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START-UP TEST REPORT

FOR RPI REACTOR CRITICAL FACILITY

by

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ABSTRACT

Rensselaer Polytechnic Institute's Reactor Critical Facility (RCF) consists of an open-pool type, very low power, nuclear-fueled reactor. The reactor's fuel was recently changed from highly enriched uranium (HEU) to low enriched uranium (LEU).

This paper describes the changes to the RCF, including the LEU fuel, and the changes and modifications necessary to accommodate the LEU fuel. This paper also provides the results of measurements and calculations of the LEU core and associated components.

The analysis shows that that the requirements of the Safety Analysis Report and Technical Specifications have been met, and that the LEU reactor configuration is well understood such that safe and proper operation are reasonably assured.

PART 1

INTRODUCTION AND HISTORICAL REVIEW

The Reactor Critical Facility (RCF) at Rensselaer Polytechnic Institute (RPI) had operated since 1963 with highly enriched uranium (HEU) fuel plates clad in stainless steel, containing a total of 6.01 kg of U^{235}. In the 1970's, concern was mounting within the Nuclear Regulatory Commission (NRC), which licenses the RCF to operate, over the physical protection requirements for highly enriched ("bomb grade") uranium. This concern resulted in changes to the Code of Federal Regulations, Title 10, Part 73.47 (10CFR73.47) , published in 1980 [1].

The rule changes required mandatory enhanced physical protection measures for quantities of HEU greater than 5 kg, which automatically included the RCF. Realizing that the physical protection requirements would be prohibitively expensive for RPI, options were explored which could exclude the RCF from these requirements.

One option was to reconfigure the core, ie., to rearrange the fuel to make it more reactive, with less, than or equal to 5.0 kg. After a great deal of effort, the core was successfully reconfigured with less than the maximum of 5 kg of U^{235}.

Then, in the early-mid 1980's, the NRC, became increasingly concerned about the possibility that any amount of HEU fuel used in non-power reactors, such as the RCF, could be diverted from its intended peaceful purposes.

As a result of these concerns, the NRC had mandated, in the form of a "show-cause" order, that all NRC-licensed non-power reactors using HEU fuel, including the RCF, be converted to low enriched uranium (LEU) fuel unless compelling reasons could be given for continued use of HEU fuel. For this reason, it was decided to convert the RCF to LEU fuel, and the successful conversion to LEU fuel has been completed, with the first criticality sustained in October 1987 [2-6].

The conversion from HEU to LEU fuel was made possible by several factors. First, the early SPERT program had made available a large number of usable high-quality LEU fuel pins; thus the manufacturing of new LEU fuel wasn't required. Second, these SPERT fuel pins were clad in stainless steel making them compatible with the existing stainless steel reactor tank and core internals and support structure.

Third, even though the LEU fuel was longer than the HEU fuel, the core structure and reactor tank were adequately sized to accept the longer fuel. Fourth, the control rods for the HEU core consisted of demountable boxes containing one flux trap absorber section and one fuel follower section. The fuel follower type control rods were removed and replaced with pure absorber flux trap type control rods. Fifth, the entire reactor structure could be disassembled and reassembled without cutting.

The entire conversion process was guided by the following steps:

(a) Nuclear/mechanical redesign of core and support structure;

(b) Safety analysis to verify new core design for conversion application;

(c) Fabrication of new core support structure components;

(d) Disassembly of old core structure and peripherals;

(e) Assembly of new core structure and peripherals;

(f) Redesign, criticality analysis, and reconstruction of the fuel storage vault;

(g) Shipment of HEU fuel off site;

(h) Shipment of SPERT fuel to RPI and storage in fuel storage vault;

(i) Analysis and verification of instrument response, shielded doses, and other reactor parameters for validation of Technical Specifications;

(j) Alteration of Technical Specifications and related documents based on operational data;

(k) Loading of new core and experimental verification of reactor safety parameters; and,

(l) Redevelopment of instructional curriculum and operational materials.

With the exception of item (l) above, all actions have been completed, and work is presently on-going to ensure the completeness of item (l).

It is the intent of this paper to organize into a single entity, the results of many experimental measurements and calculations which have been performed as a result of the fuel conversion. Also described are the changes to the facility incurred in the conversion.

The remainder of this paper is divided into several sections. Part 2 discusses the RCF and the changes to the facility from the conversion. Part 3 examines the results of measurements and calculations used to verify proper operating characteristics of the facility, and a summary and discussion is presented in Part 4.

PART 2

DESCRIPTION OF FACILITY

2.1 RCF Site

The RPI Reactor Critical Facility is located on the south bank of the Mohawk River in Schenectady, New York. The RCF site, shown in Figure 2-1, is bordered by two exclusion zones. The inner zone contains the site buildings and is enclosed by a secured chain link fence with two access gates. The civil exclusion zone, partially fenced, encompasses the access road from Maxon Road, the site parking lot and the inner zone.

The structures at the site consist of an unused guard shack at the personnel entrance, the main building, and an auxiliary building which houses the heating equipment for the main building.

The main building consists of the reactor room, which contains the reactor and associated components and the fuel storage vault; the control room, which contains the reactor control panels; a counting room, which contains radioactive particle counting apparatus; an office, and lavatory. A plan view of the main building is shown in Figure 2-2.

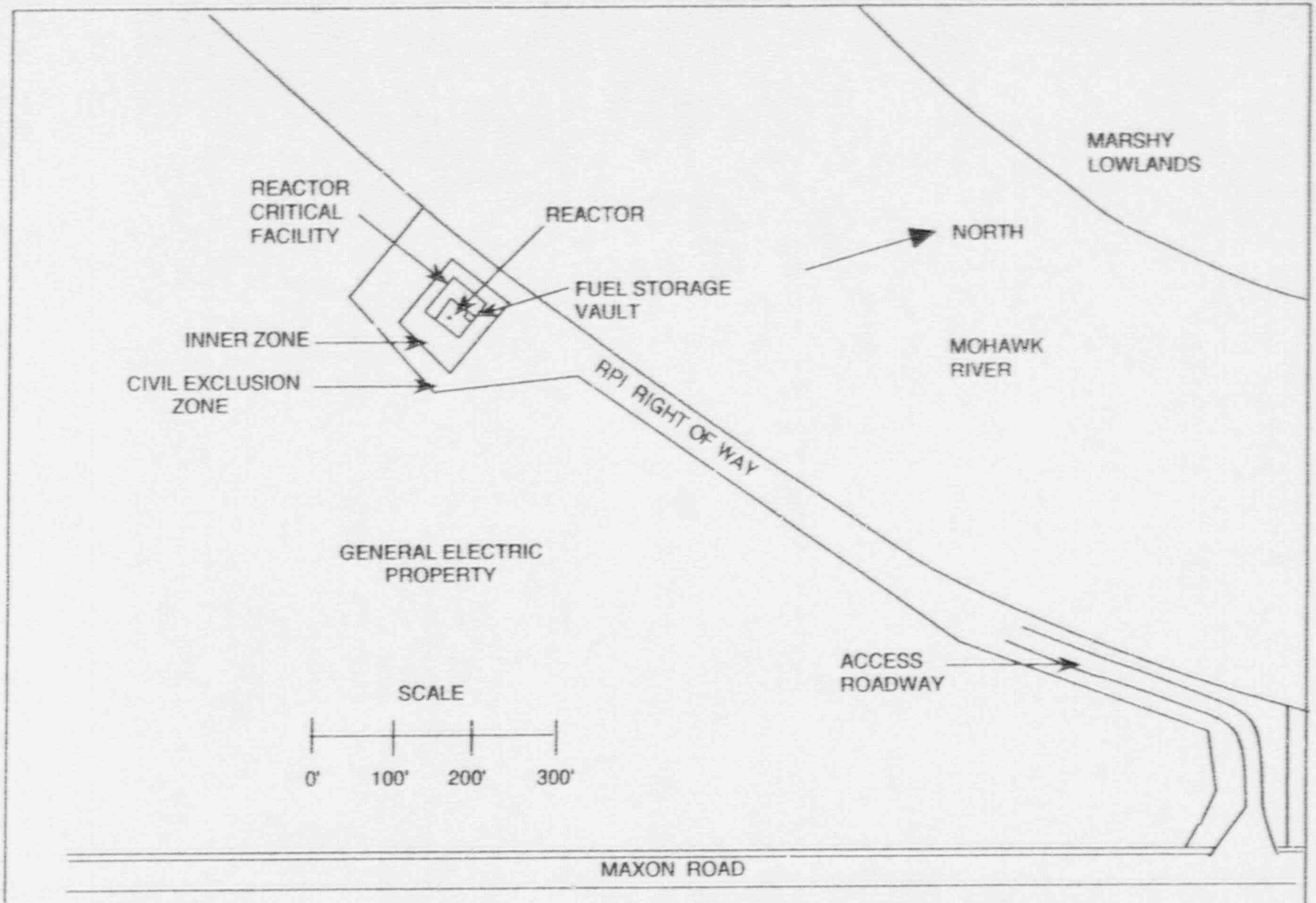
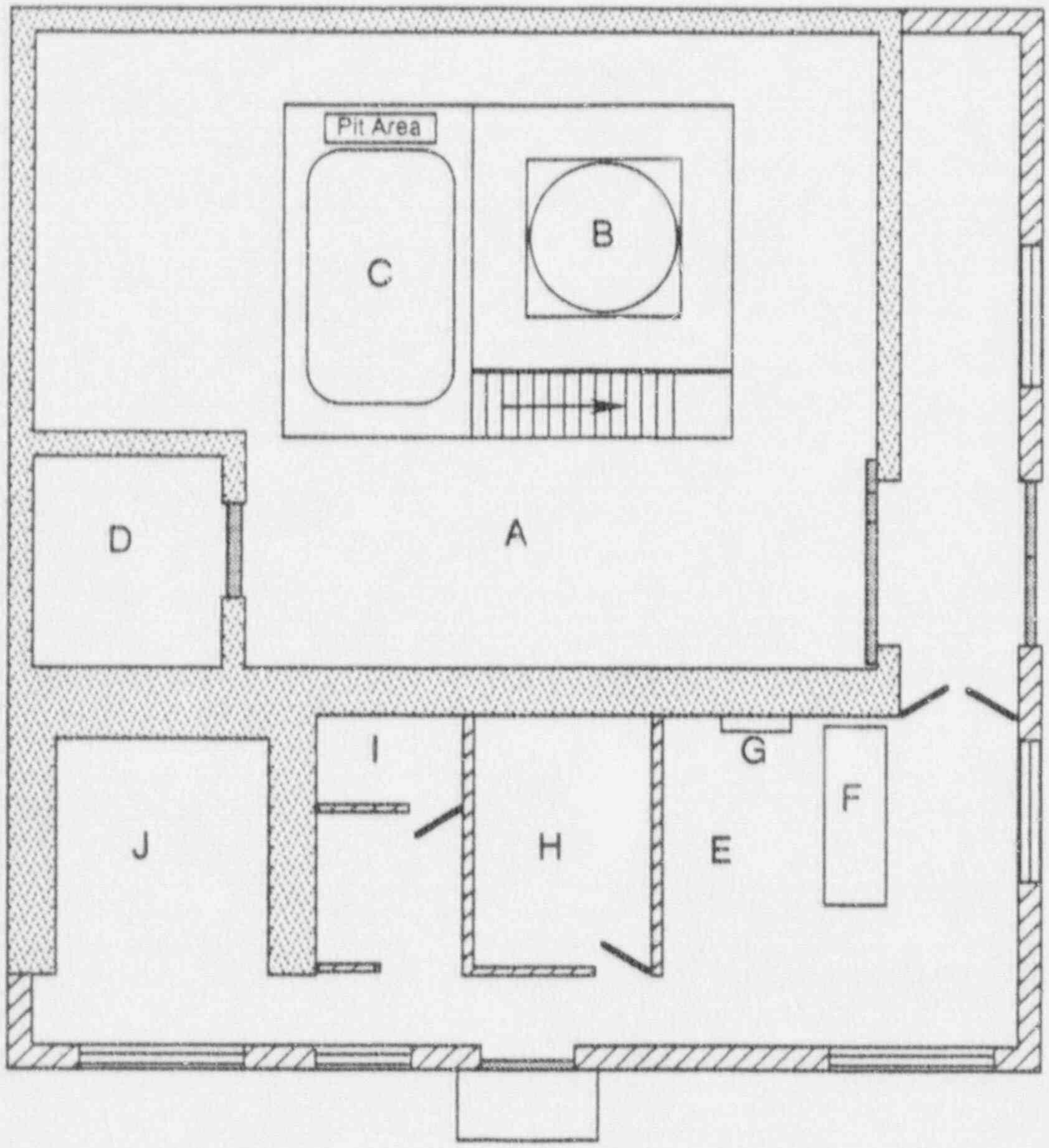


Figure 2-1: Plan View of RCF Site

Figure 2-2: Plan View of Main Building



A -- Reactor Room
B -- Reactor Tank
C -- Reactor Water Storage Tank
D -- Fuel Storage Vault
E -- Control Room

F -- Main Control Panel
G -- Auxiliary Control Panel
H -- Office
I -- Lavatory
J -- Counting Room

2.2. Reactor Components

2.2.1. Reactor and Core Support Structure

The LEU core is supported by the core support structure, which in turn is supported by, and rests, in the reactor tank, which is supported by a steel and concrete structure underneath the tank. The reactor tank is a 2000 gallon stainless steel cylindrical tank open to the atmosphere at the top. The movements of control rods and neutron source, and the removal and insertion of fuel into and out of the reactor take place through this top opening. The reactor and associated components are located in the reactor room.

