab., Argonne, IL, August 1987.

7.3 Accidents of Operating Type

7.3.1 Startup Accident

Analysis indicates that the fuel elements will not melt if a startup accident should occur. A pessimistic estimate of the energy released in a startup accident has been made under the assumption the log N-period channel fails to operate, and that scram does not take place until the power reached 1-1/2 times normal operating level. The maximum withdrawal speed of the safety blades is 7.5 in/min. The time delay from generation of a scram signal to the instant when the safety blades are free to drop is less than 100 milliseconds. During scram, the blades are assumed to accelerate (for purposes of the accident evaluation) at rates corresponding to the 1-second drop curve in Figure 11. The energy release is about 40 KJ, giving a peak temperature rise of less than 1 degree C. Fuel heat capacity is 2.35×10^6 J/m^{3}K.

The calculation assumes the neutron source to be in place and uses a conservative ratio of trip power to source power, equal to 10^{12}. If the source is absent from the core, this ratio may be of the order of 10^{16} for a clean core. The "source power" then corresponds to the spontaneous fission rate of U^{235} (0.3 fission/sec/kg[1]), and a multiplication factor of 6, due to 17% shutdown reactivity. Startup under this condition is normally prevented by an interlock which requires the B10 startup counter to read at least 5 counts/sec (Section 4.3.10). If, however, the interlock were inoperative, and the blades were withdrawn, the final period would be only about 10% shorter[2] than in the case considered above, with unimportant effect upon the fuel temperature rise.

7.3.2 Refueling Accident

Erroneous loading of a fuel element should be easy to avoid because of the small core size, good visibility of the core, and easily recognizable reference points. All fuel loading will be under the supervision of a senior licensed operator and will be in accord with the provisions of Section 6.

It is not normally possible to go critical with the safety blades in the core. In the event a loading error had taken place, the reactor could conceivably go critical with one safety blade partially in the core. Sine the position of the blades is indicated on the control console, the operator would know that an error had been made and could shut down the reactor before going to power.

7.3.3 Mishandling the Demineralizer Resin

Following an accident such as cladding failure of a fuel element, the resin would become radioactive and the dose rate at the demineralizer may temporarily rise as high as 10 rem/hr. To avoid such peaks of radioactivity, replacement of the resin, would be scheduled when the dose rate had decayed to a low value. In normal facility operation it is expected that the resin in the demineralizer would be transferred rarely, only two resin changes have been required during the past nineteen years.

With a compartment volume of over 30,000 cubic feet, the concentration of fission products inside would create an exposure rate of the order of 40 rem/hr, giving personnel adequate time to take measures for the protection of the facility and for their own safety. The dose rate outside the building does not constitute a significant hazard to the public. These figures are very conservative in that in actuality the reactor is at power less than 10% of the total hours in a year.

Loss of Coolant with Damaged Fuel

Suppose that a leak develops and the pool drains at a moderate rate, similar to the accident reviewed in Section 7.2.4, but made more serious by the fact that the cladding on one or more fuel elements is already damaged. As before, assume that the reactor had been running at full power up to the beginning of the accident, and that no emergency measures are taken to prevent pool drainage or to cool the core.

To simplify a probably much more complex situation, suppose that a single plate of one element is completely stripped of cladding, and that the fission products released are those which vaporize below 1000 degrees F., i.e., xenon, krypton, bromine, and iodine. Assume that the gaseous fission products are uniformly distributed throughout the core fuel and that all the gas atoms within recoil range (5 microns) of the surface are emitted.[1]

The release of fission products is calculated as follows:

Volume fraction released to building:

$$\frac{1 \text{ element}}{25 \text{ elements}} \times \frac{1 \text{ plate}}{18 \text{ plates}} \times$$

$$\frac{2 \times 5 \text{ microns}}{30 \text{ mils}} \times \frac{1 \text{ mil}}{25.4 \text{ microns}} = 2.9 \times 10^{-5}$$

Total energy release rate by fission products released to building:

9.42 E 9 MeV/sec from γs

3.91 E 9 MeV/sec from βs

With a building volume of over 30,000 cubic feet, the concentration of fission products inside would create an exposure rate of the order of 0.2 rem/hour, with no significant hazard to the public.

Spill of a Radioisotope

If a vial or an isotope solution being irradiated should disperse the isotope in the pool water, it would be removed by the pool cleanup demineralizer. The worse condition of this type is considered to be the spilling of a vial containing $AuCl_3$ in solution after it has been irradiated for five days in a flux of 10^{11} nv. It is calculated that a 10 gram sample dispersed in the pool water would produce 140 mrem/hr at the pool surface and area monitoring equipment will sound an alarm. The room