The reactor core and support structure, which are centered in the tank, comprise only a small portion of the total tank volume. Also located in the tank, but exterior to the core structure, are several power monitors to allow adequate reactor power measurement; several thermocouples for water temperature monitoring; immersion heaters tc increase the water temperature; and, a propeller-type agitator which can be used in conjunction with the immersion heaters to aid in reaching thermal equilibrium. All of these components are secured in the reactor tank such that they could pose no interference to the motion of the control rods.

When the reactor is in operation, the reactor tank is normally filled with water to a level of not greater than 10" above the top grid of the core. When the reactor is not in operation, this water is stored in a storage tank located in a pit beneath the reactor tank, as is the pump used for reactor fill. Reactor drain into the storage tank is by gravity flow. All water connections are hard-piped to the underside of the reactor tank.

No modifications were made to the reactor tank, to the water transport system, nor to the reactor tank components exterior to the core structure in the conversion process from HEU fuel to LEU fuel.

The core support structure is an all stainless steel structure. The structure consists of three stainless steel plates, one each at the core bottom (carrier plate), the core midplane (middle plate) and core top (top plate). This three-tiered support structure is anchored by four posts set in the floor of the reactor tank. The structure just described represents the support structure of the HEU core, and has been modified to support the LEU fuel; but, the former structure has remained intact in the revised design, thus the stability of the LEU core has not been diminished.

Enhancements to the core support structure include the addition of two stainless steel lattice plates, one positioned on top of the top plate, and the second positioned on top of the carrier plate. All plates are secured with tie rods and bolts to the four anchor posts. A diagram depicting the core support structure is shown in Figure 2-3.

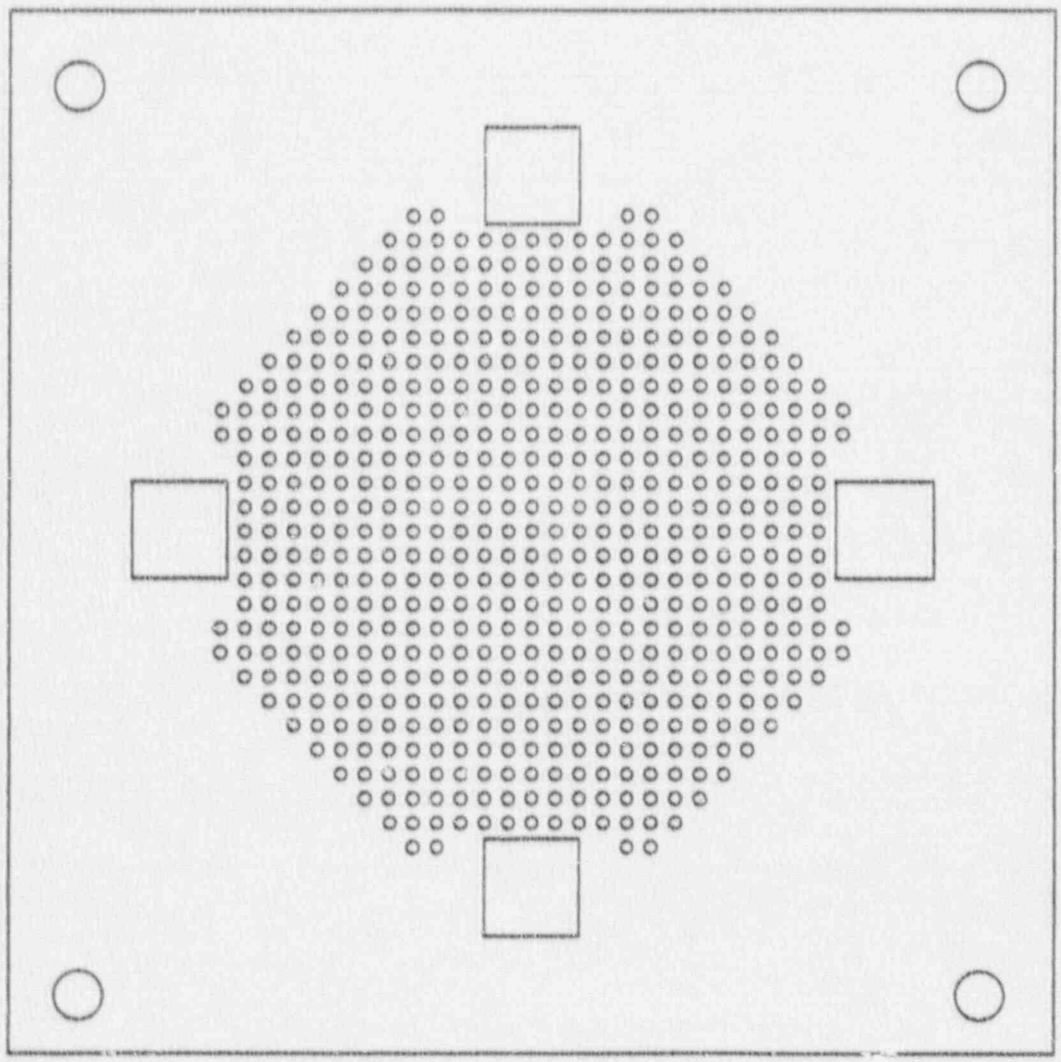
Each of the two fuel lattice plates are nearly identical and define the possible locations of fuel pin placement within the core. Individual fuel pins are loaded into the core by lowering through the drilled holes in the upper lattice plate and rest in identically-positioned, smaller-drilled holes in the lower lattice plate. The upper fuel lattice plate provides for lateral support for the fuel and the lower fuel lattice plate provides for both lateral and vertical support of the fuel.

A diagram of the upper fuel lattice plate is shown in Figure 2-4. The figure shows the fuel positions (the small circles) and the locations of the control rods (the squares) relative to the core. The lattice plate has 553 fuel pin locations, but a nominal fuel loading consists of 497 pins, built in a solid, roughly cylindrical shape. The pitch of the fuel pins allowed by the lattice plate is 1.4859 cm (0.585 inches).

2.2.2. Control Rods and Control Rod Auxiliaries

Because the active length of LEU fuel (36") is longer than that

Figure 2-4: Upper Lattice Plate



of the HEU fuel (22"), substantive changes have been made to the control rods, while the number of control rods, placement relative to the core, neutron absorption material and design, and control rod drive principles have remained virtually unchanged.

The HEU fuel core used control rods with fuel followers, that is, a control rod consisted of two different sections: the top section contained neutron absorbing material, while the bottom section contained fuel. As the control rod was inserted into the core from above, more of the neutron absorbing section was being placed into the core as more of the fuel section was being removed from the core, traversing beneath the core.

The fuel section was removed and is replaced with a second neutron absorbing section in the present control rods. Each control rod consists of a 2.75 inch square stainless steel tube, 36 inches in effective length, which pass through the upper fuel lattice plate and top plate, and rest on a hydraulic buffer on the bottom carrier plate of the core support structure.

Each of the two absorber sections is held in a stainless steel "basket", positioned one above the other. The boron-10 enriched absorber in each section is held in a stainless steel cermet that is clad with stainless steel.

The number, relative placement, and drive principles of the control rods have remained unchanged in the conversion process. The four control rods, spaced 90 degrees apart at the core periphery, are each supported by a rigid cantilever above the opening of the reactor tank.

The cantilevers also provide support for the control rod drive mechanisms. Each drive mechanism consist of a 1/20 horsepower motor, a magnetic clutch, drive shaft, gear box, the control rod rack, pinion gear, and two anti-backlash synchromotors. The motor provides the motive force for the controlled vertical motion of the control rods. The gear box, magnetic clutch and drive shaft act in conjunction to transmit the motor's rotational motion to the pinion gear meshed with the teeth of the control rod rack, which is physically connected to the top of the control rod.

The two synchromotors provide control rod position indication in the control room (course indication, inches; and fine indication, 0.01 inches). Attached to the bottom of each control rod is a stainless steel control rod guide to ensure correct alignment of the control rods.

Even though the drive principles remained unaltered in the fuel conversion process, there were minor modifications to the drive mechanisms. These included minor changes to the drive gear assembly and recalibration of the synchromotor position indication system to allow for, and indication of, an expanded control rod travel distance from 22 inches to 36 inches, the effective height of the LEU fuel core.

The control rods are remotely manually positioned by the operator through individual switches on the control panel. Electrically, the control rods operate on demand from these control room panel switches, with power supplied to the magnetic holding clutches from the Solenoid Interrupt Circuit.

Under normal conditions, the clutches remain energized to allow the motor-driver to insert and withdraw the control rods as demanded by the operator, and to hold the control rods at the desired position. Under upset conditions, such as a reactor scram signal or power failure, the clutches are deenergized and the drive shaft becomes physically disconnected from the rest of the assembly. As a result, any control rod not fully inserted into the core at the time of the upset is allowed to drop into the core under its own weight, and further manual movement by the operator is not possible until the condition is cleared.

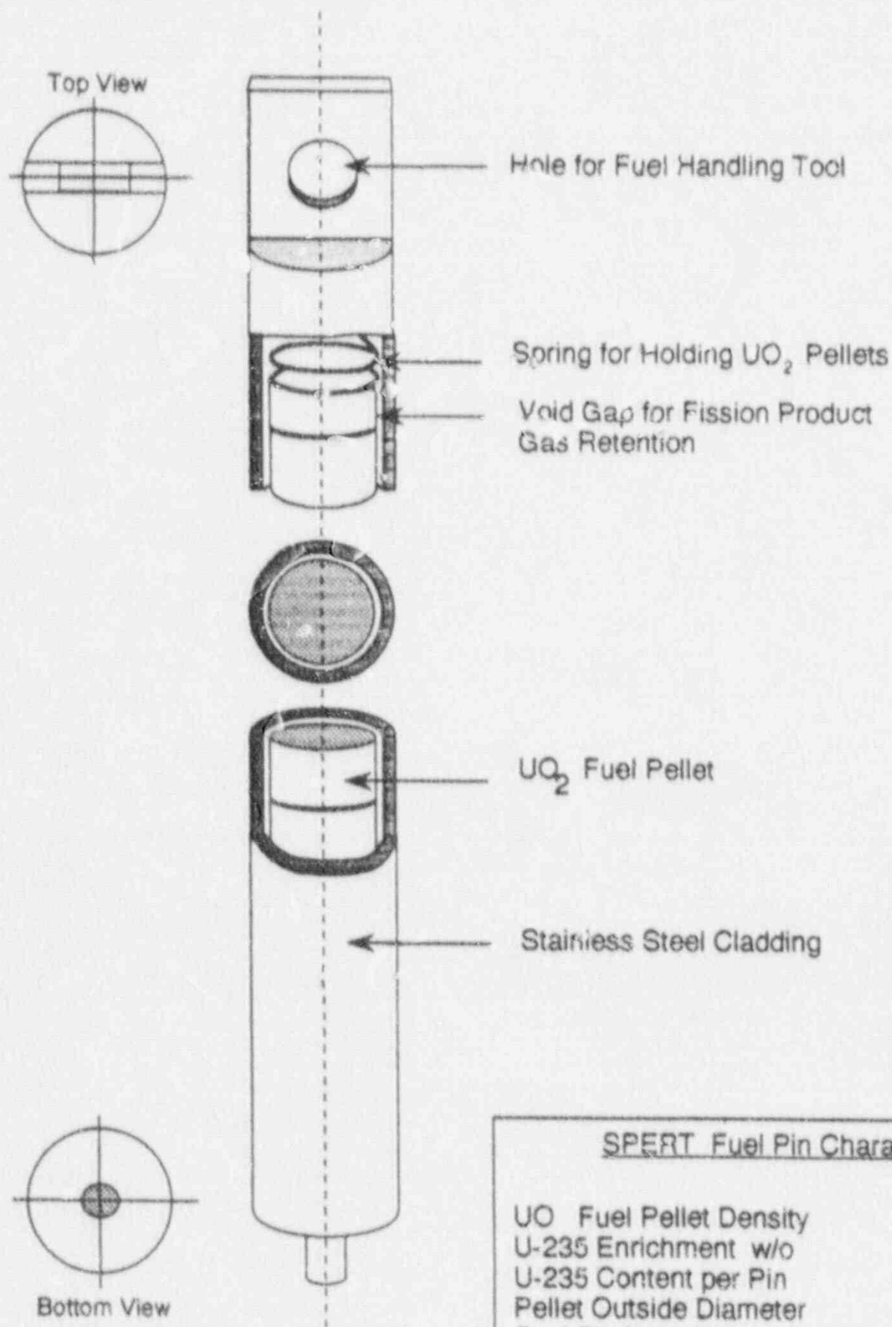
2.2.3. Reactor Fuel

The original fuel consisted of flat plates containing highly enriched (93 w/o) uranium dioxide dispersed in stainless steel cermet with Type 304LB stainless steel cladding. There were two types of fuel assemblies, stationary and movable, both of which consisted of some number of fuel plates. Stationary assemblies remained stationary during operation.

The present LEU (SPERT) fuel consists of fuel pins containing 4.81 w/o enriched UO_2 ceramic pellets clad in Type 304 stainless steel. There are no fuel assemblies per se, thus the fuel pin remains as the fundamental unit. All fuel remains stationary during reactor operation. Except for the difference in effective fuel length and cladding material, the fuel pins are very similar in size and composition to those fuel pins which comprise fuel assemblies in the commercial light water reactor industry. Figure 2-5 shows a typical fuel pin along with it's physical characteristics.

Fuel pins are individually loaded into the reactor by the operator by lowering into a selected hole in the upper fuel lattice plate until the pin rests in a vertically aligned smaller-drilled hole in the lower fuel lattice plate. The holes in the lower lattice plate are specifically sized to accept the lower tip of the fuel pin. Fuel

Figure 2-5 LEU Fuel Pin



SPERT Fuel Pin Characteristics

UO Fuel Pellet Density	10.08 gm/cm ^{3}
U-235 Enrichment w/o	4.81 +/- 0.15
U-235 Content per Pin	35.2 gm
Pellet Outside Diameter	1.067 cm
Fuel Pin Length	106.045 cm
Cladding Outside Diameter	1.184 cm
Cladding Thickness	0.0508 cm
Active Fuel Length	91.44 cm
Cladding Length	105.715 cm
Overall Fuel Pin Length	106.045 cm

pins that are not in the reactor are stored in the dry fuel storage vault. There are currently 595 fuel pins at the RCF which constitutes 20.94 kg of U^{235} inventory.

3.3. Fuel Storage Vault

The fuel storage vault occupies an 8 foot by 10 foot room in the northern corner of the reactor room. Access into the vault is through a normally locked vault-type door, and opening the vault is performed by licensed operators only. The vault ceiling, walls, and floor are made of 1 foot reinforced concrete.

A storage rack constructed of unistrut is mounted on the wall opposite to the access door. Bolted to the unistrut frame are 41 stainless steel tubes, 12.7 cm in diameter, each wrapped with 0.0381 cm thick cadmium sheaths. All of the cadmium-covered tubes were increased in length from approximately 71 cm to about 107 cm to fully accommodate the longer LEU fuel.

Each tube is designed to store fifteen fuel pins, which gives a total storage capacity of 615 fuel pins. At this loading, conservative calculations yield an infinite multiplication factor of much less than 0.90, even under completely flooded conditions. Fuel that is not loaded into the reactor is stored within the vault.

PART 3

MEASUREMENTS AND CALCULATIONS

3.1 Reactivity Measurements

The measurements of core reactivity are divided into two classes: those at or above critical (zero or positive reactivity), and those below critical (negative reactivity) [7]. The measurements in the first class are simple and rather straightforward.

At the critical point (reactivity equals zero), reactor power remains constant with no control rod motion, as demonstrated by straight vertical lines on the power recorders. At above critical, reactivity is determined by the positive period method. After attaining a stable positive period, the reactor period is measured, and through the use of the Inhour equation, which directly relates reactor period to reactivity, core reactivity can be easily determined.

Below the critical point, absolute core reactivity is extremely difficult to measure. The absolute core reactivity below the critical point is determined from reciprocal multiplication measurements.

Some of the measurements (described later) are performed in a two-step process. The initial reactivity of the core is first determined. Then, following reactor shutdown, a perturbation is introduced into the core (ie., a fuel pin is removed, or a void is added). The core reactivity is again determined (ideally, under otherwise identical conditions as the initial reactivity measurement). The difference in the two measured reactivities is attributed to the introduction of the perturbation.

3.2 Analytical Methods

The nuclear analysis of the LEU core was carried out by three computer codes: LEOPARD, DIFXYRZ, and PLATAB [6]. The LEOPARD code computes detailed neutron flux spectra and spatial flux distribution in a pincell (for calculational purposes, the core is made up of pincells or small rectangular regions of given size, which contain either fuel and water [for locations interior to the core], or water or absorber [for regions located in the reflector exterior to the core].) and from these constructs two, or four-energy-group constants for neutron diffusion codes.

The cross section library was processed from ENDF/B-4 nuclear data. The erroneous resonance capture data for U-238 in ENDF/B-4 are not relevant here because LEOPARD uses the experimental Hellstrand correlation for the U-238 resonance integral.

The diffusion code DIFXYRZ is mesh point centered and uses the same difference formulation as does the widely used PDQ code series. The DIFXYRZ code differs from the three dimensional PDQ code in being limited to two dimensions (X-Y or R-Z geometries), in using fixed field inputs, in it's convergence methods, in being only 500 standard Fortran lines in length with every variable defined, and in operating on personal computers as well as on mainframes.

DIFXYRZ has been extensively benchmarked against PDQ results and against known solutions. The combined LEOPARD/DIFXYRZ code package was verified by calculations on two stainless steel clad, high density UO_2, 4.00 w/o enriched, square array critical experiments.

For the strong local absorbers in the control rods, it was necessary to compute effective diffusion parameters to give results compatible with transport theory calculations. The required effective diffusion parameters were calculated using the Fortran code PLATAB. This code is also brief, with every variable defined, and it has been benchmarked against calculated results reported in the literature. Experience in the literature has shown that the compensation of errors appears to be more effective for two-group calculations. Two-group calculations were used in all of the work reported here except for the calculations of effective delayed neutron fraction.

The calculated critical mass was determined for various water and fuel temperatures by first running LEOPARD at each temperature to provide the temperature dependent macroscopic cross sections necessary for input into the diffusion calculations. Given the cross sections and specific fuel loading and placement, DIFXYRZ was used to provide the effective multiplication factor, K_{eff}.

If the resultant Keff did not equal a critical condition (ie., $K_{eff} = 1$), the fuel loading was adjusted and the diffusion code was run again until criticality was achieved. This process continued for each temperature at which the cross sections were initially determined.

Control rod worths were calculated by first computing the effective diffusion parameters for the absorbers using the PLATAB code. Then, control rod in, and control rod out calculations were performed with DIFXYRZ. The resultant change in K_{eff} for the two configurations provided a measure of control rod worth.

The fuel pin worth was calculated at a given temperature by running DIFXYRZ with cross section inputs from LEOPARD. The diffusion code was first run for a given core loading, and then rerun with the removal or addition of individually selected fuel pins. The difference in K_{eff} between the base loaded core and the perturbed core provided a measure of the pin worth.

The isothermal temperature coefficient of reactivity was computed by calculating diffusion parameters for the core and reflector at various temperatures using LEOPARD. Using these constants as inputs, the effective multiplication factor was computed using DIFXYRZ. A relationship was thus formed between temperature and core reactivity. The calculated values of reactivity were then fitted by quadratic functions of temperature.

The void coefficient of reactivity at a given temperature was determined by LEOPARD calculations with a given percent of water removed locally from selected pincells in the core interior. The core reactivity was then calculated using DIFXYRZ. By comparing this core reactivity to that of a fully moderated core at equal temperature, a relationship of reactivity as a function of. void percent was formed.

3.3 Critical Fuel Loading

An inverse multiplication method has been utilized for the initial core loading and approach to criticality. The initial loading step consisted of 105 fuel pins placed symmetrically as possible about the core center. The second and third steps consisted of the addition of 45 fuel pins. Subsequent fuel additions were determined by the extrapolation of inverse multiplication plots in conjunction with the established rules provided below:

- Fuel additions were limited to one-half the difference between the loaded mass and the extrapolated critical mass or to 50 fuel pins, whichever was less;
- Each fuel pin was placed in a lattice position that preserved, as nearly as possible, a cylindrical core with 1/4 core symmetry;
- The number of fuel pins added in a given step did not exceed that of the previous step;
- Each fuel pin was added in a lattice position such that the minimum surface to volume ratio was achieved;
- After the minimum critical mass was achieved and until the core was loaded to achieve the desired response, fuel additions were limited to four fuel pins per step.

Table I provides relevant information on the initial approach to criticality (the number of fuel pins loaded, detector counts [with all control rods fully withdrawn and the neutron source in its shielded position], and the 1/M points). Figure 3-1 plots the 1/M points versus fuel loading. Appendix A outlines in greater detail the

Table I

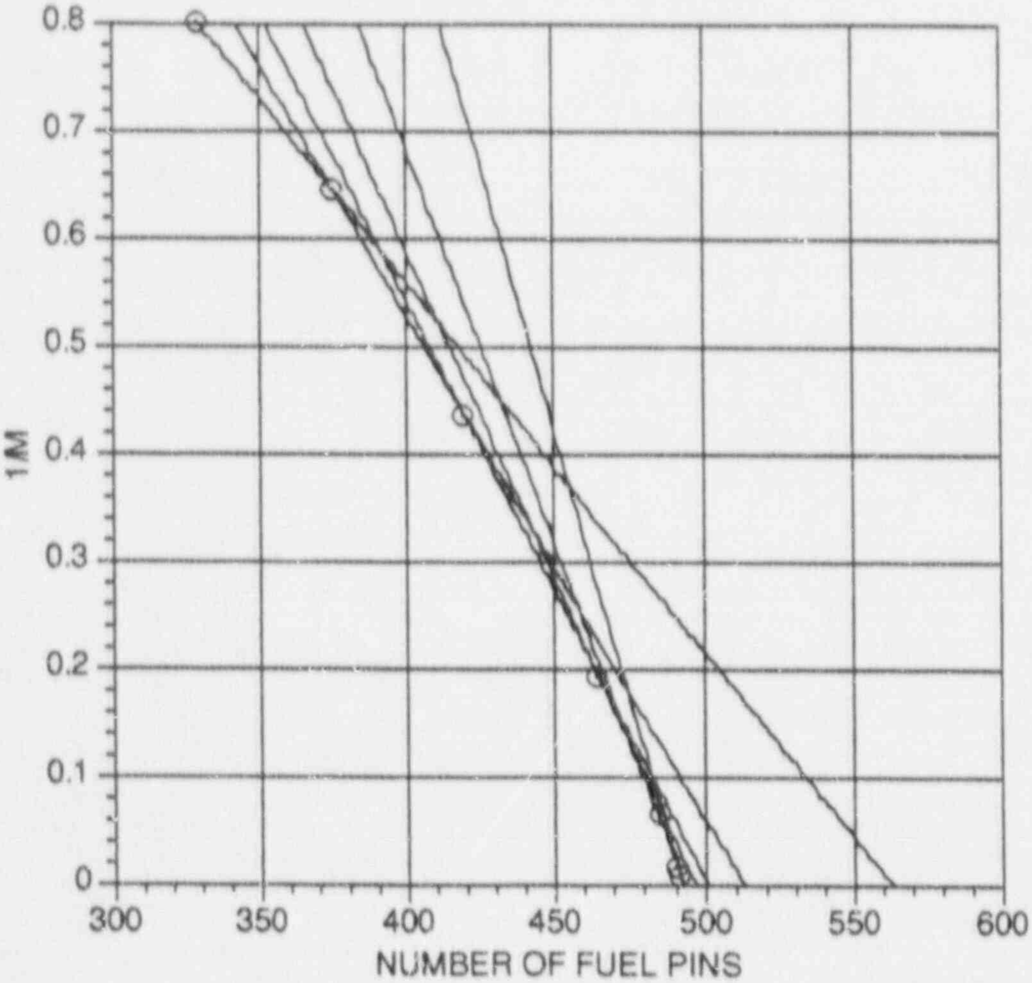
Initial Approach to Criticality

Fuel Load Step	Number of Pins in Core	Counts per 100 seconds	1/M
1	105	25	n/a
2	150	20	1.25
3	195	18	1.39
4	240	18	1.39
5	285	16	1.56
6	330	25	0.80
7	375	31	0.645
8	420	46	0.435
9	465	105	0.192
10	486	311	0.0643
11	491	1356	0.0147
12	492	2704	0.00740
13	493	n/a	n/a

Notes:

1) The reactor was slightly supercritical at 493 pins.
2) Moderator temperature was approximately 60^{0} F.
3) Counts shown above with all control rods fully withdrawn and neutron source in its shielded position (ie., not in core).
4) Initial count changed from 25 to 20 at step 6.