WPI LEU CORE

MIDPLANE FLUXES FROM CORE CENTER

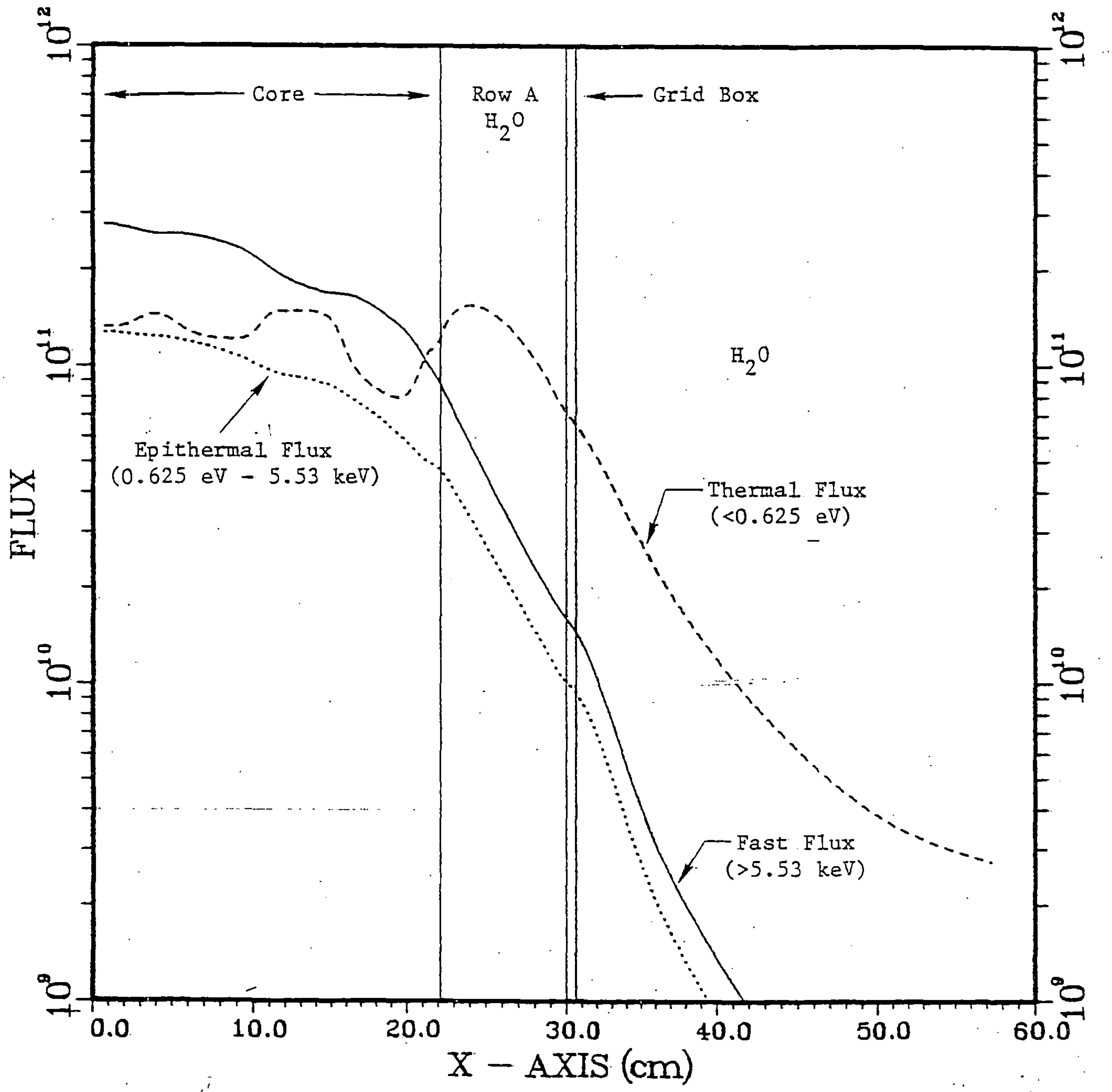


Figure 6

WPI LEU CORE, MIDPLANE THERMAL COLUMN FLUXES

FROM CORE FACE OF THERMAL COLUMN

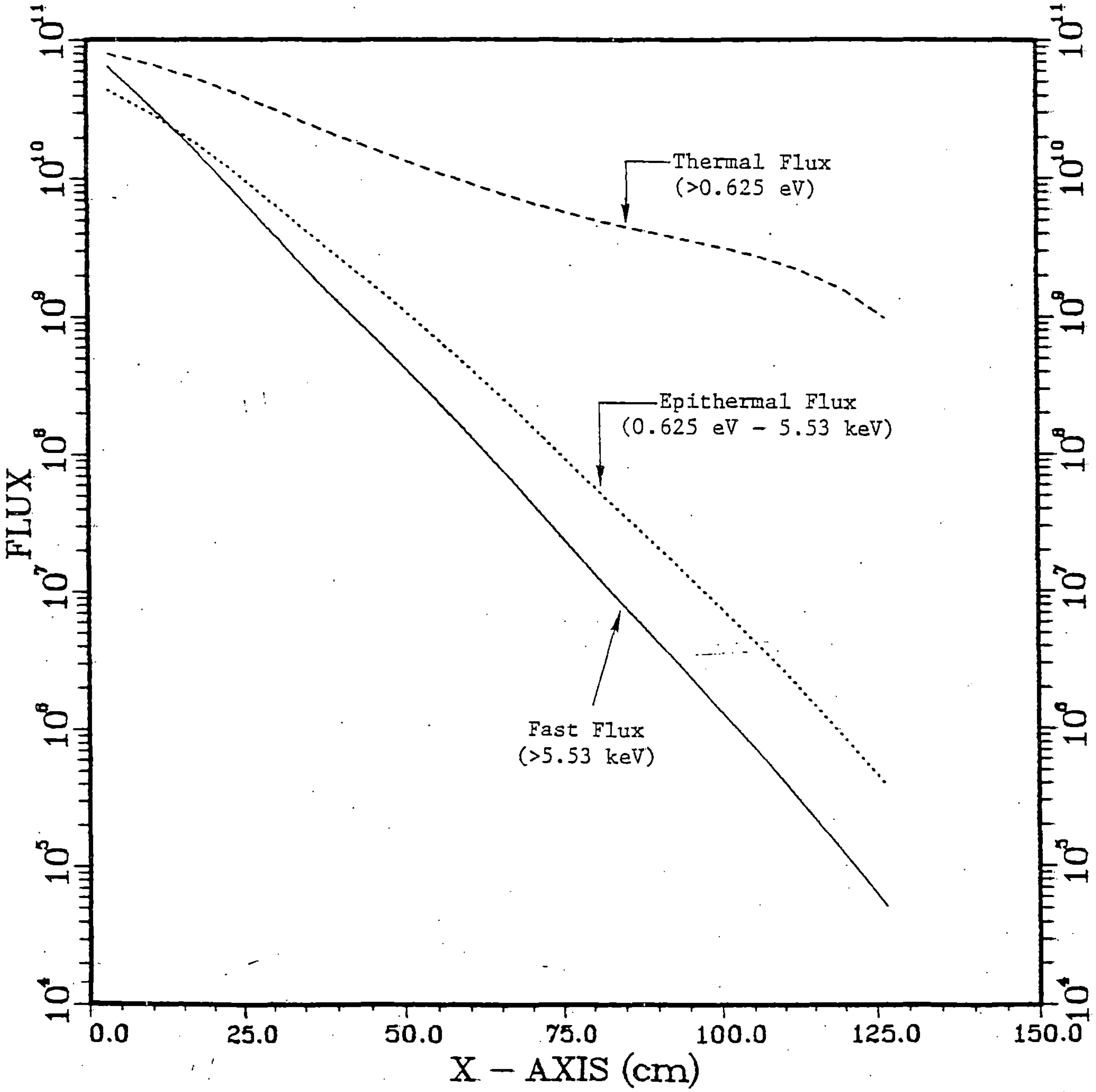
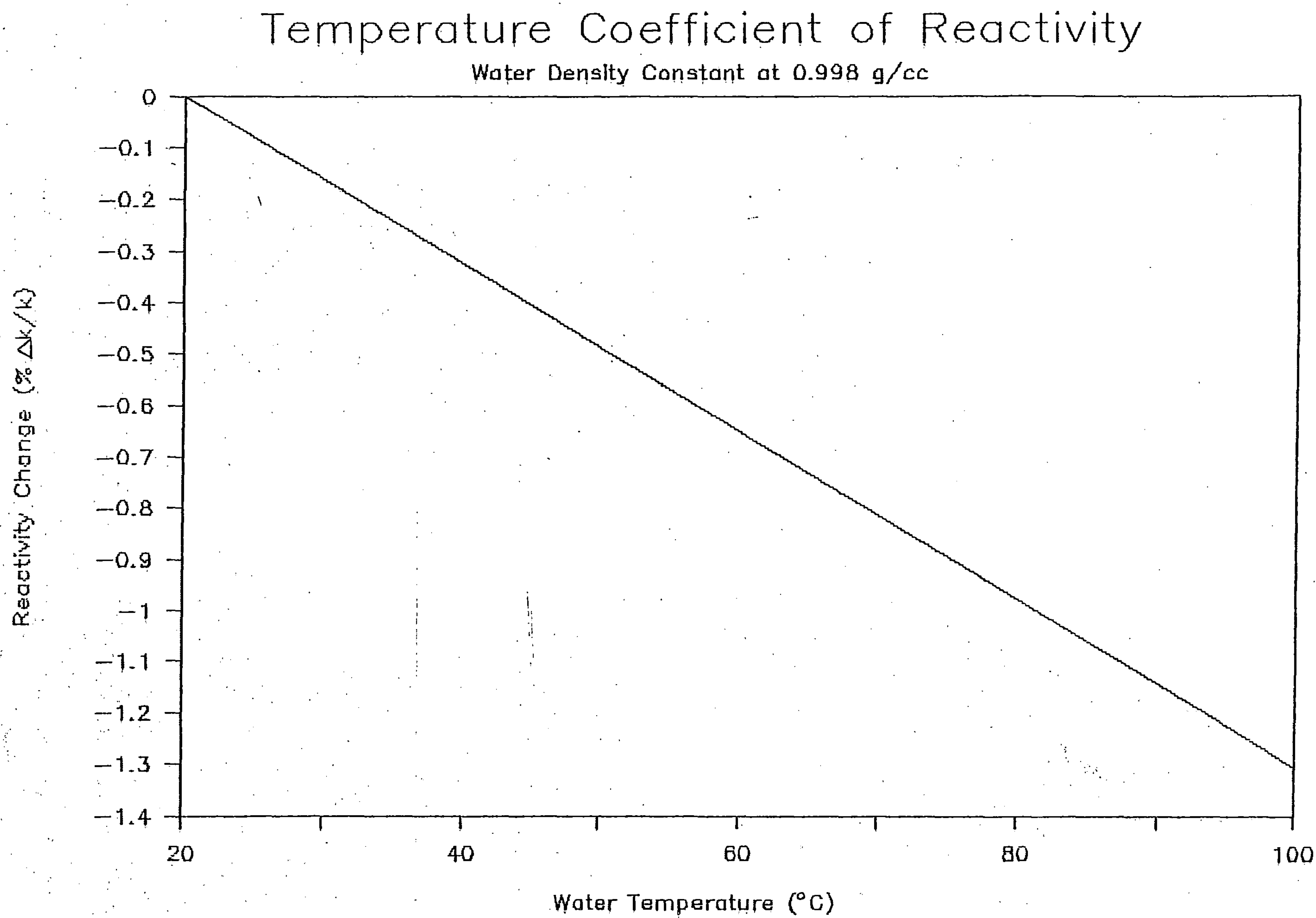


Figure 7



Temperature Coefficient of Reactivity
Water Density Constant at 0.998 g/cc
Reactivity Change (% Δk/k)
0
−0.1
−0.2
−0.3
−0.4
−0.5
−0.6
−0.7
−0.8
−0.9
−1
−1.1
−1.2
−1.3
−1.4
20
40
60
80
100
Water Temperature (°C)

Figure 8

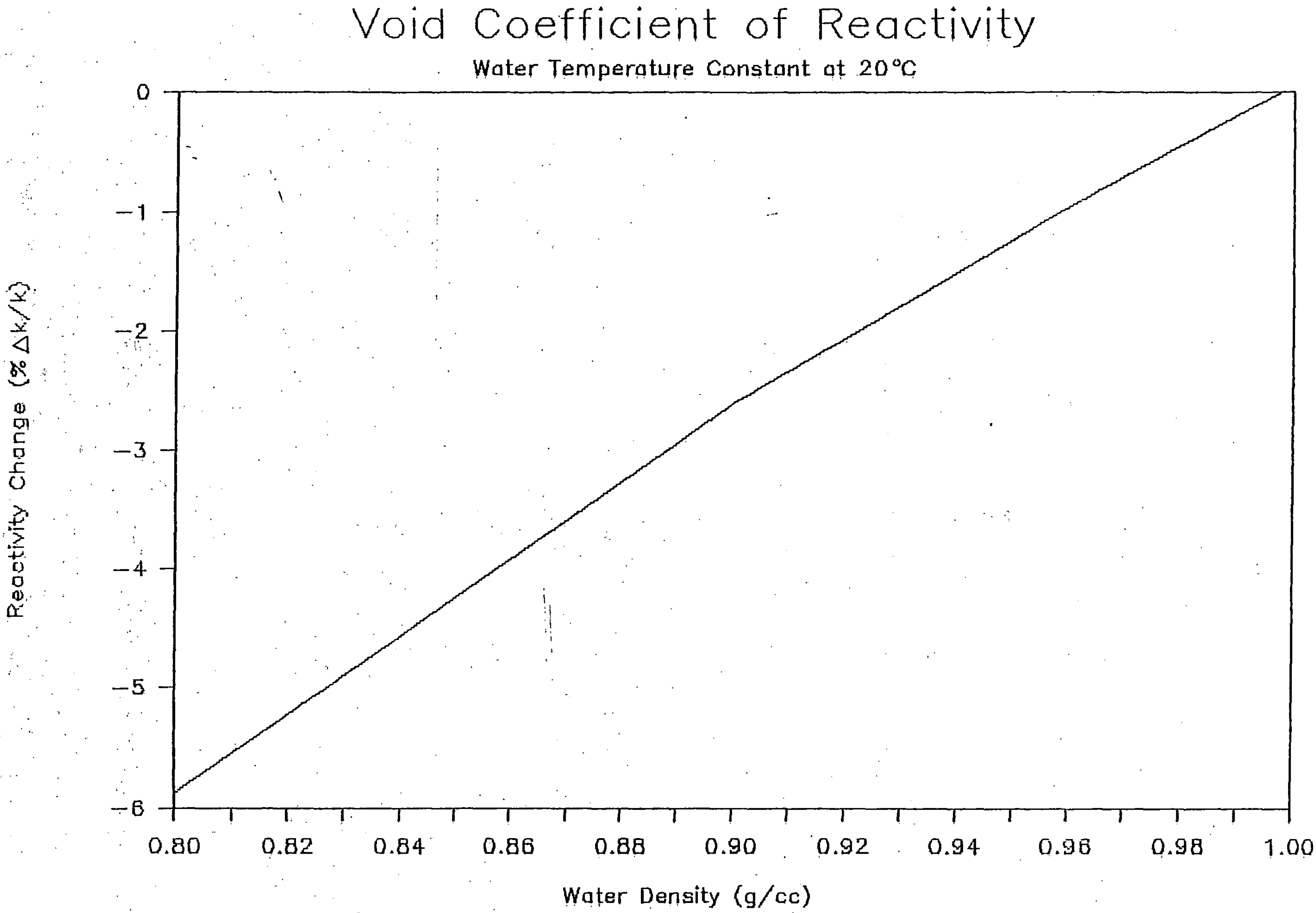


Figure 9

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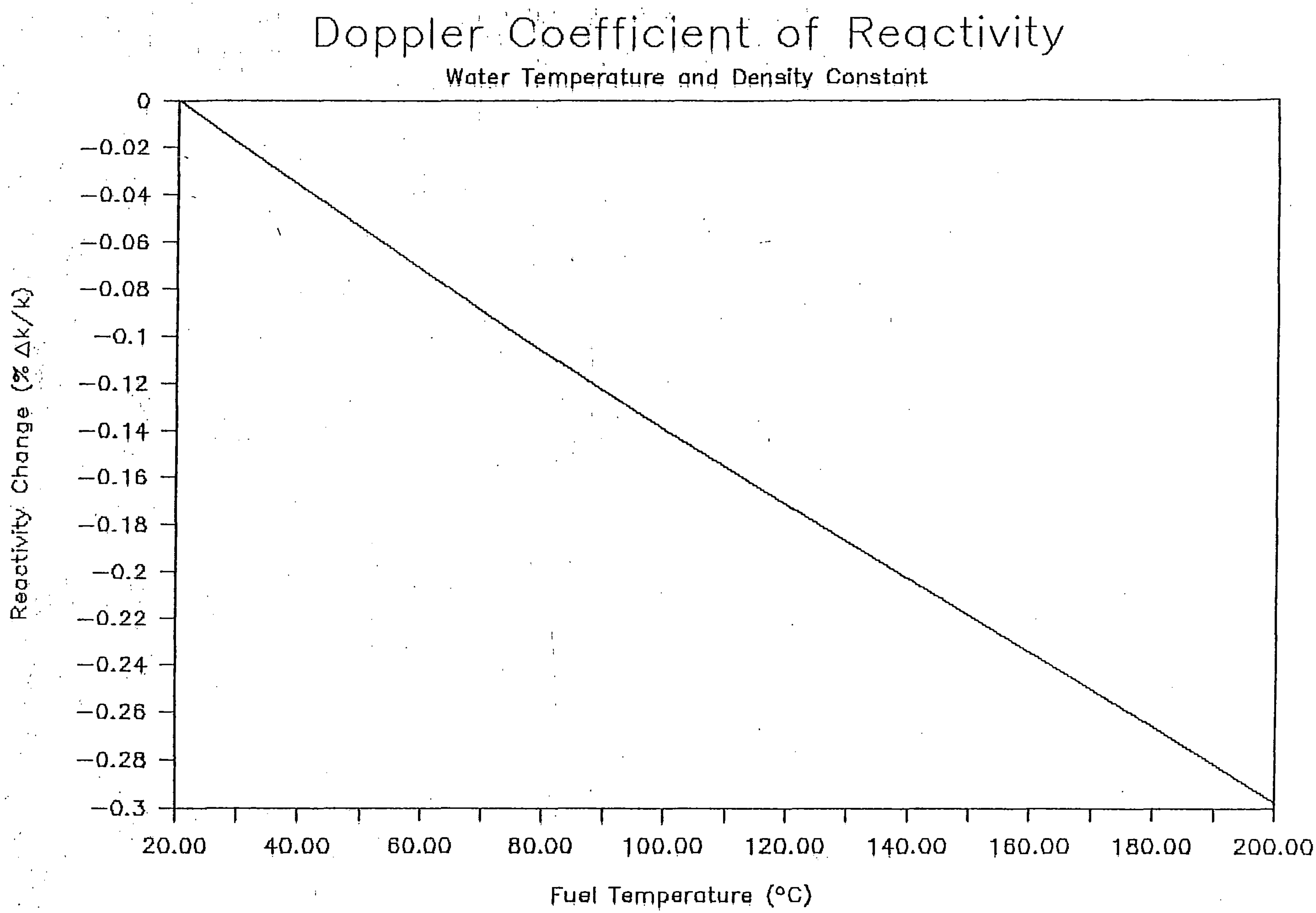
Doppler Coefficient of Reactivity
Water Temperature and Density Constant
0
−0.02
−0.04
−0.06
−0.08
−0.1
−0.12
−0.14
−0.16
−0.18
−0.2
−0.22
−0.24
−0.26
−0.28
−0.3
Reactivity Change (% Δk/k)
20.00
40.00
60.00
80.00
100.00
120.00
140.00
160.00
180.00
200.00
Fuel Temperature (°C)