Figure 3-1 Plot of 1/M versus Fuel Loading



methodology used in the inverse multiplication measurement [7,8].

Generally, in an inverse multiplication measurement such as this, the detector counts taken from the initial fuel addition is used in all subsequent 1/M calculations. As can be seen from Table I, the first five loading steps give a 1/M which do not prove useful. This was not all together unexpected because of the low loading and counting statistics. It was then deemed prudent to revise the initial count for 1/M calculations from 25 to 20 (which is still +/- one standard deviation of the first five counts). Figure 3-1 therefore is plotted from step 6 onward.

The plot of criticality (K_{eff}) vs. temperature, for both measurements and calculations (for a given fuel loading), is shown in Figure 3-2. The error between calculated and measured values of K_{eff} is less than 0.5%, which is a commonly accepted degree of margin in the literature.

Measurements of primary interest of the fuel loading are performed with all control rods fully withdrawn from the core to provide clean core measurements. The DIFXYRZ calculations are performed under similar conditions, but the calculations fail to compensate for the worth of the stainless steel control rod guides which would be be fully inserted into the core when the control rods are fully withdrawn.

Measurements have been performed to determine the worth of these control rod guides. These measurements show that the presence of the control rod guides lower the core reactivity relative to that of a truly clean core by 0.35 \$. Therefore the measured values of K_{eff} have been compensated for the worth of the guides, and this is reflected in Figures 3-2.

3.4 Control Rod Worths

Measurements have been performed to determine the worth of individual control rods, and to determine the worth of the control rods as a bank. These measurements show that the typical individual control rod worth is 0.55 \$. Because of core and control rod symmetry, all control rods are of similar worth. The worth of the entire bank is 2.20 \$.

The measurements showed, with a nominal fuel loading of 497 pins (at approximately 68^0 F), that the excess reactivity was 0.20 \$. With a total control rod bank worth of 2.20 \$, and a 0.20 \$ excess, the shutdown margin is 2.00 \$. Under these conditions, and with one control rod fully stuck out of the core, the core would be shutdown by 1.45 \$ (-2.20 - (-0.55) + 0.20).

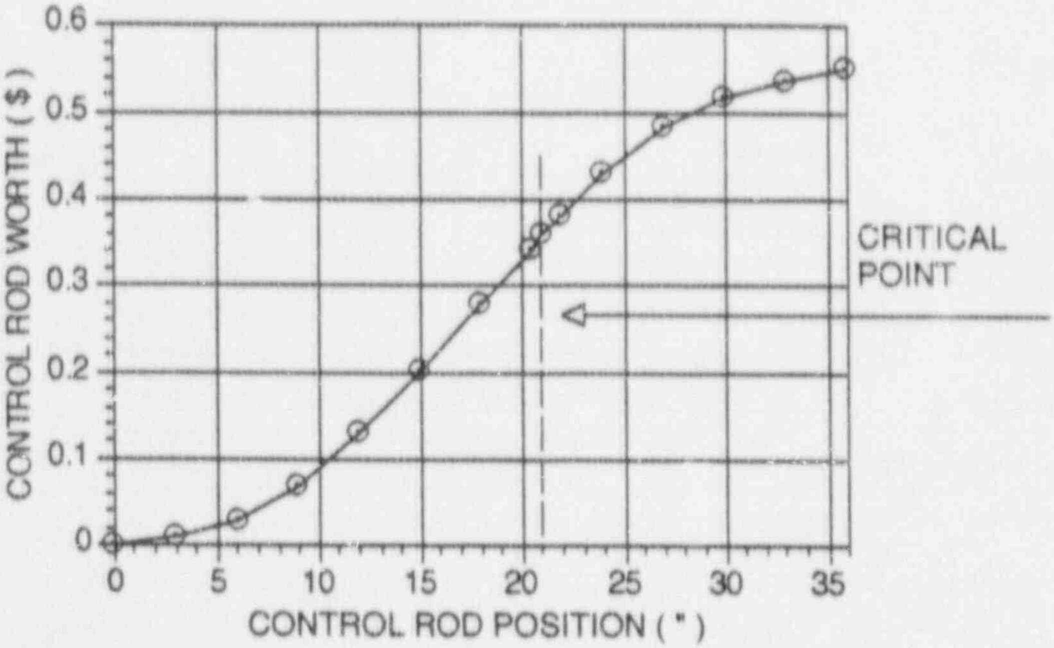
The integral and differential control rod worth curves for a typical control rod and for the entire bank of control rods are shown in Figures 3-3 and 3-4 respectively. The differential rod worth curve is created by differentiating the integral rod worth curve with respect to axial position. A more detailed description of the control rod measurement methodology used is presented in Appendix B.

The measurements of an individual control rod worth are normally performed with the other thr[illegible]ds fully withdrawn. Measurements and calculations have also been performed to measure the effects of control rod shadowing, that is, how does the presence of one control rod, partially inserted into the core, effect the worth of a second control rod. In these measurements, the worth of an individual control rod is determined with a second control rod partially inserted. The measurements show that the effect of control rod shadowing is negligible (<0.02$).

Calculations were performed to determine the effects of rod shadowing. The calculations were done with various control rod inventories in the core (ie., all four rods in, any three rods in, two adjacent rods in, etc.). Table II shows the results of these calculations and that the effects of rod shadowing are minimal as shown by the small variance in single control rod worth among the

FIGURE 3-3: INDIVIDUAL CONTROL ROD WORTH

A. INTEGRAL WORTH CURVE



(standard error approx. 0.5 cents)

B. DIFFERENTIAL WORTH CURVE

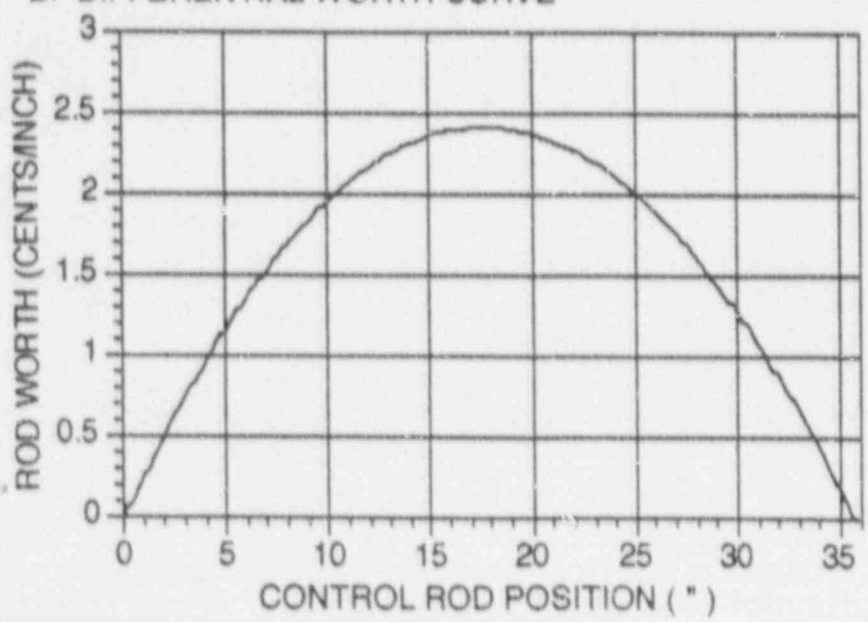
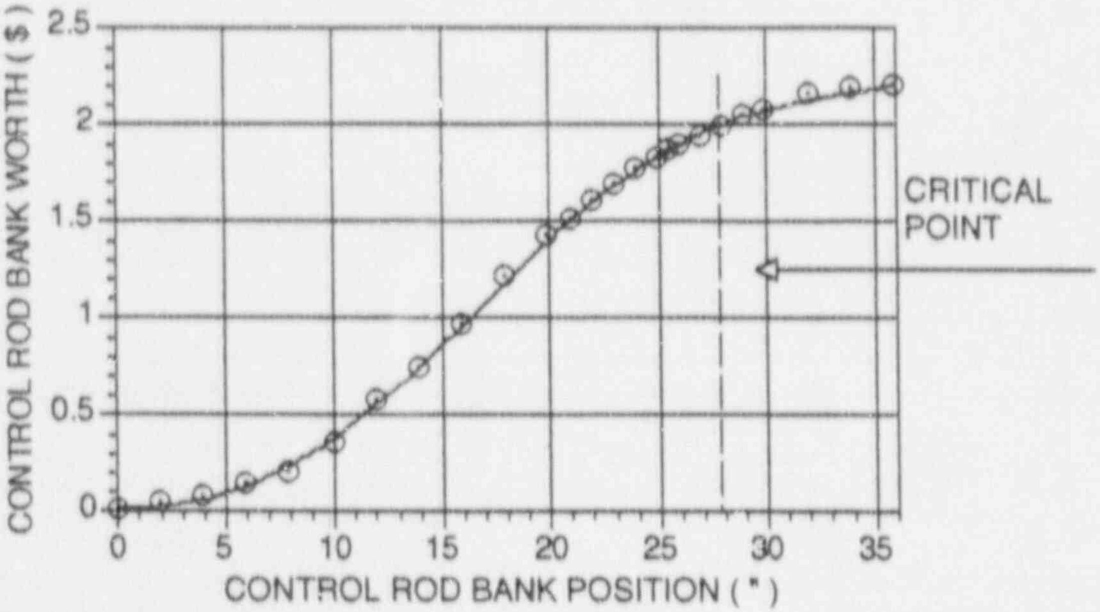


FIGURE 3-4: BANK CONTROL ROD WORTH

A. INTEGRAL BANK WORTH CURVE



(standard error approx. 0.5 cents)

B. DIFFERENTIAL BANK WORTH CURVE

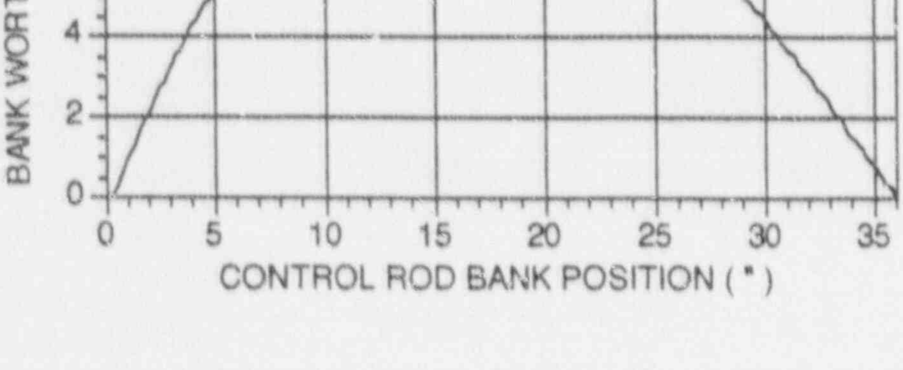


TABLE II

Control Rod Worths and Shadowing Effects

Single Control Rod Worth ($)

Technical Specification limit [1]	<-0.43 $
measured value	-0.55 $
calculated value	-0.52 $

note 1: With the maximum reactivity allowed above cold clean, (0.60 $), and with the minimum allowed shutdown margin, (0.70 $), from three operable control rods and one stuck out of core, then (0.60 + 0.70)/3 = 0.43 $. (Ref T.S. 3.1)

Multiple Control Rod Worths

Calculated value ($) [2]

all four control rods inserted
total rod worth -2.20 $; worth per inserted rod -0.55 $

any three control rods inserted
total rod worth -1.62 $; worth per inserted rod -0.54 $

two opposite-facing control rods inserted
total rod worth -1.07 $; worth per inserted rod -0.54 $

two adjacent-facing control rods inserted
total rod worth -1.05 $; worth per inserted rod -0.53 $

any one control rod inserted
total rod worth -0.52 $; worth per inserted rod -0.52 $

note 2: Normalized to measured all four control rods inserted worth of -2.20 $.

various cases.

The Technical Specifications (TS) does not directly require a minimum worth of a single control rod, nor of the control rods as a bank [9]. TS paragraph 3.1 (Reactor Control and Safety Systems) does require a minimum shutdown margin of 0.70 \$, with the the most reactive control rod fully withdrawn. The same section limits the reactor core excess reactivity to 0.60 \$. Given these factors, it is implied that the minimum control rod worth of an individual control rod is 0.43 \$ ({-0.60 + -0.70}/3 = -0.43 \$). Therefore, the measured control rod worth (-0.55 \$) is greater than the implied worth (-0.43 \$), and thus, the implied Technical Specifications limit is satisfied.

Assuming the reactor had the maximum excess reactivity allowed by TS, (0.60 \$), and with the TS minimum allowed shutdown margin (-0.70 \$) with one control rod stuck out of the core, the worth of the three operable control rods would need to be ≤ -1.30 \$ (-0.60 \$ + -0.70 \$). Given a symmetric core and insignificant rod shadowing effect, the worth of three operable control rods is -1.65 \$. Therefore, the TS requirement on shutdown margin has also been satisfied.

The Technical Specifications sets other limits on control rod worths. One such limit (section 3.1.3) is set on the maximum control rod reactivity insertion rate: <0.12 \$/second up to 10 times source level, and $\leq$ 0.05 \$/second at all higher levels. Control rod travel time, from full in to full out, is approximately 12 minutes.

At this rate, and given the maximum reactivity/inch for a single control rod, the maximum reactivity insertion rate from the withdrawal of a single control rod is 0.0012 \$/second. A similar calculation for withdrawal of all four control rods as a bank yields a maximum reactivity insertion rate of 0.0048 \$/second. These values are well within the TS limits.

A final TS requirement (section 3.1.4) concerns the maximum control rod drop time from fully withdrawn to fully inserted (ie., as during a reactor scram). TS requires a maximum drop time of 900 milliseconds (ms) (which includes a maximum magnet release time of 50 milliseconds). The most recent control rod drop time measurement (performed 2/15/90) showed that each control rod dropped in less than 775 ms , with each magnet release time less than 25 ms. Therefore the this TS requirement has been satisfied.

The Safety Analysis Report has determined that the worst postulated accident is of that of a positive reactivity addition excursion [10]. The analysis of this accident assumes that only 1.00 $ of negative reactivity is added to the core following the excursion (the addition of 0.60 $). The analysis then shows that the resulting increase in fuel temperature is less than 0.1 ^{0} C (from ambient of 20 ^{0} C). This value is magnitudes less than the TS maximum allowed temperature of 2000 ^{0}C. Assuming such an accident could occur, and that one control rod fails to insert, then -1.65 $ (-2.20 -(- 0.55)) would be inserted by the three operable rods. Therefore, the analysis in Safety Analysis Report is conservative.

As a final check on control rod worth, a prompt-drop experiment is performed which measures the shutdown margin with three control rods fully withdrawn and one control rod fully inserted. The control rod to be inserted is placed at the critical position, and then scrammed. The reactor power, before and after the control rod insertion, are related to provide shutdown margin.

Since the full insertion of a single control rod results in a subcritical core, the power will decay away to very low levels on a negative period. The experiment is not very accurate since the exact power due to the the prompt drop (from the rapid control rod insertion) is not well defined because power is rapidly decaying. The results of this experiment shows that the reactor is shutdown by 0.43 $, when the more accurate measurement shows 0.35 $.

3.5 Fuel Pin Worths

Measurements and calculations have been performed to determine the reactivity worth of the fuel pins. Unlike the control rods, whose position relative to the core structure remains stationary, the location of the fuel pins can assume any position allowed by the lattice plates (Refer to Figure 2-4). [This is not to imply that fuel pins are randomly placed within the lattice plate, which has 553 fuel pin locations. A nominal core loading consists of 497 pins, built in a solid, roughly cylindrical shape. Visual verification of core loading is performed prior to any reactor operation.]

The reactivity worth of the fuel pins, and the control rods, is related to the thermal flux in the vicinity of the fuel pin or control rod. A fuel pin placed in an area of high flux (ie., near the core center) would have a greater worth than if the same fuel rod were positioned in an area of lower flux (ie., near the core periphery). Therefore, since the worth of the fuel pins is spatially dependent, measurements and calculations were performed on fuel pins at various locations throughout the core.

Fuel pin worth measurements are rather straightforward. The reactivity of a base case core is determined by the positive period method. Following shutdown, a fuel pin is removed from the core (if located within the core volume) or added to the core (if the pin location is outside the core volume). The reactivity of the perturbed core is then measured, and the difference in reactivity between the perturbed core and the base case core is attributed to the effected pin, and thus, the pin worth. In all pin worth measurements, except two (where noted), the listed reactivity worth was measured by fuel pin removal.

Three different core locations were chosen for the measurements and calculations: 1) along the core radius from the center fuel pin outward toward a control rod (13 pins total); 2) along the core radius from the center fuel pin outward toward the core periphery, along a 45^0 diagonal midway between two control rods (9 pins total); and, 3) nine fuel pins located on the core periphery. Because the core is generally quarter-core symmetric, it is expected that fuel pins symmetrically positioned to those measured would show similar worths, and this has been verified by measurement.

Figure 3-5 shows the locations of the fuel pins measured along both core radii, and Figure 3-6 shows the locations of the fuel pins measured on the core periphery. Table IIIA compares the measured and calculated fuel pin worths of those located along the core radius, from the center pin outwards toward the control rod (at 20^0 C), and this comparison is plotted in the accompanying Figure 3-7.

Table IIIB compares the measured and calculated fuel pin worths for those fuel pins located along the core diagonal (at 20^0 C), and this comparison is plotted in the accompanying Figure 3-8. Table IV compares the measured and calculated fuel pin worths of those located along the core periphery (at 20° C), and this comparison is plotted in the accompanying Figure 3-9. Table V shows the calculated fuel pin worths along both radii at 60° C. A description of the fuel pin measurement methodology used is presented in Appendix C.

The majority of the values listed, for both measured and calculated, show a positive value for fuel pin worth. All fuel pin measurements were performed by measuring the perturbed core reactivity (ie., fuel pin removed from or inserted to the base core), and comparing to the reactivity of the base core. Thus a positive fuel pin worth (measured by pin removal from the base core) implies

Figure 3-5

Core Locations of Radial Fuel Pin Measurements
(1/4 core shown only)

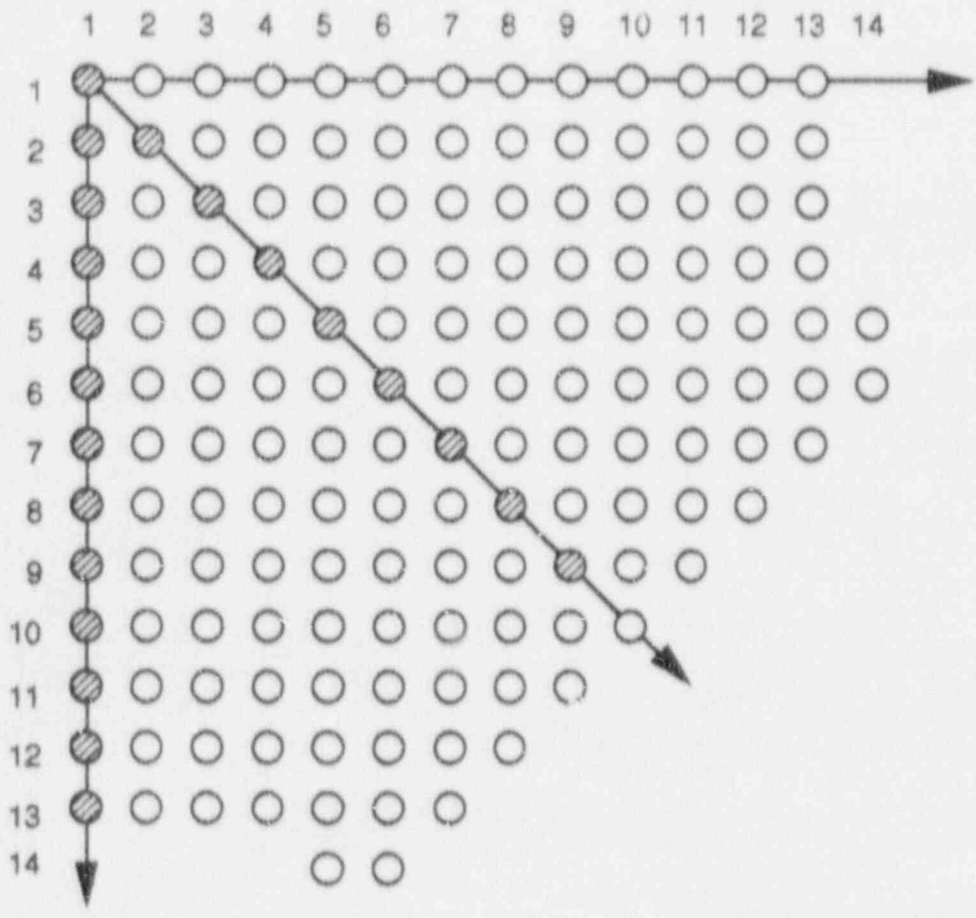


Figure 3-6

Core Locations of Peripheral Fuel Pin Worth Measurements

(1/4 core shown only)

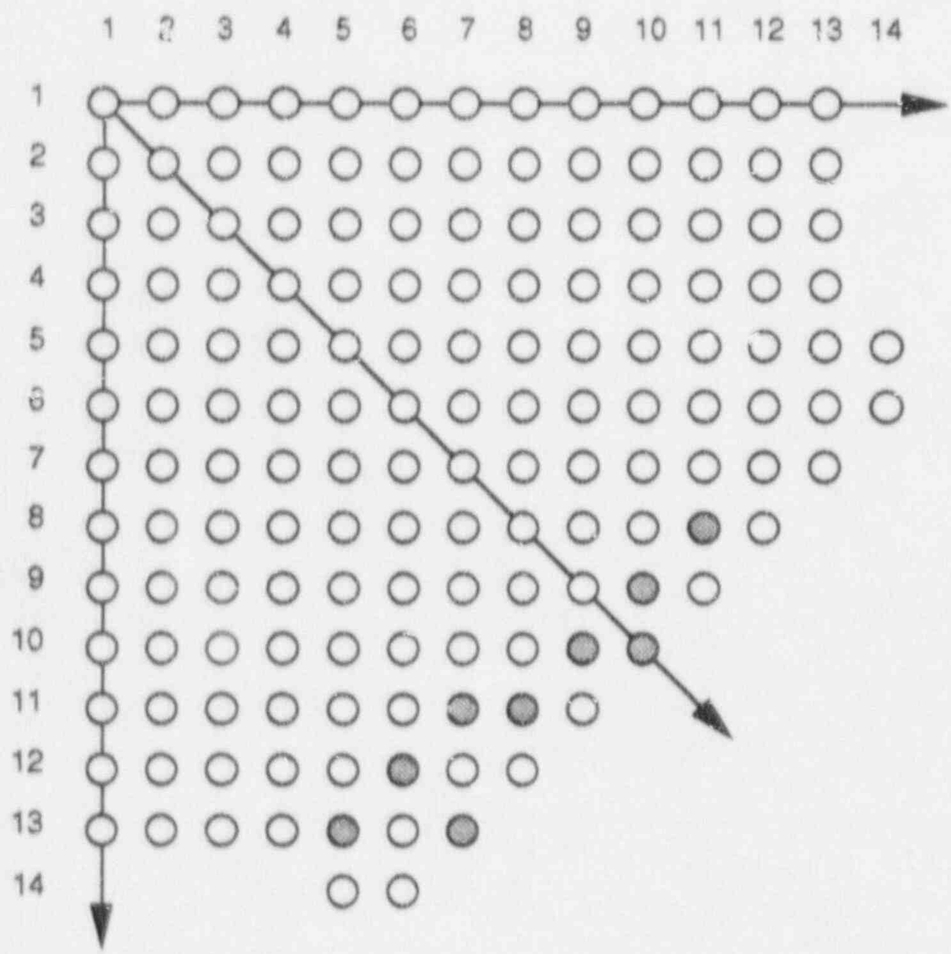


TABLE III

RADIAL PIN WORTHS:MEASURED vs. CALCULATED

A. Pins located from center along core axis toward control rod

pin location	measured (cents)	calculated (cents)
(1,1)	14.9	19.2
(1,2)	14.0	19.0
(1,3)	13.9	18.4
(1,4)	13.3	17.5
(1,5)	12.4	16.4
(1.6)	12.0	15.0
(1,7)	10.2	13.4
(1,8)	9.2	11.7
(1,9)	8.1	9.9
(1,10)	6.3	8.0
(1,11)	4.5	6.0
(1,12)	2.3	3.6
(1,13)	-1.1	- 0.9

B. Pins located from center along core diagonal

pin location	measured (cents)	calculated (cents)
(1,1)	14.9	19.2
(2,2)	14.7	18.8
(3,3)	14.1	17.6
(4,4)	12.9	16.0
(5,5)	10.8	13.9
(6,6)	9.8	11.4
(7,7)	6.5	9.0
(8,8)	2.2	6.2
(9,9)	1.5	1.9

See Figure 3-5 for pin locations.
Measurements at 20^0 C.
(standard error approx. 0.9 cents)

Figure 3-7

Pin Worths: Measured versus Calculated

(from center pin outward toward control rod channel)