Figure 10

[redacted]

Figure 22 Fuel Element

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ANALYSES FOR CONVERSION OF THE WORCESTER POLYTECHNIC INSTITUTE REACTOR FROM HEU TO LEU FUEL

J. E. Matos and K. E. Freese
RERTR Program
Argonne National Laboratory
Argonne, IL 60439

August 1987

SUMMARY

This report contains the results of design and safety analyses performed by the RERTR Program at Argonne National Laboratory (ANL) for conversion of the Worcester Polytechnic Institute (WPI) reactor from the use of HEU fuel to the use of LEU fuel. The objectives of this study were to: (1) maintain the present HEU core size, (2) utilize a proven fuel element design that is economical to manufacture, (3) maintain or improve upon the present margins of safety, and (4) maintain as closely as possible the technical specifications and operating procedures of the present HEU core.

The LEU fuel element design chosen to meet these objectives has the same external dimensions and structural components as the present WPI HEU fuel elements, but contains 18 fuel plates with fuel meat that is identical with the LEU UAl_x fuel plates that have been used successfully in the Ford Nuclear Reactor at the University of Michigan since December 1981.

Documents for the WPI reactor that were reviewed by ANL as bases for the design and safety evaluations were the WPI Safety Analysis Report, the WPI Technical Specifications, and the NRC Safety Evaluation Report related to the renewal of the WPI operating license in December 1982.

The methods and codes that were utilized by ANL have been qualified using comparisons of calculations and measurements in the LEU demonstration cores in the Ford Nuclear Reactor at the University of Michigan and in the Oak Ridge Research Reactor at the Oak Ridge National Laboratory.

Only those reactor parameters and safety analyses which could change as a result of replacing the HEU fuel in the core with LEU fuel are addressed in this report. The attached summary table provides a comparison of the key design features of the HEU and LEU fuel elements and a comparison of the key reactor parameters that were calculated for each core. The results show that all of the objectives of this study are fully realized and that the WPI reactor facility can be operated as safely with the new LEU fuel as with the present HEU fuel.

SUMMARY TABLE

Calculated Core Physics and Safety Parameters for Conversion of the Worcester Polytechnic Institute Reactor from HEU to LEU Fuel

Fuel Design Data	HEU Core	LEU Core
Type Fuel	UAl Alloy	UAl_x-Al
Enrichment, %	93.4	19.75
Number of Fuel Plates	10	18
Meat Thickness, mm	0.991	0.762
Cladding Material	1100 Al	6061 Al
Cladding Thickness, mm	0.762	0.381
^{235}U per Element, g	136.3	167.0
U Density, g/cm^3	0.39	1.78
Reactor Parameters		
Excess Reactivity, % $\Delta k/k$ all blades out	0.29	0.01[a] 0.37[b]
Worth of Regulating Blade, % $\Delta k/k$	0.7	0.7
Shutdown Margin with Max. Worth Blade Stuck, % $\Delta k/k$	6.0	6.0
Prompt Neutron Lifetime, μs	69.0	61.2
Delayed Neutron Fraction, %	0.77	0.77
Temperature Coefficient, $\Delta k/k \times 10^{-4}$ per °C	−1.7	−1.6
Void Coefficient, $\Delta k/k \times 10^{-3}$/% Void	−2.0	−2.4
Doppler Coefficient, $\Delta k/k \times 10^{-5}$ per °C	0.0	−1.8
Overpower Factor to Onset of Nucleate Boiling	24.8	44.2
Limiting Step Reactivity to Initiate Clad Melting, %$\Delta k/k$	1.5	2.0

[a]LEU core arrangement identical with HEU core.
[b]Fuel element in position B3 moved to position F3.

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ANALYSES FOR CONVERSION OF THE WORCESTER POLYTECHNIC INSTITUTE REACTOR FROM HEU TO LEU FUEL

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August 1987

1. INTRODUCTION

This report contains the results of design and safety analyses performed by the RERTR Program at Argonne National Laboratory (ANL) for conversion of the Worcester Polytechnic Institute (WPI) reactor from the use of HEU fuel to the use of LEU fuel. The objectives of this study were to: (1) maintain the present HEU core size, (2) utilize a proven fuel element design that is economical to manufacture, (3) maintain or improve upon the present margins of safety, and (4) maintain as closely as possible the technical specifications and operating procedures of the present HEU core.

The LEU fuel element design chosen to meet these objectives has the same external dimensions and structural components as the present WPI HEU fuel elements, but contains 18 fuel plates with fuel meat that is identical with the LEU UAl_x fuel plates that have been used successfully in the Ford Nuclear Reactor (FNR) at the University of Michigan since December 1981. The WPI reactor is already licensed to use HEU fuel elements (from Washington State University) with this geometry. Detailed data on the qualification of the UAl_x fuel being used in the FNR can be found in Ref. 1.

Documents for the WPI reactor that were reviewed by ANL as bases for the design and safety evaluations were the WPI Safety Analysis Report,[2] the WPI Technical Specifications,[3] and the NRC Safety Evaluation Report[4] related to the renewal of the WPI operating license in December 1982.

The methods and codes that were utilized by ANL have been qualified using comparisons of calculations and measurements in the LEU demonstration cores[5-7] in the FNR at the University of Michigan and in the Oak Ridge Research Reactor (ORR) at the Oak Ridge National Laboratory. Additional qualification has been obtained via international benchmark comparisons sponsored by the IAEA.

The design and safety analyses in this report provide comparisons of reactor parameters and safety margins for the WPI HEU and LEU cores. Only those parameters which could change as a result of replacing the HEU fuel in the core with LEU fuel are addressed.

2. REACTOR DESCRIPTION

The WPI reactor is a 10 kW open-pool training reactor designed and built by the General Electric Company in 1959. The normal core configuration (Fig. 1) is based on 24 fuel elements containing approximately 3.3 kg of 93% enriched uranium arranged in a rectangular array inside an aluminum grid box.

The core is moderated, reflected, and cooled by light water that is circulated by natural convection. The thermal column side of the core is also reflected by graphite. Three safety blades and a manually actuated regulating blade control the reactivity. The blades move vertically in two aluminum shrouds extending the length of the core. There is one 6 in. diameter beam port on the side of the core opposite from the thermal column.

3. FUEL ELEMENT DESCRIPTIONS

The geometries, materials and fissile loadings of the current HEU fuel elements and the replacement LEU fuel elements are shown in Table 1. The external dimensions, sideplates, end fittings, and other structural components of both elements are identical, except that the LEU elements utilize 6061 Al instead of 1100 (or 2S) Al.

Table 1. **Descriptions of the Current HEU and Replacement LEU Elements**

	HEU	LEU
Number of Fuel Plates/Element	10	18
Fissile Loading/Plate, g ^{235}U	13.6	9.3
Fissile Loading/Element, g ^{235}U	136.3	167.0
Fuel Meat Composition	U-Al Alloy	UAl_x-Al
Fuel Meat		
Thickness, mm (mil)	0.99 (39)	0.76 (30)
Width , mm (in.)	63.5 (2.5)	54.4-63.5 (2.14-2.50)
Length , mm (in.)	610 (24.0)	572-610 (22.5-24.0)
Cladding Material	1100 (or 2S) Al	6061 Al*
Cladding Thickness, mm (mils)	0.76 (30)	0.38 (15)

*10 ppm natural boron were added to the compositions of the cladding and all fuel element structural materials to represent the alloying materials in 6061 Al. The aluminum in the fuel meat was 1100 Al with no boron impurity.

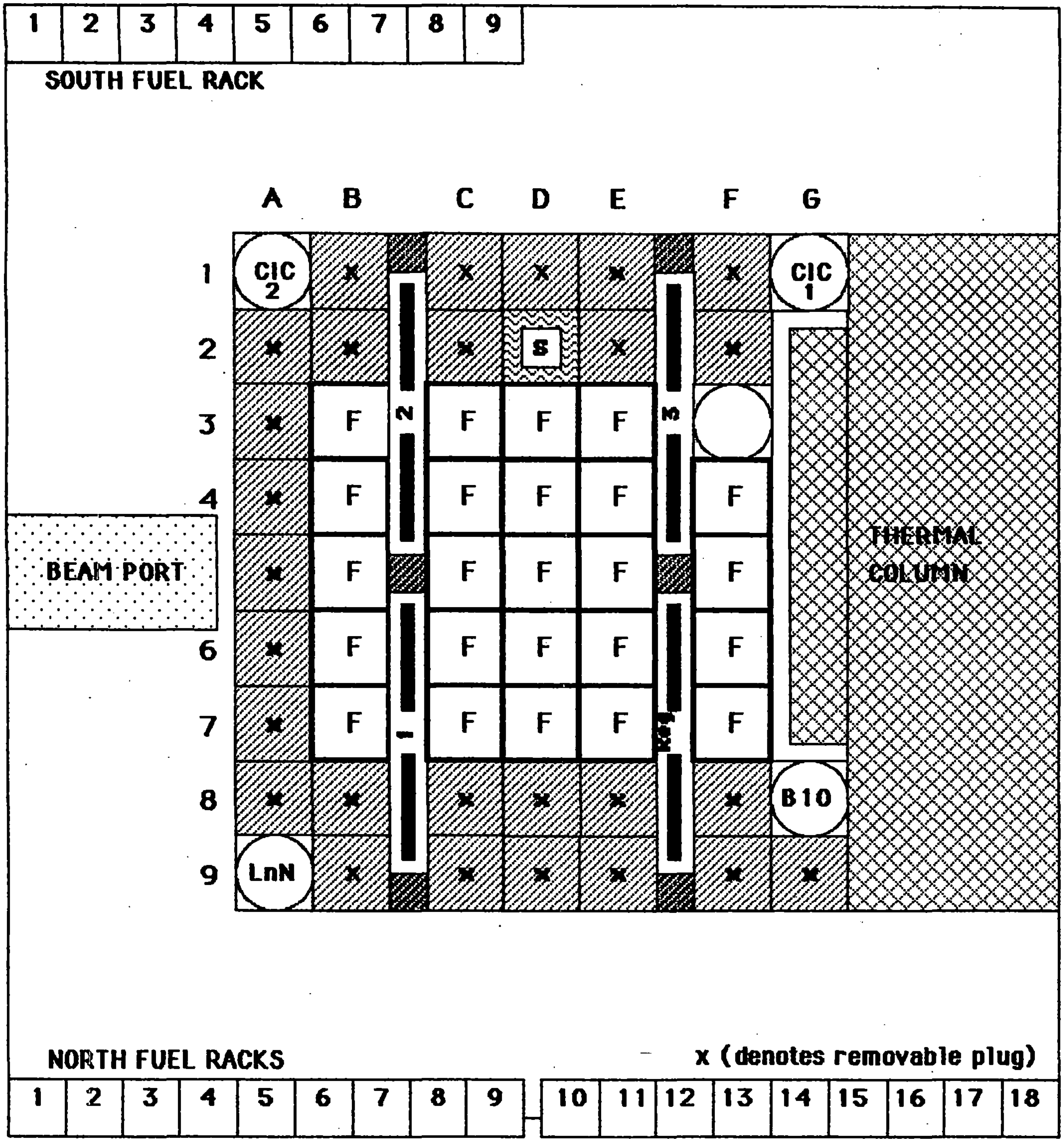


Fig. 1 WORCESTER POLYTECHNIC INSTITUTE REACTOR

Typical Core Arrangement and Location of Fuel Racks

4. CALCULATIONAL MODELS

4.1 Nuclear Cross Section Models

Microscopic cross sections in ten energy groups (Table 2) were prepared at 23°C (73°F) using the EPRI-CELL code[10] for the HEU and LEU fuel element geometries and fissile loadings. The integral transport calculations in EPRI-CELL were performed for 69 fast groups and 35 thermal groups (<1.855 eV), which were then collapsed to ten broad energy groups for use in diffusion theory calculations.