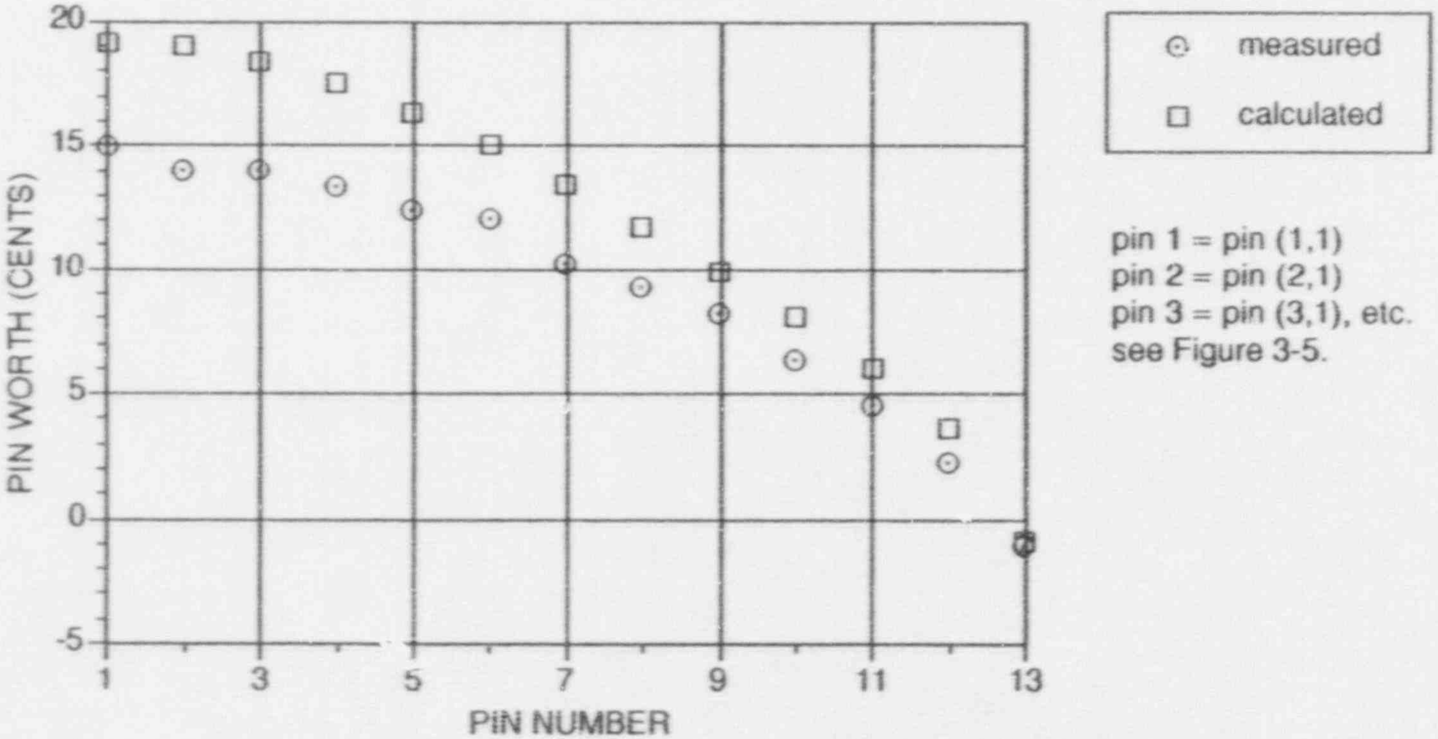


Figure 3-8

Pin Worths: Measured versus Calculated

(from center pin outward along core diagonal)

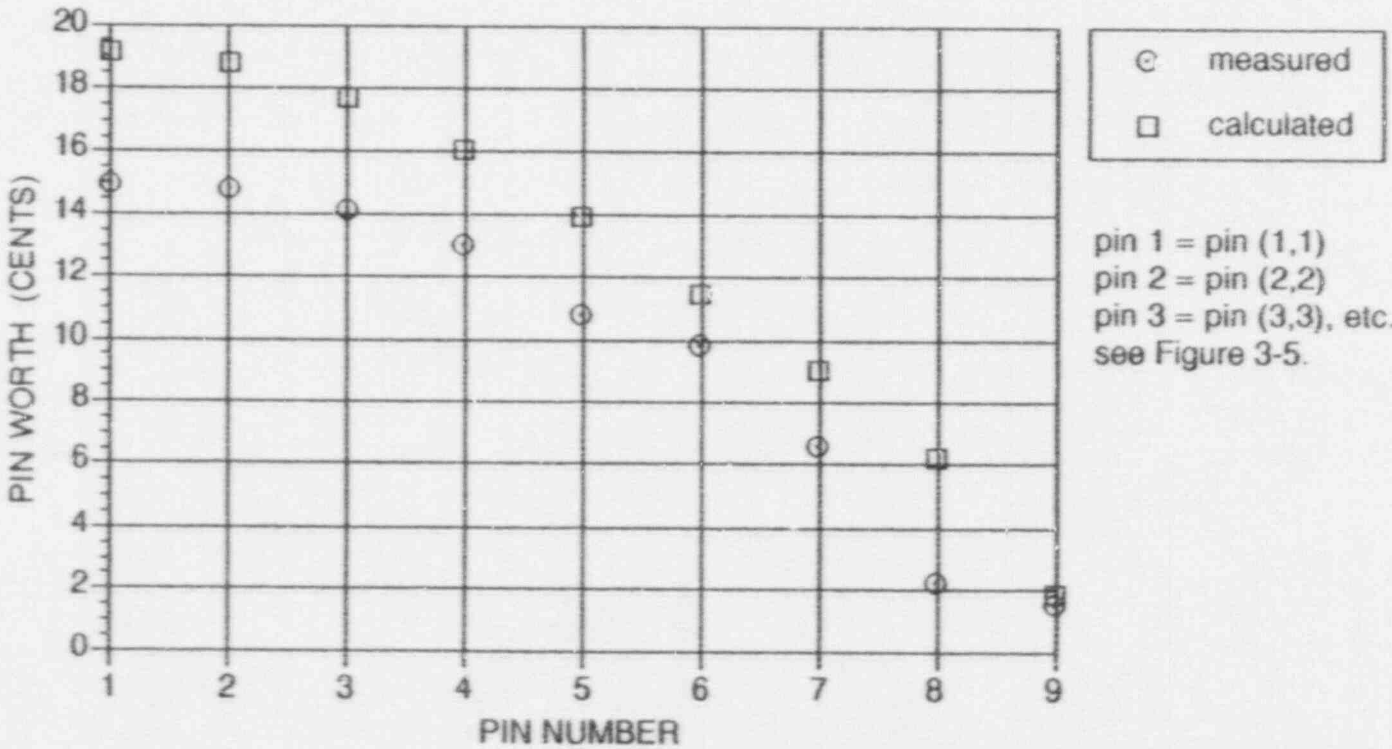


TABLE IV

PERIPHERAL PIN WORTHS: MEASURED vs. CALCULATED
Pins located on core periphery

pin location	measured (cents)	calculated (cents)
(8,11) or (11,8)	-4.5	-4.1
(9,10) or (10,9)	-3.0	-3.0
(5,13)	-4.2	-4.2
(6,12)	-2.8	-2.6
(7,11)	-0.7	-0.1
(7,13) [added]	9.1	9.2
(10,10) [added]	4.7	5.0

See Figure 3-6 for fuel pin locations.
Measurements at 20^0 C.
(standard error approx. 0.9 cents)

Figure 3-9

Pin Worths: Measured versus Calculated

(along core periphery)

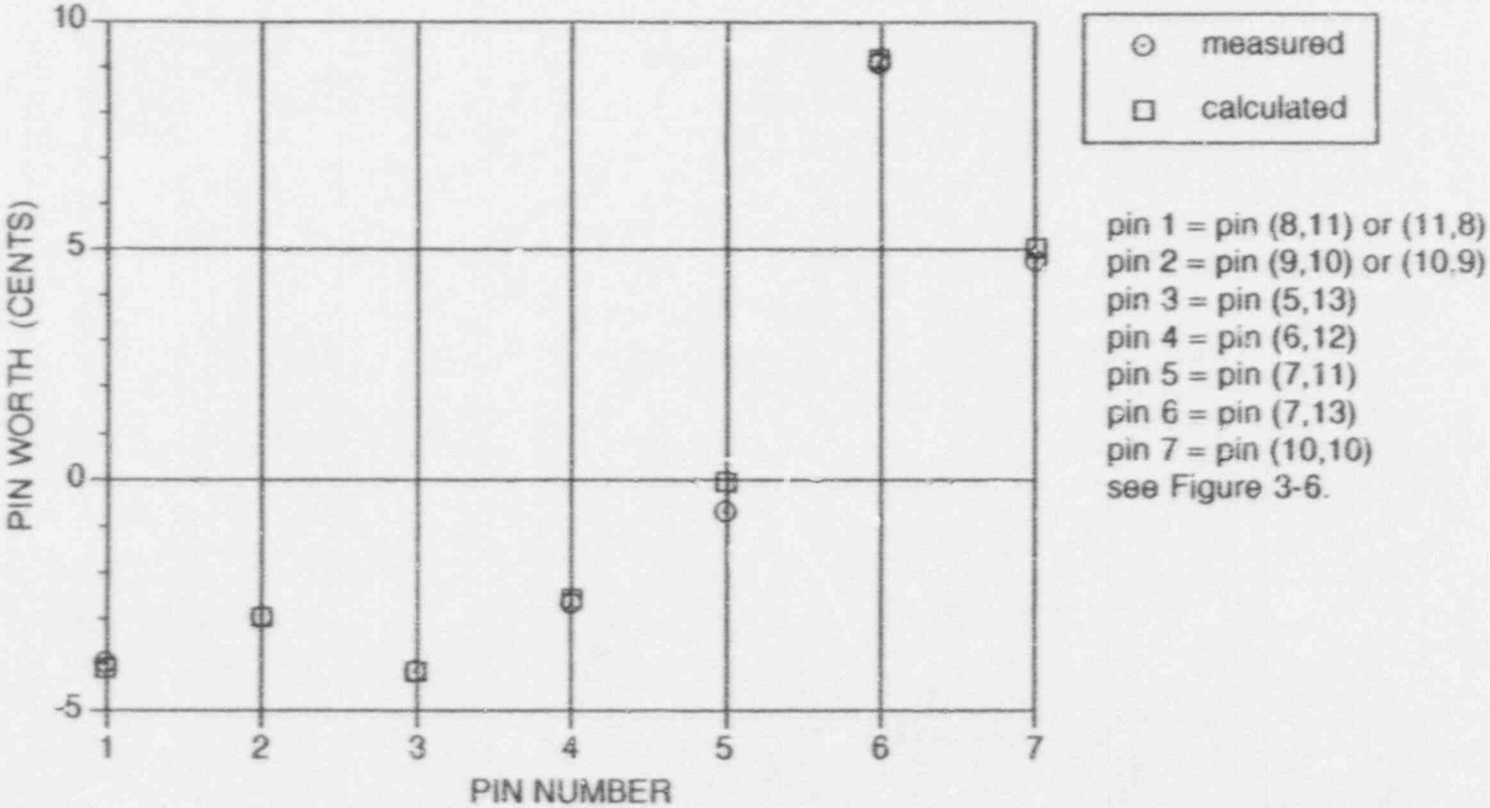


TABLE V

RADIAL PIN WORTHS -@ 60° C

pin location	calculated value (cents)
(1,1)	18.6
(1,2)	18.4
(1,3)	18.3
(1,4)	17.5
(1,5)	16.3
(1,6)	15.0
(1,7)	13.4
(1,8)	11.7
(1,9)	10.0
(1,10)	8.1
(1,11)	6.1
(1,12)	3.6
(1,13)	-1.1
(2,2)	18.1
(3,3)	17.7
(4,4)	16.1
(5,5)	14.0
(6,6)	11.6
(7,7)	9.0
(8,8)	6.1
(9,9)	1.6

See Figure 3-5 for pin locations.

that the core reactivity increased due to the pin removal by the amount shown. A negative worth implies a lower core reactivity due to the pin removal. Similar arguments are used for those pins inserted into the base core.

The removal of an interior fuel pin results in an increase in core reactivity because the spacing between fuel pins (pitch) is small compared to the average neutron diffusion length, and the magnitude is related to the thermal neutron flux in the area of the pin. The removal of an interior fuel pin results in an effect of increased neutron moderation which outweighs the effect of the reduced fissile material, and thus core reactivity increases. A core lattice of greater pitch would result in a smaller change.

Technical Specifications section 3.1, Reactor Control and Safety Systems, states that the maximum reactivity worth of any clean fuel pin shall be 0.20 $. Even though there is some disagreement between the measurements and calculations of interior fuel pin worths, there is no doubt that the center pin is of greatest worth and that it is indeed within the TS limit.

3.6 Moderator Isothermal Temperature Coefficient of Reactivity

Measurements of the overall moderator temperature coefficient of reactivity have been performed from ambient conditions to approximately 140^0 F (60^0 C), while calculations have been performed from the TS minimum allowed operating temperature of 50^0 F (10^0 C) to 176^0 F (80^0 C).

To measure the temperature coefficient, the reactor is taken critical at the ambient moderator temperature, with three control rods fully withdrawn and the fourth control rod positioned at the critical height. Two reactor tank immersion heaters are then energized. After some degree of heatup, the single control rod is repositioned to a new critical height. Given the temperature difference between the two critical conditions and the reactivity added by the control rod motion, the temperature coefficient is calculated. A similar process continues as the heatup continues. A description of the moderator temperature coefficient measurement methodology is presented in Appendix D (method 1 was used in the measurement).

The results of the measurements and calculations are plotted in Figure 3-10. The figure shows that the temperature coefficient becomes increasingly more negative with increasing temperature, and that the measurements and calculations are in good agreement. The general shape of the curve is expected since water density decreases as temperature increases, and therefore the moderating capacity diminishes with temperature.

Technical Specifications 3.2, Reactor Parameters, requires that the temperature coefficient shall be negative at above 100 ^{0} F. The same TS section also requires that the net positive reactivity inserted from the minimum operating temperature (50 ^{0} F) to the temperature at which the coefficient becomes negative shall be less than 0.15 \$. Measurements and calculations indeed show that the temperature coefficient of reactivity is negative at above 100 ^{0} F, and thus this portion of the TS requirement is met.

Even though the measurements showed only negative temperature coefficients, implying that the ambient moderator temperature was greater than that temperature where the temperature coefficient turns from positive to negative, calculations have been performed over this temperature range.

As shown in Figure 3-10, the cross-over point where the coefficient turns from positive to negative is 53.8^0 F (12.1^0 C). Analysis shows that the amount of positive reactivity added over the range from the minimum allowed operating temperature to the cross-over point is 0.001 \$. Since measurements and calculations are in such good agreement over the temperature range of the measurements, it is reasonable to expect that the calculated cross-over point of the temperature coefficient is a reliable estimate of the actual cross-over point. Since the calculated reactivity inserted between the minimum operating temperature and the cross-over point is less than the TS maximum, the TS criteria is satisfied.

3.7 Void Coefficient of Reactivity

Measurements and calculations of the void coefficient of reactivity have been performed. The core reactivity of an unvoided base case core is first measured. Voids are then selectively introduced into the core, and the core reactivity is again measured. Knowing the volume of the void and the difference in reactivity between the perturbed core and the base case core allows for computation of the void reactivity coefficient.

Initial measurements were conducted by positioning a polystyrene foam stringer of known volume and density into different locations of the core. These initial measurements showed a negative coefficient, but the method was less than optimum. The stringer was placed into a loaded core, from above, and the resultant position of the stringer was not accurately known. The lack of this information made calculational modeling very difficult. This method was also less precise since it may not have resembled voids which form and surround the fuel pins.

A second method uses a 36-inch long, thin polystyrene foam sheet which was wrapped around a single fuel pin at various core locations. This allowed resemblance of voiding around the length of a fuel pin, and its position was easily verified. The positioning of the sheet was changed radially through out the core to obtain the core average void coefficient. The difference in reactivity between the voided core and the unvoided base case core allowed computation of the void coefficient. The methodology used to measure the void reactivity coefficient is presented in Appendix E.

The measurements showed that the void coefficient was negative at approximat[illegible] F, and that the core volume average void coefficient was $-0.00113\ \$/\mathrm{cm}^3$ of void. The calculations also showed a negative void coefficient at this temperature, with a core volume average void coefficient of $-0.00115\$/\mathrm{cm}^3$ of void.

Technical Specifications section 3.2, Reactor Parameters, requires a negative void coefficient above 100^0 F, and have a minimum average negative value of $-0.00043\$/\mathrm{cm}^3$ of void. Calculations performed at 104^0 F provide an average void coefficient of $-0.00116\$/\mathrm{cm}^3$ of void. Because there is very little change in the calculated void coefficient between 72 and 104^0 F, (but in the conservative direction) it is expected that the measured value, over this temperature range, would change very little also.

Therefore, the expected measured average void coefficient at 104^0 F is approximately -0.00113 \$/cm^3 of void, which is very conservative to the TS limit of -0.00043\$/cm^3 of void.

3.8 Relative Power Shape

Measurements have been made to determine the relative axial flux distributions of the core. Following reactor operation for a short period, fuel pins are removed from the core and counted in a NaI scintillation counting system.

This counting system detects the decay gammas from the decaying fission products in the fuel. The number of decay gammas is directly related to the number of fission products, which is directly related to the number of fissions. The number of fissions is directly related to the number of thermal neutrons which cause the fissions. Therefore, the decay gammas are directly related to the thermal neutrons.

After counting at various axial locations along the pins, and correcting for background and radioactive decay, the corrected counts are normalized and plotted as a function of axial location.

The plot of the normalized axial flux versus axial position is presented in Figure 3-11. The plot shows the expected cosine shape as a function of axial position.

Measurements have also been performed to determine the relative radial flux distribution. Following irradiation for a short time, pins along a core radius (from the center pin outward towards a control rod) were removed and counted, at the same axial location along the pins. After correcting for background and radioactive decay, the counts are normalized and plotted as a function of radial position. Figure 3-12 shows the normalized radial flux distribution for pins located along the core axis. The radial distribution follows the expected shape of a first-order bessel function.

Radial measurements were also made along the core diagonal, and the distribution showed the same shape as that along the core axis. A description of the relative flux distribution measurement methodology used is presented in Appendix F.

3.9 Absolute Power Calibration

Absolute power measurements are performed to verify that actual core power is reflected in the power indicators and recorders on the main control panel.

Bare gold foils, and gold foils completely covered with cadmium, are placed axially along the center fuel pin. The fuel pin is then reinserted into its center core position and the reactor is operated for a known length of time at a known power. The reactor is then shutdown, and following a short waiting period, the foils are removed and counted in an appropriate counting system.

The counting of the bare gold foils provides information on both the fast and thermal flux, while the cadmium covered foils provide information on the fast flux only since the cadmium absorbs all thermal neutrons. The difference between the counts provides the thermal flux only, which is the item of interest.

Once the axial distribution of thermal flux on the center pin is known, the core average thermal flux is determined since the radial flux distributions is also known (described previously). Calculation of the average core power can then be determined given the macroscopic fission cross-section of U^{235}, the core volume and the average energy released per fission. This measured power is then used as a calibration point for the reactor power instrumentation.

The measurement of absolute core power (last measured in 4/90) provided a core average power of 0.068 watts, while the recorded average core power over the measurement period was

0.071 watts. From this, it can be seen that the recorded power is conservatively set by about 5% above true power. The scram setpoints, which are driven off the power monitors, are thus also conservatively set by about 5%. This conservatism helps to ensure that the core power remains within the licensed limits.

Part 5
Discussion and Conclusions

The results of the measurements contained within this report, along with applicable Technical Specification limits, are presented in Table VI. The Table shows that each Technical Specification related item has been satisfied. Also, this report has shown that operation of the facility satisfies the accident analysis contained in the Safety Analysis Report.

Operation of the facility is guided by procedures approved by the Nuclear Safety and Review Board (NSRB). A licensed senior reactor operator is always present at the controls when the react. is in operation, and all fuel movements are performed under the direction of a licensed senior reactor operator.

The primary use of the facility is for instructional purposes for RPI students, and others not affiliated with the university. Experiments performed by students, under the guidance of a reactor operator, and generally accompanied by the Facility Supervisor, are very similar in method to the types of measurements discussed in this paper. This is done for several reasons.

Table VI

Measurements and Technical Specification Limits

	LEU Core	Tech. Specs.
Excess Reactivity at 68^0 F	0.20 $[1]	< 0.60 $
Subcritical Reactivity with One Rod Stuck	- 1.45 $	<- 0.70 $
Shutdown Margin	- 2.00 $	< -1.00 $
Core Average Isothermal Temperature Coefficient	< 0 for T > 65^0 F	< 0 for T > 100^0 F
Core Average Void Coefficient of Reactivity	-0.0011 $/cc @ 72^0 F	< -0.00043$/cc for T > 100^0 F
Integrated Reactivity due to Temperature Change, 50^0 F to Temperature at which a_T=0	< 0.01 $	< 0.15 $
Reactivity Worth of Standard Fuel Pin	< 0.15 $[2]	0.20 $

1) At nominal fuel loading of 497 pins.

2) Most reactive fuel pin.

First, the methods are well known and understood, as are the expected results. Secondly, and more importantly, is that the performance of repetitive experiments can provide assurance that the reactor and associated components have not undergone any degree of unnoticed degradation For instance, TS requires that the control rod worth be determined only following major changes in core configuration. Measuring the same control rod worth (during a student laboratory) in an identical core configuration over a many year period, essentially ensures that no internal degradation to the control rod has occurred.

It is felt that the operating characteristics of the RPI Reactor Critical Facility are well understood, as demonstrated through the many measurements and calculations, such that the history of satisfactory operation of the facility will continue in the future.

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APPENDIX A

Method for Determining Critical Fuel Loading

A. Discussion

This procedure is intended to provide guidance on determining the critical mass of the reactor, at the ambient coolant temperature, using the inverse multiplication method. After the initial fuel load is made, and with all control rods full out of the core, timed counts on the startup channels are made. This represents the base counts (C_0). After the second fuel loading is complete, and with all control rods full out, timed counts are again made on the startup channels (C_1). The ratio of C_0/C_1 represents the first inverse multiplication point to plotted as a function of fuel loading.

After the third loading, and with all control rods full out, timed counts are taken (C_2). The ratio of C_0/C_2 is plotted against fuel loading. A line is drawn between these two points and extrapolated to where the ratio equals zero. The fuel loading at this extrapolated point represents the estimate of critical loading. Using this point estimate of critical loading and the rules for loading (mentioned below), load the fourth fuel load step, take timed counts, calculate C_0/C_3, graph, extrapolate, and find the new estimated critical loading. Continued in a similar fashion until critical conditions are met.

B. Notes and Cautions

Calculations should have been performed (ie., DIFXYRZ) which can provide some estimate of critical loading, but it is not to be used as the expected point of criticality. Therefore, criticality should be anticipated following each loading step.

Since two startup channels are available to provide data for the inverse multiplication plot, both sets of data should be plotted to provide two extrapolated estimates of critical loading. It is highly recommended that the channel which provides the most conservative estimate of critical loading (ie., the one which gives the smaller number of fuel pins) be used in calculating the next fuel load step.

The following restrictions are placed on fuel loading:

1. Do not add more than 1/2 of the difference between the present loading and the extrapolated critical mass or to 50 pins, whichever is less (except for the initial step);

2. Place each fuel pin in a lattice position that preserves, as nearly as possible, a cylindrical core with 1/4 core symmetry;

3. The number of fuel pins added in a given step should not exceed that of the previous step; and,

4. Once the critical loading has been achieved and until the core is loaded to achieve the desired effect, limit fuel additions to four pins per step.

All fuel movements are to be supervised by the reactor operator. Fuel movements into or out of the reactor shall be performed with all control rods fully inserted.

To ensure accurate counting, counts must be allowed to stabilize following the movement of control rods to full out. As criticality is approached, the time to stabilize gets longer and longer.

C. Auxiliary Equipment

1. Graph paper and straight edge.

D. Procedure
(Note: It is assumed that the reactor pre-startup checks have been successfully completed.)

1. With all control rods inserted, load an initial number of fuel pins into the core and record fuel placement. Make and record timed counts (ie. 100 seconds) from both startup channels Withdraw control rods until all are full

out of the core. Make and record timed counts from both startup channels with all rods out (These counts represent $C_0(A)$ from channel A, and $C_0(B)$. from channel B.). Record moderator temperature and fuel loaded. Insert control rods fully.