Table 2. **Ten Group Energy Group Boundaries**

Group No.	Upper Energy	Lower Energy	Group No.	Upper Energy	Lower Energy
1	10.0 MeV	0.639 MeV	6	1.166 eV	0.625 eV
2	0.639 MeV	9.119 keV	7	0.625 eV	0.417 eV
3	9.119 keV	5.531 keV	8	0.417 eV	0.146 eV
4	5.531 keV	1.855 eV	9	0.146 eV	0.057 eV
5	1.855 eV	1.166 eV	10	0.057 eV	2.53×10^{-4} eV

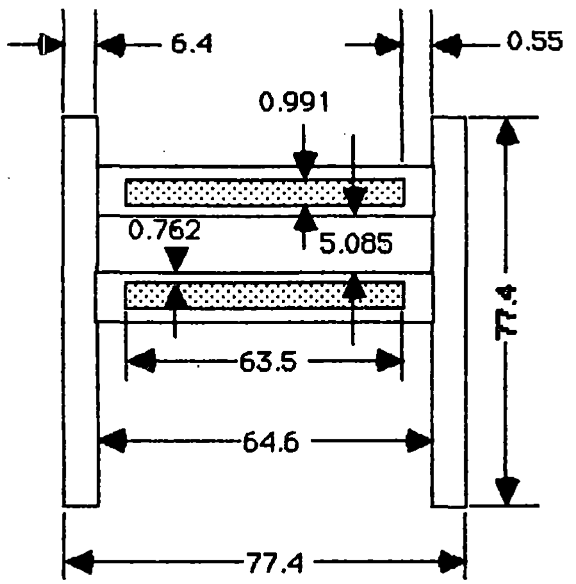
Figure 2 shows the dimensions of the HEU and LEU fuel elements and the fuel element models that were used in the diffusion theory calculations for the reactor. Note that the fueled and non-fueled regions of each element were modeled separately. A non-fueled region consists of a sideplate and the fuel plate aluminum (plus associated water) between the fuel meat and the sideplate.

Figure 3 shows the unit cell geometries that were used in EPRI-CELL to generate microscopic cross sections for the fueled and non-fueled regions of each fuel element. A non-fuel region is represented by the "extra region" containing calculated volume fractions of aluminum and water. The local fine group spectrum over the "extra region" only was used to collapse the fine-group non-fuel cross sections into 10 broad groups. Both cell calculations were done using a fixed buckling of 0.00172 cm^{-2}, which corresponds with the anticipated axial extrapolation length of about 8 cm in each fuel element in the reactor diffusion theory calculations.

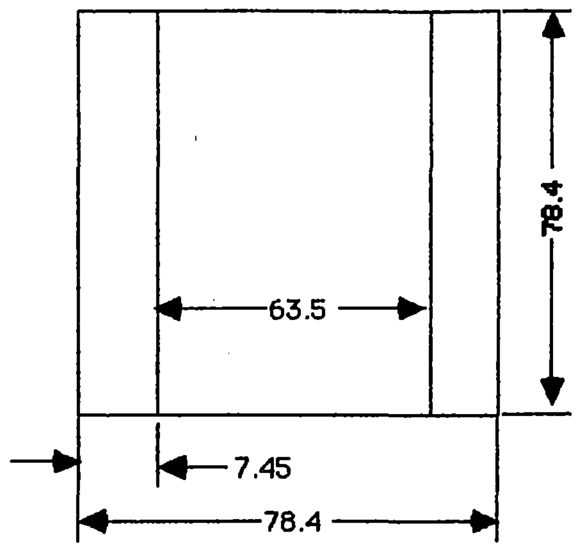
Microscopic cross sections for the grid box were calculated by replacing the "extra region" in Fig. 3 with a 6.35 mm thickness of aluminum and collapsing the cross sections using the local fine group fluxes over the grid box. Cross sections for the water reflector, the in-core flux trap and the fuel element end fittings were calculated using a unit cell model consisting of a pure U-235 fission spectrum on a 10 cm thick slab of water. Cross sections for the graphite thermal column were prepared using a model that consisted of a cylindrical region of graphite 50 mm in diameter surrounded by a homogenized source representing the core. Local fine group fluxes over the graphite only were used in collapsing the graphite cross sections into 10 broad groups.

Fig. 2 Models for WPI HEU and LEU Fuel Elements

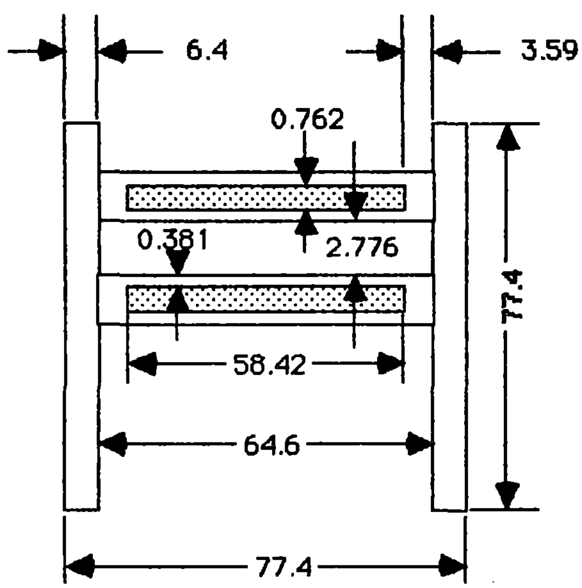
(Dimensions in mm)



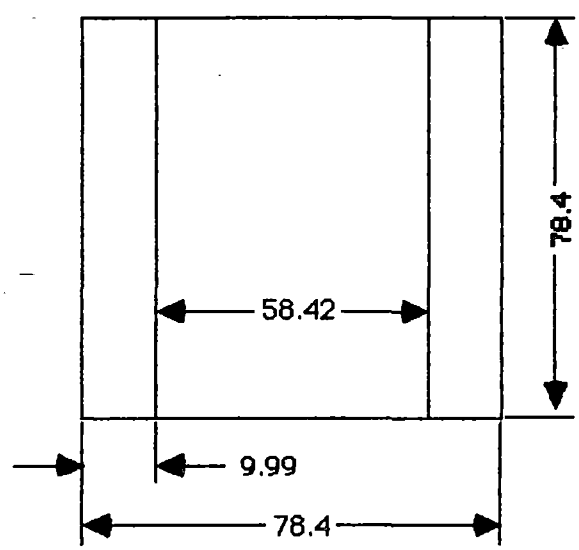
HEU Fuel Element
10 Fuel Plates



DIF3D Model for
HEU Fuel Element



LEU Fuel Element
18 Fuel Plates

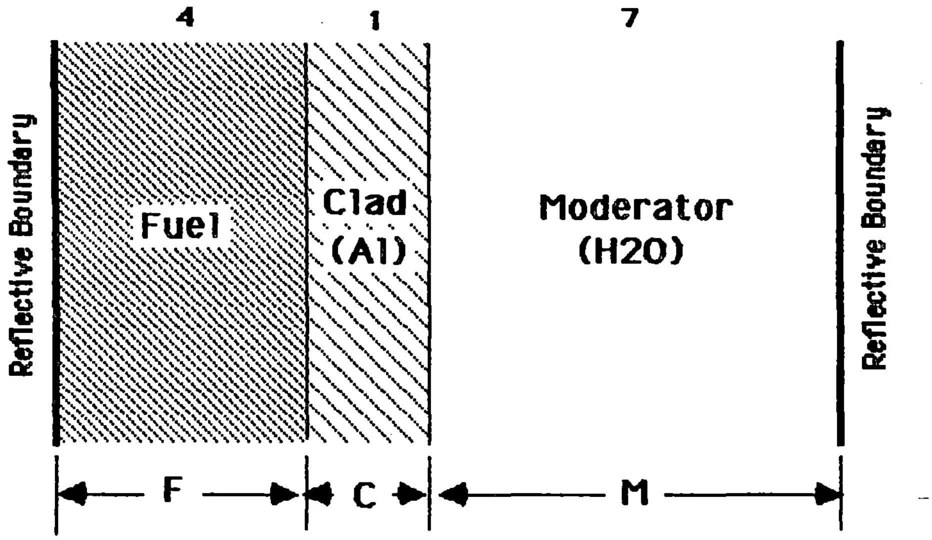


DIF3D Model for
LEU Fuel Element

Fig. 3. EPRI-CELL Model for Generating Fuel Element Cross Sections (Dimensions in mm)

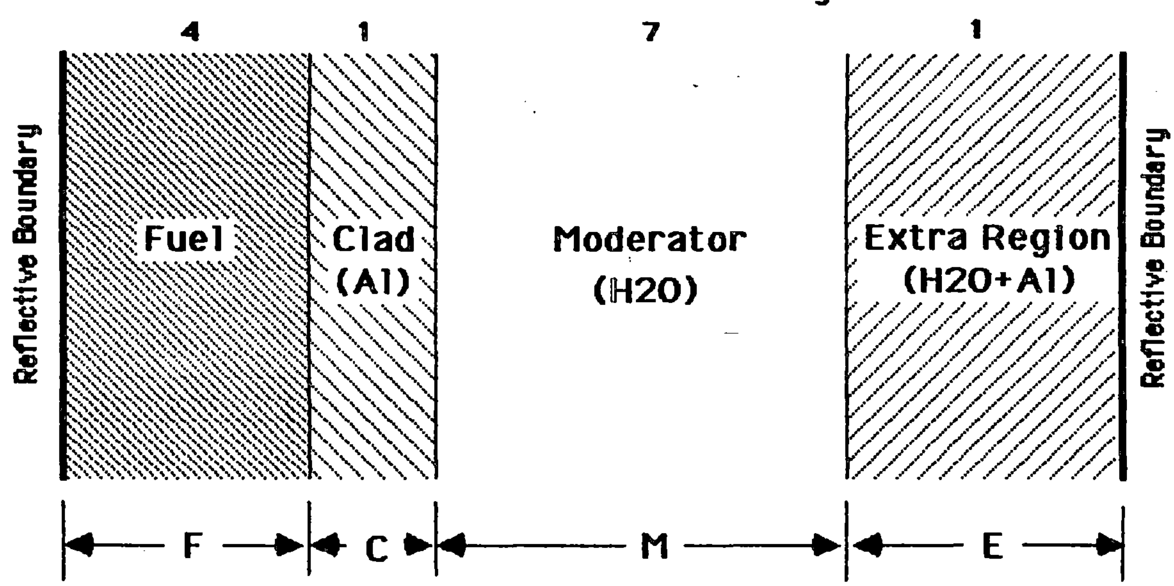
Core	Fuel Plates /Elem.	Half Fuel Region (F)	Half Moderator Region (M)	Clad Region (C)	Half Extra Region (E)	Extra Region Al/H2O Volume Fractions
HEU	10	0.4955	2.5425	0.762	0.6344	0.8268/0.1732
LEU	18	0.3810	1.3880	0.381	0.7725	0.7141/0.2859

Number of Mesh Points in Each Region



Unit-Cell Specifications for Fueled Portion of Fuel Element

Number of Mesh Points in Each Region



Unit-Cell Specifications for Non-Fueled Portion of Fuel Element
(Cross Section Collapse with Fluxes over Extra Region Only)

4.2 Reactor Model

The reactor arrangement shown in Fig. 1 was modeled in detail in three dimensions using the DIF3D diffusion theory code.[11] As mentioned in Section 4.1, the fueled and non-fueled regions of each fuel element were modeled separately. Explicit representations were also included for the grid box, control blades, control blade shrouds, thermal column and its container, and the water gap between the grid box and the thermal column container.

The poison section of the control blades consists of boral (boron carbide and aluminum) with 35 wt% B_4C. The regulating blade is composed of stainless steel.

5. DYNAMIC DESIGN EVALUATIONS

5.1 Cold Clean Excess Reactivities

The WPI Technical Specifications limit the excess reactivity of the cold clean core to a maximum of 0.5% $\Delta k/k$.

The excess reactivity of the cold clean HEU core with 24 fuel elements and the three control blades and regulating blade fully withdrawn was calculated to be 0.29% $\Delta k/k$. This value is in excellent agreement with the measured cold clean excess reactivity of 0.23-0.26% $\Delta k/k$.

The corresponding excess reactivity of the reference LEU core (24 fuel elements with 167 g ^{235}U per element and a water-filled flux trap in grid position F3) was calculated to be 0.01% $\Delta k/k$.

To provide flexibility for fine adjustment of the actual excess reactivity at startup, DOE intends to supply WPI with a special LEU fuel element having 2 fixed fuel plates and 16 removable fuel plates.

Table 3 and Fig. 4 show the calculated excess reactivity of the LEU reference core as a function of the number of LEU fuel plates in the special element if it were located in grid position F3. Inserting the special element with two fixed plates would increase the excess reactivity by about 0.17% $\Delta k/k$. Adding 16 additional plates would increase the excess reactivity by about 0.052% $\Delta k/k$ per plate to a total excess reactivity of 1.01% $\Delta k/k$ in a 25 element core.

Another LEU core configuration that should be considered involves moving the fuel element in grid position B3 into grid position F3, which is adjacent to the graphite thermal column. The excess reactivity in this case was calculated to be 0.37% $\Delta k/k$. The excess reactivity would increase by about 0.07% $\Delta k/k$ if the special element with two fixed plates were inserted into grid position B3 and by another 0.57% $\Delta k/k$ if 16 additional plates were added.

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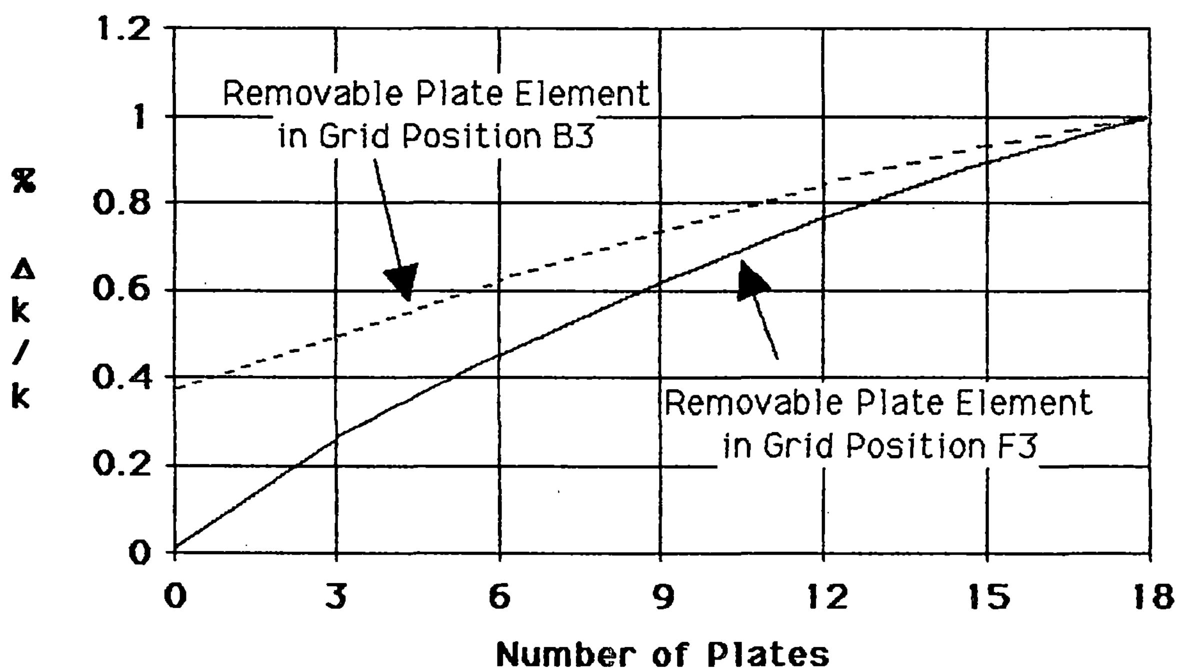
Table 3. **Excess Reactivity of LEU Reference Core as a Function of the Number of Fuel Plates in a Special Element With Removable Plates in Grid Position F3**

Number of Plates in Position F3	g ^{235}U in Position F3	Excess Reactivity, % Δk/k	
0	0.00	0.01	Reference Core
3	27.84	0.26	
6	55.68	0.45	
9	83.52	0.62	
12	111.36	0.77	
15	139.20	0.90	
18	167.00	1.01	25 Element Core

Fig. 4 Excess Reactivity of LEU Core vs Number of Fuel Plates in Grid Position F3 or B3.



5.2 Sensitivity Calculations for the LEU Reference Core

The as-built LEU fuel elements can have ^{235}U loadings in the fuel meat and ppm boron equivalents in the 6061 Al cladding that are different from the nominally specified values.

Fuel plate manufacturing specifications usually allow a fissile loading variation of ±2%. For the WPI LEU elements, this means that ^{235}U loadings between 163.7 and 170.3 g would be acceptable. However, the loading variation in as-built fuel elements is actually less than 1 g. The data in Table 4 show that the sensitivity of the LEU reference core to a loading variation of ±1 gram of ^{235}U per fuel element is about 0.12% $\Delta k/k$.

6061 Al will be used to manufacture the fuel plate cladding and structural materials of the LEU elements. Spectrographic analyses of the alloying materials in recent 6061 Al samples yield about 10 ppm natural boron equivalent for the Fe, Cr, Ni, Cu, Si, Mn, etc. However, the physical natural boron imputity content is usually specified in spectrographic analyses as <10 ppm because an additional (and expensive) chemical analysis needs to be performed to measure boron impurity contents of <10 ppm.

As stated in Table 1, all calculations for the LEU reference core included 10 ppm natural boron in the compositions of the fuel plate cladding and fuel element structural materials to represent the alloying materials in 6061 Al. No natural boron was included to represent the physical boron impurity content since its value was not known. Instead, the calculations shown in Table 4 were performed to determine the sensitivity to several assumed total boron equivalents. The results show a decrease in excess reactivity of about 0.38% $\Delta k/k$ per 10 ppm natural boron in the aluminum structural materials of each fuel element.

When these results are factored in, the most probable final core configuration is likely to have a fuel element in grid position F3 adjacent to the graphite thermal column and perhaps two or more fuel plates in the special element with removable plates in grid position B3.

Table 4. **Sensitivity Calculations for WPI LEU Reference Core**

Grams ^{235}U per Element	ppm Boron Equiv.	Excess React., % $\Delta k/k$	
Sensitivity to g ^{235}U per Element			
166	10	-0.11	
167	10	0.01	Reference Core
168	10	0.14	
Sensitivity to total ppm Nat. Boron Equivalent in 6061 Al			
167	10	0.01	Reference Core
167	20	-0.36	
167	30	-0.74	

5.3 Neutron Fluxes

Fast (>5.53 keV), epithermal and thermal (<0.625 eV) neutron fluxes at the core midplane of the HEU and reference LEU cores with the control blades fully-withdrawn are shown in Fig. 5 for: (1) a traverse from the center of the core through the control blade shrouds in the direction of the beam tube, and (2) a traverse into the thermal column from the thermal column face.

Fluxes in the thermal column of the HEU and LEU fueled cores are practically indistinguishable. In the core, the HEU and LEU thermal flux profiles in the region of the control blade shrouds are slightly different because of the smaller fuel meat width and wider non-fueled regions modeled in the LEU core (see Fig. 2).

Table 5 shows a comparison of the 3D average neutron fluxes in the core (including the control regions) and in the water-filled flux trap in grid position F3. The average thermal flux in the LEU core is reduced by about 14% because of its higher fissile loading. In the flux trap, however, where experiments are normally performed, the thermal flux in the LEU core is lower by only 3%.

Table 5. **Average Fluxes (nv x 10^{10}) in Core and Flux Trap**

	Energy Boundary	5 x 5 Core*		Flux Trap in Grid Position F3	
		HEU	LEU	HEU	LEU
Fast	>5.53 keV	11.1	11.3	3.5	3.5
Epithermal	0.625 eV-5.53 keV	5.4	5.5	2.2	2.2
Thermal	<0.625 eV	9.6	8.3	11.5	11.2

*Including Control Regions

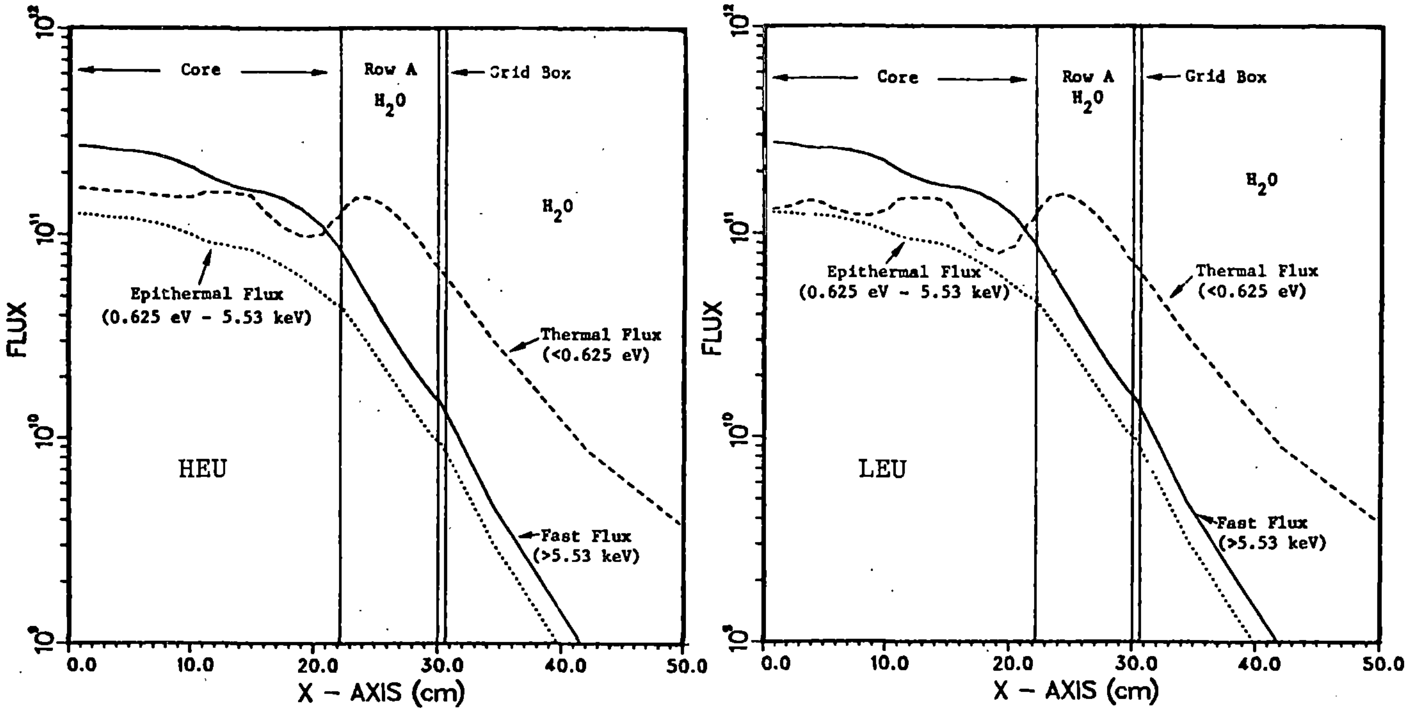
5.4 Power Distributions and Power Peaking Factors

Power distributions and nuclear power peaking factors that were calculated for the HEU and reference LEU cores with all control blades fully-withdrawn are shown in Fig. 6. The power distributions showing the total power (in kW) and the percent of total power in each fuel element are practically identical.

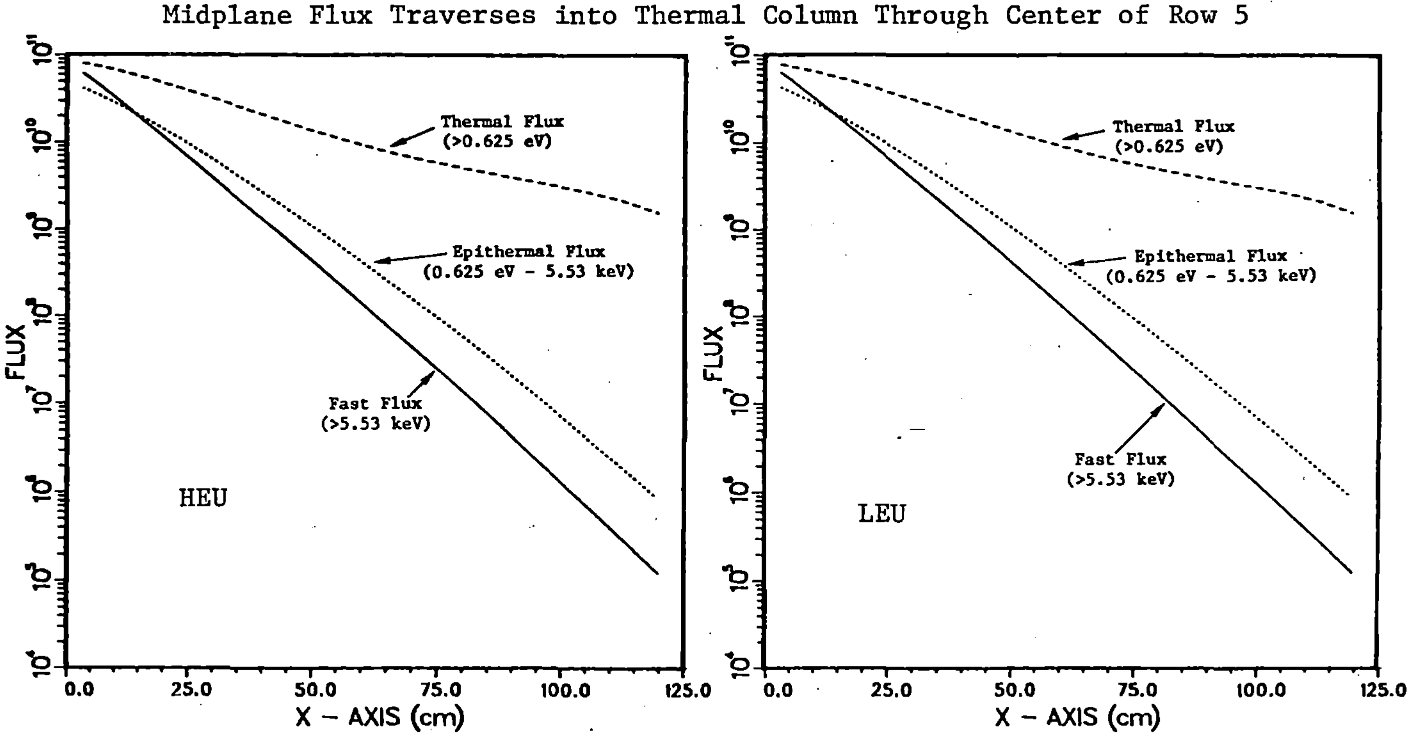
From the point of view of thermal-hydraulic safety margins, the most important parameter is the total 3D power peaking factor (the absolute peak power density in a fuel element divided by the average power density in the core). In these 3D calculations, the total power peaking factor is defined as the product of two components: (1) a radial factor defined as the average power density in each element divided by the average power density in the core, and (2) an element factor defined as the peak power density in each

Fig. 5. Neutron Flux Profiles in the HEU and LEU Cores with All Control Blades Fully-Withdrawn.

Midplane Flux Transfers from Core Center (Position D5) Toward Beam Tube



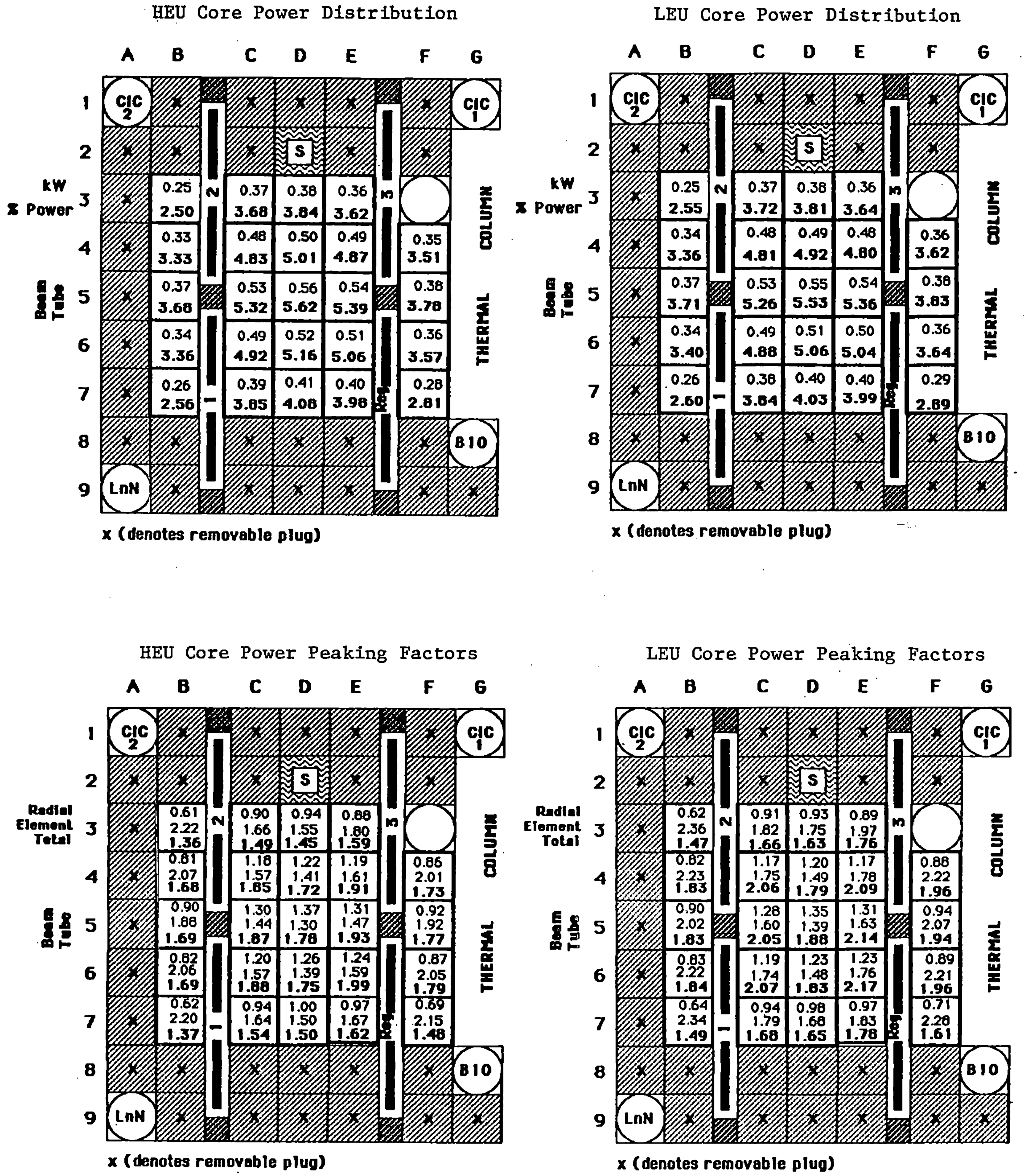
Midplane Flux Traverses into Thermal Column Through Center of Row 5



Distance from Core Face of Thermal Column-cm

Distance from Core Face of Thermal Column-cm

Fig. 6. Power Distributions and Nuclear Power Peaking Factors for HEU and LEU Cores with All Blades Fully-Withdrawn.



element divided by the average power density in that element. The element factor is a pointwise factor that includes both planar and axial power peaking within an element. Note that the element factors shown in Fig. 6 were calculated using power densities computed at the edge of the mesh interval with highest power within each fuel element and that these power densities are nearly independent of the mesh spacing utilized[8] in the computer model.

From the data in Fig. 6, the total power peaking factors in the LEU core range between 4% and 12% higher than in the HEU core. However, the limiting fuel element from a thermal-hydraulic safety margin point of view is located in grid position E6 in both cores and is about 9% higher in the LEU core.

5.5 Criticality Condition and Shutdown Margin

The WPI Technical Specifications have a criticality condition stating that the reactor shall be subcritical when the three control blades are at their fully withdrawn postions and the regulating blade is in its fully inserted positon. In addition, the stainless steel regulating blade shall have a reactivity worth that is less than the effective delayed neutron fraction. The shutdown margin requirement states that the reactor shall be subcritical by at least 1.0% $\Delta k/k$ with the control blade of maximum worth (and the regulating blade) stuck out of the core.

5.5.1 Blade Descriptions and Diffusion Theory Model

The three control blades each consist of vertical aluminum-clad boral blades 10.5 in. wide x 40.5 in. long x 0.375 in. thick. The cladding thickness is 0.060 in. on all sides. The boral poison section is 0.255 in. thick and consists of a mixture of 35 wt% B_4C and 65 wt% 1100 Al. The regulating blade consists of a vertical stainless steel blade 10.65 in. wide x 40.5 in. long x 0.125 in. thick.

All four blades move vertically between aluminum shrouds which form a water channel that is 0.7275 in. thick. When fully withdrawn, the poison section is 30 in. above the bottom of the fuel meat. When fully inserted, the poison section is 4 in. below the bottom of the fuel meat.

In the reactor diffusion theory model, the boral poison section of each control blade was modeled as one region. The aluminum shrouds, the water gaps between the shrouds and each control blade, and the aluminum cladding of each control blade were homogenized and modeled as two separate regions on opposite sides of the boral poison section. The stainless steel regulating blade was modeled in a similar manner.

5.5.2 Methods for Calculating Blade Worths

Normal diffusion theory was used to calculate the reactivity worth of the regulating rod because stainless steel is not a highly absorbing material. However, normal diffusion theory is not valid for the highly absorbing material of the control blades.

To model the poison section of the control blades, blackness coefficients, which form a pair of internal boundary conditions applicable on the surfaces of the absorber slab, were evaluated from integral transport calcluations with the EPRI-CELL code. Effective diffusion parameters, Σ_a and D, for the poison section of the control blades were then determined as functions of these blackness coefficients and utilized in the diffusion model.

This methodology is described in detail in Ref. 12 along with comparisons of calculated and measured control rod worths in the FNR at the University of Michigan. Calculated and measured values are in quite good agreement. Additional verification of the methodology has been obtained through comparisons[12] with detailed Monte Carlo calculations for several other research reactors.

5.5.3 Calculated Reactivity Worths of Blades

Calculated reactivity worths for the regulating blade and the control blades of the HEU and reference LEU cores are shown in Table 6.

Table 6. **Reactivity Worths of Regulating Blade and Control Blades**

Blade Configuration	Worth of Blades, % Δk/k HEU Core	LEU Core
Regulating Blade Fully-Inserted 3 Control Blades Fully-Withdrawn	0.7	0.7
Control Blades 1 and 2 Fully-Inserted Control Blade 3 and Reg. Blade Withdrawn	6.5	6.5
Control Blades 2 and 3 Fully-Inserted Control Blade 1 and Reg. Blade Withdrawn	7.8	7.7
Control Blades 1 and 3 Fully-Inserted Control Blade 2 and Reg. Blade Withdrawn	8.2	8.1
Control Blade 2 Fully-Inserted Control Blades 1, 3, and Reg. Blade Withdrawn	3.6	3.5

5.5.4 Criticality Condition

The reactivity worth of the regulating blade in both the HEU and reference LEU cores was calculated (Table 6) to be 0.7% Δk/k when the tip of the blade is inserted 4 in. below the bottom of the fuel meat. This worth is less than the calculated effective delayed neutron fraction (0.77%, Table 8) in both cores. Since the maximum permissible cold, clean excess reactivity is 0.5% Δk/k, the reactor will be subcritical by a minimum of about 0.2% Δk/k with only the regulating blade inserted.

5.5.5 Shutdown Margin

The reactivity worths of various configurations of the control blades in the HEU and reference LEU cores with the regulating blade fully-withdrawn are shown in Table 6. The control blade with maximum reactivity worth in both cores is blade No. 3 (see Fig. 1). When this blade is stuck out of the core, control blades No. 1 and 2 have a reactivity worth of 6.5% $\Delta k/k$. If the critical core were loaded to its maximum cold clean excess reactivity of 0.5% $\Delta k/k$, the reactor would be subcritical by 6.0% $\Delta k/k$.

5.5.6 Regulating Blade Withdrawal

The WPI Technical Specifications state that the maximum reactivity addition rate through movement of the regulating blade shall be 0.006% $\Delta k/k$ per second.

The reactivity worth profile of the regulating blade is an S-shaped curve with maximum slope at the fuel meat centerline. Since the nominal height of the fuel meat is 23.5 in., the fuel meat centerline is located 11.75 in. above the bottom of the meat. Table 7 shows the calculated worths of the regulating blade at 2.94 in. above and below the fuel meat centerline in the HEU and reference LEU cores.

The slope of the regulating blade worth curve at the fuel meat centerline is 0.036% $\Delta k/k$ per in. in both the HEU core and the reference LEU core. Since the maximum withdrawal rate of the regulating blade is 3.8 in. per minute, the maximum reactivity addition rate is 0.002% $\Delta k/k$ per second in both cores.

Table 7. **Reactivity Worth of Regulating Blade Near Core Midplane with the Three Safety Blades Withdrawn**

Height of Regulating Blade From Core Midplane	Worth of Regulating Blade, % $\Delta k/k$ HEU Core	Worth of Regulating Blade, % $\Delta k/k$ LEU Core
+ 2.94"	0.18	0.19
− 2.94"	0.39	0.40

5.6 Reactor Kinetics Parameters

The prompt neutron generation times and effective delayed neutron fractions of the HEU and reference LEU cores were calculated using standard perturbation theory techniques in the PERT2D code.[13] Axial extrapolation lengths were first determined using fluxes from the 3D reactor calculations. A two-dimensional reactor model was then used to compute the real and adjoint flux distributions needed for the perturbation calculations. The results are shown in Table 8.

Table 8. **Prompt Neutron Generation Times and Effective Delayed Neutron Fractions**

Parameter	HEU Core	LEU Core
Prompt Neutron Generation Time, μs	69.0	61.2
Effective Delayed Neutron Fraction, %	0.77	0.77

5.7 Temperature, Void, and Doppler Coefficients

The WPI Technical Specifications state that the temperature and void coefficients of reactivity shall be more negative than -2×10^{-5} $\Delta k/k \cdot °F$ (-1×10^{-5} $\Delta k/k \cdot °C$) and -2×10^{-3} $\Delta k/k \cdot \%$ void, respectively, at 80°F (26.7°C).

Core average reactivity feedback coefficients were computed separately (using 3D reactor models) as functions of temperature or void fraction for each of three physical effects: (1) hardening of the neutron spectrum due to increasing the water temperature only, (2) the increase in neutron leakage due to decreasing the water density only, and (3) the increase in absorption of the ^{238}U epithermal resonances due to increasing the temperature of the fuel meat only.

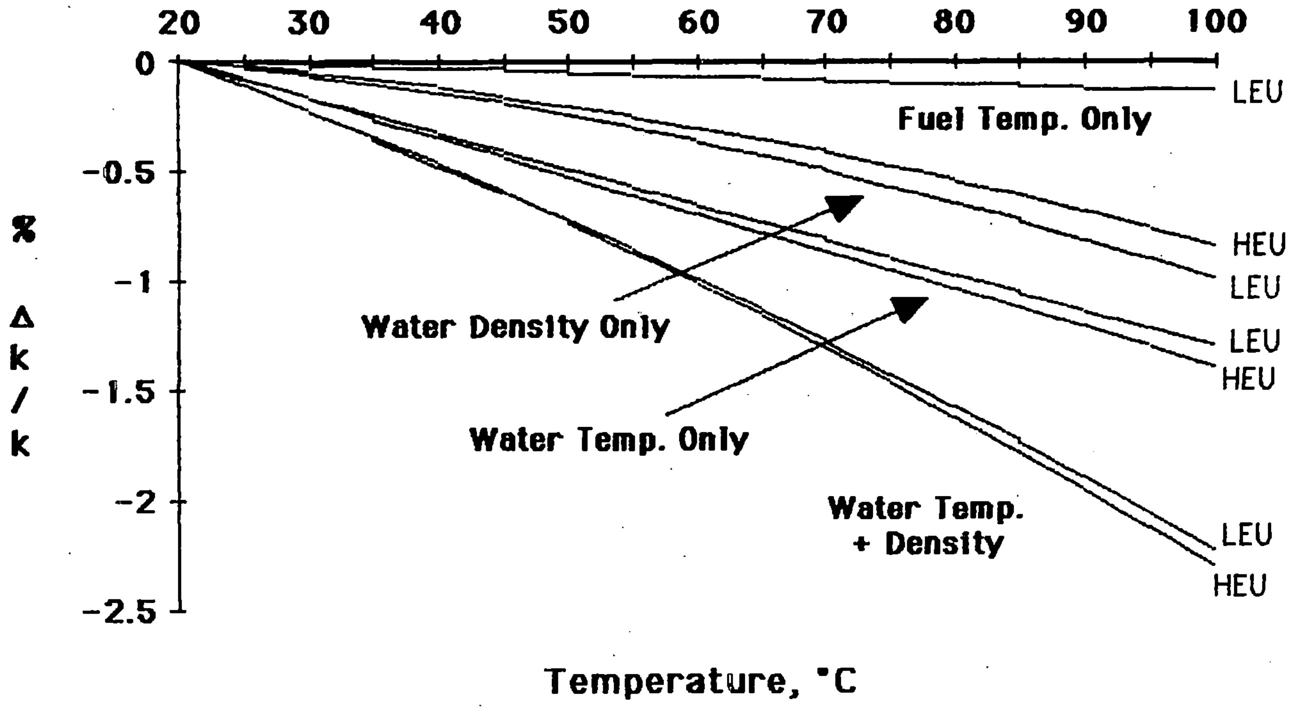
Slopes of the core average reactivity feedback components between 25°C and 30°C are shown in Table 9 along with the whole-core void coefficient for a 1% change in water density only. Temperature and void coefficients are positive in the water gap left by the control blades.

Table 9. **Core Average Temperature, Void, and Doppler Coefficients With All Blades Withdrawn**

Effect		HEU Core	LEU Core
Water Temperature Only		−1.7	−1.6
Water Density Only,		−0.5	−0.6
$\Delta k/k \times 10^{-4}$ per °C	SUM	−2.2	−2.2
Fuel Temperature (Doppler) Only, $\Delta k/k \times 10^{-5}$ per °C		0.0	−1.8
Void Coefficient (0–1% Void), $\Delta k/k \times 10^{-3}$ per % void		−2.0	−2.4

The HEU and LEU cores have the same reactivity feedback coefficient when the water temperature and density effects are combined. However, since the Doppler coefficient is significantly larger in the LEU core, the total feedback coefficient is larger with LEU fuel. Note also that the fuel temperature normally rises more rapidly than the water temperature. The whole-core void coefficient is significantly larger in the LEU core because it has a harder neutron spectrum. These coefficients are plotted in Fig. 7 over the temperature range between 20°C and 100°C.

Fig. 7 Temperature, Void, and Doppler Coefficients with All Blades Withdrawn.



5.8 Alterations of Core Geometry

Alteration of the core geometry affects the available excess reactivity. The configuration of the present 24 element HEU core and the 24 element reference LEU core are shown in Fig. 1. Calculated changes in the excess reactivity for several alterations of this core geometry are shown in Table 10.

Table. 10. **Reactivity Changes as a Result of Alterations of Core Geometry**

Alteration	HEU Core (Reactivity Change, % Δk/k)	LEU Core (Reactivity Change, % Δk/k)
1. A fuel element is added to position F3	+1.0	+1.0
2. The 5 elements in Row B are moved to positions C8, D8, E8, C2, and E2	-0.6	-0.8
3. The 4 elements in Row F are moved to positions A4-A7	-1.0	-1.0
4. The center fuel element (D5) is removed	-5.1	-4.6

Alteration 1 will not make the reactor critical if at least one of the three control blades is in the core. Alterations 2-4 would make the reactor subcritical.

5.9 Thermal-Hydraulic Safety Margins

Thermal-hydraulic safety margins were evaluated using NATCON, a natural-convection heat-transfer code[14] written by the RERTR Program at ANL for analysis of plate-type research reactors.

Input parameters required for analysis of the limiting (hottest) channel in the core include (1) nuclear power peaking factors for the limiting fuel element, and (2) engineering hot channel factors that account for fuel fabrication tolerances, uncertainties in calculated parameters, and uncertainties in certain measured parameters such as the reactor power.

The nuclear power peaking factors calculated for the WPI HEU and reference LEU cores are shown in Fig. 6. The limiting fuel element in both cores (from a thermal-hydraulic safety-margin point of view) is the element located in grid position E6, adjacent to the regulating blade. The total nuclear power peaking factors were computed to be 1.99 in the HEU core and 2.17 in the reference LEU core.

The hot channel factors that were used are based in part on the LEU fuel plate specifications for the FNR at the University of Michigan and in part on steady-state computations with natural convection flow. The original specifications for the current WPI HEU fuel elements are not available.

The hot channel factors and the hot channel components used in the calculations[14] for the WPI reactor are summarized in Table 12.

Table 12. **Hot Channel Factors and Hot Channel Components**

Uncertainty	F_q (a)	F_H (b)	F_h (c)
Fuel Plate Thickness	1.07	-	-
^{235}U Loading	1.02	1.02	-
^{235}U Homogeneity	1.20	1.10	-
Coolant Channel Thickness	-	1.16	1.14
Power Measurement	1.05	1.05	-
Calculated Power Density	1.10	1.10	-
Coolant Flow Rate	-	1.10	1.10
Heat Transfer Coefficient	-	-	1.20
Multiplicative Combination	1.51	1.65	1.51
Statistical Combination	1.24	1.24	1.41[d]

[a]Factor for uncertainties that affect the heat flux.
[b]Factor for uncertainties in flow or enthalpy change in coolant channel.
[c]Factor for uncertainty in heat transfer to the coolant.
[d]Factors for coolant channel thickness and coolant flow rate are treated statistically. Factor for heat transfer coefficient is multiplicative.

5.9.1 Overpower Factor

The margin to Onset of Nucleate Boiling (ONB) for the hot channel is based on the Bergles-Rohsenow correlation[15] and the above nuclear power peaking factors and hot channel factors for both the 10 plate HEU elements and the 18 plate LEU elements. The margin to ONB is sometimes refered to as the overpower factor. That is, the ratio of the reactor power at ONB to the nominal reactor power (10 kW).

In these calculations, the reactor power was increased until an indication of ONB was obtained in the hottest channel, according to the Bergles-Rohsenow correlation for three combinations of the engineering hot channel factors. The coolant inlet temperature in all cases was 26.7 C. The results are shown in Table 13.

Table 13. **Calculated Overpower Factor**

Hot Channel Factors			Reactor Power for Incipient Boiling,	Maximum Clad Surface Temp., °C		Maximum Fuel Centerline Temp., °C	
F_q	F_H	F_h	kW	10 kW	ONB	10 kW	ONB
				HEU Core			
1.00	1.00	1.00	482[a]	31	109	31	109
1.24	1.24	1.41	248[b]	33	109	33	108
1.65	1.51	1.51	169[c]	36	108	36	108
				LEU Core			
1.00	1.00	1.00	818[a]	33	109	33	109
1.24	1.24	1.51	442[b]	36	109	36	109
1.65	1.51	1.51	301[c]	39	108	39	108

[a]Nominal Case, no hot channel factors.
[b]Hot channel factors combined statistically, except for heat transfer coefficient factor, which is multiplicative.
[c]Multiplicative hot channel factors.

With the use of statistically combined subfactors, the margin to ONB (or the overpower factor) for the HEU core with 10 plates per element is 24.8 and the margin for the LEU core with 18 plates per element is 44.2. With the larger multiplicatively combined factors, the margins to ONB were calculated to be 16.9 and 30.1 for the 10 plate HEU elements and 18 plate LEU elements, respectively. The reason for the larger overpower factor in the LEU core is the lower initial heat load with 18 fuel plates per element rather than 10 fuel plates per element in the HEU core.

The maximum temperatures at the surface of the cladding and at the fuel meat centerline at ONB are a few °C above the saturation temperature (106°C) of the coolant. For the three sets of hot channel factors, the temperature for incipient boiling is reached at different power levels.

6. ACCIDENT ANALYSIS

Several accident scenarios were evaluated by the Nuclear Regulatory Commission (NRC) staff in December 1982 as part its review[4] of an application by WPI for a renewed operating license. The accidents that were considered included: (1) inadvertent excess reactivity insertion, (2) metal-water reactions, (3) graphite fire, (4) mechanical damage, including crushing of the core, (4) water damage, and (5) a fuel-handling accident.

In the Safety Analysis Report[2] submitted to support its licence renewal, WPI had considered a number of accident scenarios. The NRC staff concluded[4] that the consequences of all of these accidents fell within the envelope of accidents analyzed by the NRC staff and by the Los Alamos National Laboratory (LASL) and Brookhaven National Laboratory (BNL) under contract to NRC. Consequently, the accident scenarios considered here are the five accident scenarios that were considered by the NRC staff, LASL, and BNL.

Of these, only two accident scenarios (inadvertent excess reactivity addition and fuel handling accident) could be affected by changing the core fuel from HEU to LEU, and only these two scenarios are addressed here.

6.1 Inadvertent Excess Reactivity Insertion

ANL has evaluated the maximum inadvertent stepwise reactivity insertion that would lead to initiation of clad melting in the WPI HEU and LEU cores using the PARET code.[16]

6.1.1 Comparison of Calculations with SPERT-I Experiments

The PARET code was originally developed at the Idaho National Engineering Laboratory for analysis of the SPERT-III experiments, which included both pin-type and plate-type cores and pressures and temperatures into the range typical of power reactors. The code was modified by the RERTR Program at ANL to include a selection of flow instability, departure from nucleate boiling, single- and two-phase heat transfer correlations and a properties library applicable to the low pressures, temperatures, and flow rates encountered in research reactors.

To validate the PARET code, calculated and measured data were compared[17] for three SPERT-I[18,19] HEU cores B-24/32 (32 element core with 24 fuel plates per element), B-12/64, and D-12/25. These cores were similar in design to many plate-type research reactors in current operation. The D-12/25 core was of particular interest because the tests included both nondestructive and destructive transients. The results of these analyses are generally quite good and validate the PARET code for use in calculating research reactor transients.

6.1.2 Analyses for the WPI Reactor

The same model and methods that were used for analysis of the SPERT-I HEU cores were also used to analyse the HEU and reference LEU cores of the WPI reactor.

Inputs to the code included the prompt neutron generation time, effective delayed neutron fraction, reactivity feedback coefficients, and nuclear power peaking factors. Axial power distributions were represented by chopped cosine shapes having calculated peak-to-average power densities of 1.51 (1.76) in the hot channel and 1.43 (1.68) in the average channel of the HEU (LEU) core.

Calculations with various stepwise reactivity insertions were performed until the peak temperature at the surface of the cladding reached the aluminum solidus temperature (660°C in the 1100 Al of the HEU core and 582°C in the 6061 Al of the LEU core). The limiting step insertion was calculated to be 1.5% $\Delta k/k$ in the HEU core and 2.0% $\Delta k/k$ in the LEU core. The limiting value is higher in the LEU core because the LEU core has a prompt negative Doppler coefficient and larger void coefficient than the HEU core.

The WPI Technical Specifications limit the maximum cold clean excess reactivity to 0.5% $\Delta k/k$. For a stepwise reactivity insertion of 0.5% $\Delta k/k$ and no reactor scram, the peak fuel centerline and cladding temperatures were calculated to be 100.5°C in the HEU core and 100.4°C in the reference LEU core. These temperatures are far below the solidus temperatures of the cladding in both cores.

6.2 Fuel Handling Accident

The most serious hypothetical fission product release scenario considered in the NRC rewiew was dropping a fuel-handling cask containg a fully irradiated fuel element, leading to a breach of the cladding integrity and subsequent release of fission products to the atmosphere. A detailed analysis of this type of accident performed for an MTR-type fuel element (NUREG/CR-2079) extrapolated[4] to the WPI reactor's 10 kW power level would result in whole-body and thyroid doses that are small fractions of the 10 CFR 20 requirements.

This conclusion is also valid for the WPI reactor loaded with LEU fuel since the fission product inventories of the HEU and LEU cores would be virtually identical for the same operating conditions.

The only difference between the HEU and LEU cores is that the LEU core would contain very small quantities of plutonium because its ^{238}U inventory is much larger. The RERTR Program at ANL performed a detailed analysis[20] comparing the radiological consequences of a hypothetical accident in a generic 10 MW reactor using HEU and LEU fuels. This analysis concluded that the buildup of plutonium in the LEU fuel of a 10 MW reactor with an average ^{235}U burnup of over 50% does not significantly increase the radiological consequences over those of HEU fuel. Because the ^{235}U burnup over the lifetime of the 10 kW WPI reactor is expected to be very low, the buildup of plutonium and its radiological consequences will be negligible.

REFERENCES

1. The University of Michigan, "Safety Analysis for Utilization of Low Enrichment Uranium (LEU) Fuel in the Ford Nuclear Reactor", October 1979.

2. Safety Analysis Report for the Worcester Polytechnic Institute Reactor for Presentation to the United States Nuclear Regulatory Commission, September 25, 1979.

3. Worcester Polytechnic Institute, "Appendix A to License No. R-61, Technical Specifications for the Worcester Polytechnic Institute Reactor, Docket No. 50-134," (1982).

4. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Renewal of the Operating License for the Worcester Polytechnic Institute Open-Pool Training Reactor," NUREG-0912, December 1982.

5. M. M. Bretscher and J. L. Snelgrove, "Comparison of Calculated Quantities with Measured Quantities for the LEU-Fueled Ford Nuclear Reactor," Proc. International Meeting on Research and Test Reactor Core Conversions from HEU to LEU Fuel, Argonne National Laboratory, Argonne, IL, November 8-10, 1982, ANL/RERTR/TM-4, CONF-821155, pp. 397-425 (1983).

6. M. M. Bretscher, "Analytical Support for the Whole-Core Demonstration at the ORR," Proc. 1986 International Meeting on Reduced Enrichment for Research and Test Reactors, Gatlinburg, TN, November 3-6, 1986, ANL/RERTR/TM-9 (to be published).

7. R. J. Cornella and M. M. Bretscher, "Comparison of Calculated and Experimental Wire Activations," Proc. 1986 International Meeting on Reduced Enrichment for Research and Test Reactors, Gatlinburg, TN, November 3-6, 1986, ANL/RERTR/TM-9 (to be published).

8. IAEA Guidebook on Research Reactor Core Conversion from the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels, IAEA-TECDOC-233, Appendix F, pp. 443-628 (1980).

9. J. E. Matos, E. M. Pennington, K. E. Freese, and W. L. Woodruff, "Safety-Related Benchmark Calculations for MTR-Type Reactors with HEU, MEU, and LEU Fuels," paper included in IAEA Safety and Licensing Guidebook on Research Reactor Core Conversions from HEU to LEU Fuel, Volume 2, Analytical Verification, Draft #7, June 1985.

10. B.A. Zolatar et al., "EPRI-CELL Code Description," Advanced Recycle Methodology Program System Documentation, Part II, Chapter 5 (Oct. 1975).

11. K.L. Derstine, "DIF3D: A Code to Solve One-, Two-, and Three-Dimensional Finite-Difference Diffusion Theory Problems," ANL-82-64, April 1984.

12. M. M. Bretscher, "Blackness Coefficients, Effective Diffusion Parameters, and Control Rod Worths for Thermal Reactors," ANL/RERTR/TM-5, Sept. 1984.

13. T. A. Daly, et al., "The ARC System Two-Dimensional Adjunct Calculations," ANL-7720 (Oct. 1972).

14. R. S. Smith and W. L. Woodruff, "Thermal-Hydraulic Analysis and Safety Margins for Natural Convection Cooled Research Reactors," ANL Internal Memorandum, July 13, 1987.

15. A. E. Bergles and W. M. Roshenow, "The Determination of Forced Convection Surface Boiling Heat Transfers," Transactions of the ASME 86 (Series C - Journal of Heat Transfer), pp. 365-371 (August 1964).

16. C. F. Obenchain, "PARET - A Program for the Analysis of Reactor Transients," IDO-17282, Idaho National Engineering Laboratory (1969).

17. W. L. Woodruff, "A Kinetics and Thermal-Hydraulics Capability for the Analysis of Research Reactors," Nuclear Technology 64, pp. 196-206, (Feb. 1984). And W. L. Woodruff, "The PARET Code and the Analysis of SPERT I Transients," Proc. International Meeting on Research and Test Reactor Core Conversions from HEU to LEU Fuel, Argonne National Laboratory, Argonne, IL, November 8-10, 1982, ANL/RERTR/TM-4, CONF-821155, pp. 560-578 (1983).

18. A. P. Wing, "Transient Tests of the Fully Enriched, Aluminum Plate-Type, B Cores in the SPERT I Reactor," IDO-16964, Idaho National Engineering Laboratory (1964).

19. M. R. Zeissler, "Non-Destructive and Destructive Transient Tests of the SPERT I-D, Fully Enriched, Aluminum Plate-Type Cores: Data Summary Report," IDO-16886, Idaho National Engineering Laboratory (1963).

20. W. L. Woodruff, D. K. Warinner, and J. E. Matos, "A Radiological Consequence Analysis with HEU and LEU Fuels," Proc. 1984 International Meeting on Reduced Enrichment for Research and Test Reactors, Argonne National Laboratory, Argonne, IL, October 15-18, 1984, ANL/RERTR/TM-6, CONF-8410173, pp. 472-490 (1985).