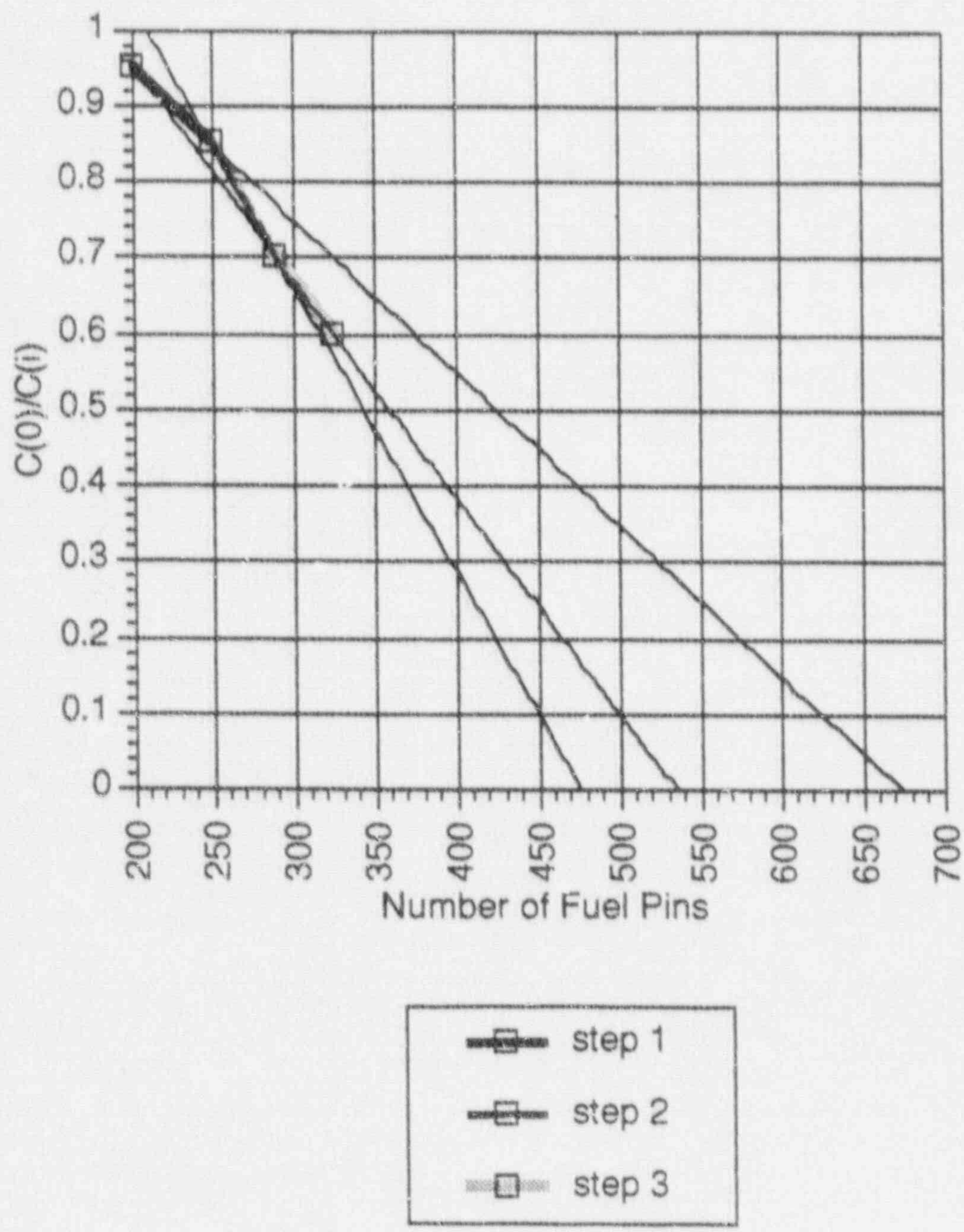
2. With all control rods full in, load additional pins into the core and record fuel loading. Make and record timed counts (ie. 100 seconds) from both startup channels. Withdraw control rods until all are full out of the core. Make and record timed counts from both startup channels with all rods out (These counts represent $C_1(A)$ from channel A, and $C_1(B)$. from channel B.). Record moderator temperature. Calculate the ratios $C_0(A)/C_1(A)$ and $C_0(B)/C_1(B)$. Plot these ratios on the vertical axis, as a function of total fuel loading on the horizontal axis. Insert control rods fully.

3. With all control rods full in, load additional pins into the core and record fuel loading. Make and record timed counts (ie. 100 seconds) from both startup channels. Withdraw control rods until all are full out of

the core. Make and record timed counts from both startup channels with all rods out (These counts represent $C_2(A)$ from channel A, and $C_2(B)$. from channel B.). Record moderator temperature. Calculate the ratios $C_0(A)/C_2(A)$ and $C_0(B)/C_2(B)$. Plot these ratios as a function of total fuel loading. Using these two points, (and always using the two most recent points), linearly extrapolate down to the horizontal axis (where the ratio equals zero). The intersection of this line and the fuel loading axis gives the estimated critical loading.

4. Using the estimated critical loading found above and the loading restrictions, determine the next loading step.
5. With all control rods full in, add the calculated number of fuel pins to the core. Continue the process in a similar fashion, calculating C_0/C_i and plotting for each step, and extrapolating down to zero to find the next estimated critical loading. A sample critical loading plot is shown in Figure 1.
6. The final loading step should produce a critical, or just slightly super critical reactor with all rods out. Record the final loading, core configuration and temperature.

FIGURE 1
Sample inverse Multiplication Plot



APPENDIX B

Method for Determining Control Rod Worth

A. Discussion

This procedure is intended to provide guidance on determining the worth of the control rods, both individually and as a bank, utilizing subcritical measurements combined with supercritical measurements and the positive period method. Because core reactivity cannot be determined when the reactor is subcritical, inferences must be made from supercritical measurements, since core reactivity can be determined at that point. The inferences come in the form of slope matching.

A plot of integral control rod worth shows that the curve is continuous, ie., there are no breaks. Using the positive period method in the supercritical range gives the portion of the rod curve above criticality. Therefore, since the curve is continuous, the slope of the curve just past criticality should equal the slope of the curve just before criticality. Given the neutron counts below criticality as a function of control rod travel, coupled with the curve slope just past criticality, the entire rod worth curve can be generated.

As a check on the measurements, shutdown margin can be roughly approximated by a prompt drop experiment.

B. Notes and Cautions

Depending on the size of the steps of the control rod withdrawal prior to criticality, smaller steps may be recommended as criticality is approached.

C. Auxiliary Equipment

1. Stopwatch, wristwatch or equivalent. (recommended)
2. Inhour curves.
3. Graph paper.

D. Procedure

(Note: It is assumed that the reactor pre-startup checks have been successfully completed. The following is for a single control rod (#4); for bank measurements, insert "rod bank" in place of "rod #4", and start at step 3 with all rods fully inserted.)

1. Record the initial fuel loading and placement.
2. Withdraw all rods to full out except #4.
3. With rod #4 full in, make and record timed counts (ie., 100

seconds) on both startup channels. Record moderator temperature and control rod #4 position.

4. Raise rod #4 a predetermined amount (2 or 3 inches).
5. With rod #4 at the desired position, make and record timed counts on both startup channels. Record moderator temperature and control rod #4 position.
6. Continue raising the rod in steps, and making timed counts as a function of control rod position.
7. Note and record the critical rod position. Record moderator temperature.
8. Raise the rod out a predetermined amount. Measure and record the doubling time and calculate and record reactor period. Calculate and record reactivity from the inhour curves. Record moderator temperature and control rod position.
9. Continue in a similar fashion until the control rod is full out.
10. Calculate the slope of reactivity per inch from the supercritical data, closest to the critical position. For each count from startup channel A (B), assume some arbitrary constant and divide this constant by each count.

From these figures, calculate the slope (reactivity per inch) from the counts closest to criticality. Calculate a correction factor which would equate the supercritical slope to the subcritical slope, and correct the subcritical figures by this correction. The end result will be control rod worth as a function of position.

11. Graph the results from above as a function of rod position. This is the integral rod worth curve. From this curve, differentiate with respect to position to determine the differential rod worth curve. Plot the differential rod worth curve. Determine shutdown margin and excess reactivity. Figures 1A and B show sample rod worth curves.
12. For the prompt drop measurement, insert control rod #4 to the critical position. Record reactor power (P_0). Scram the control rod, and record the final power (P_1) {at the start of the asymptotic slope}. Determine the shutdown margin from {($(P_0/P_1) - 1$).

FIGURE 1A: INTEGRAL ROD WORTH

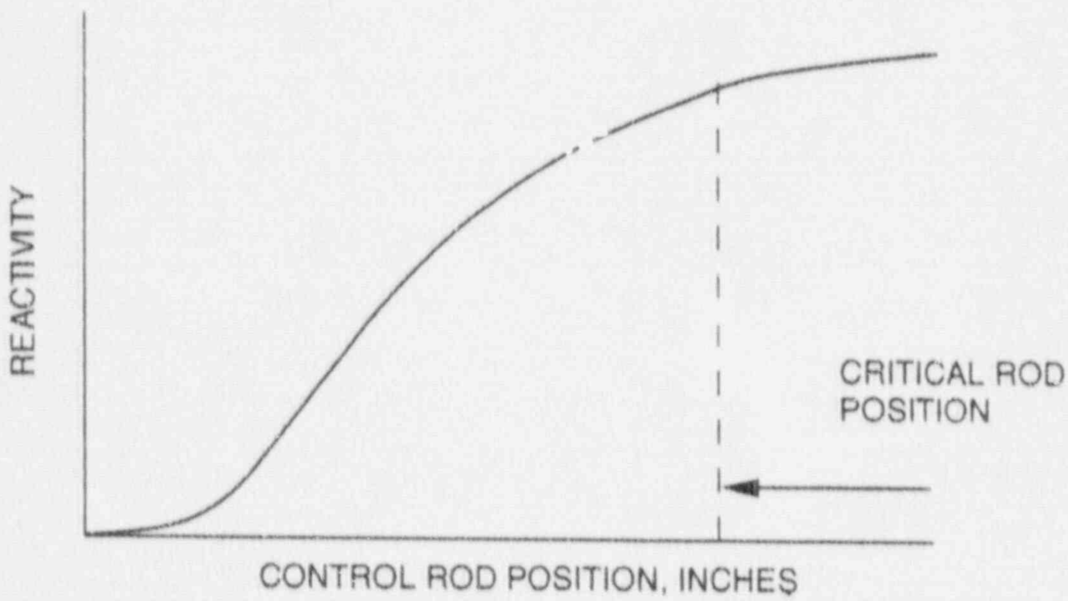
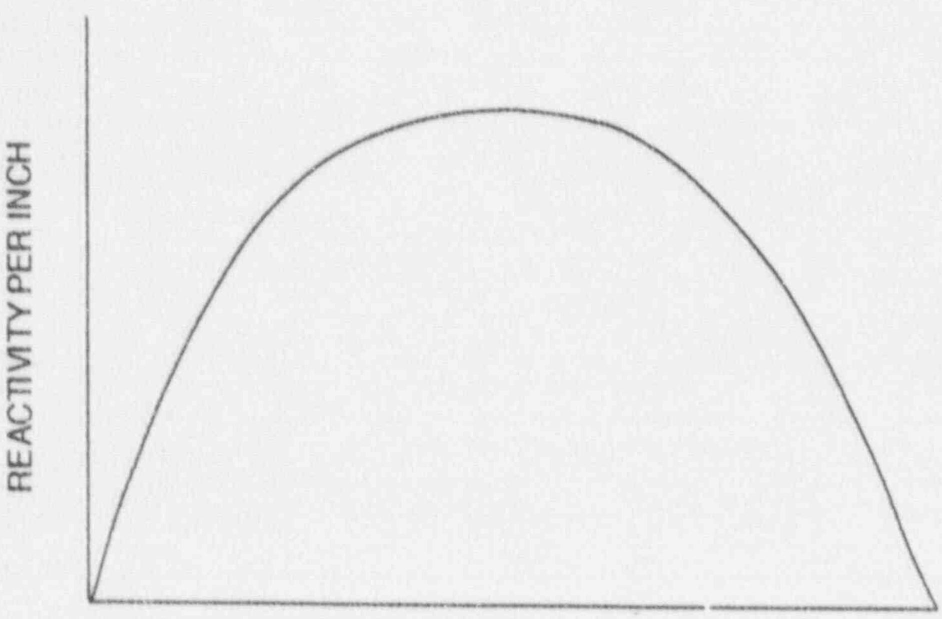


FIGURE 1B: DIFFERENTIAL ROD WORTH



APPENDIX C

Method for Determining Fuel Pin Worth

A. Discussion

This procedure is intended to provide guidance on measuring the worth of individual fuel pins, although the same technique can be applied to groups of pins. The core reactivity of a known core configuration is measured using the positive period method. This measurement represents the base case against which all other measurements will be compared. Following this measurement, and with the reactor shutdown, the fuel pin to be measured is removed from, or added to, the core. The core reactivity is again measured. The difference in the two reactivities can be attributed to the absence or placement of the selected fuel pin, and is equal to the worth of the fuel pin.

B. Notes and Cautions

For the most accurate measurements, it is recommended that all core reactivity measurements be done with all the control rods fully out of the core. This removes any effects from control rods on the measurements. Depending on the initial core configuration (ie., loading), this may not always be possible. Note that generally, removing a fuel pin from the core interior results in a higher core reactivity, whereas removing a pin on the core periphery may result in reduced reactivity. Likewise, inserting a fuel pin on the core interior results in lower reactivity and adding a pin on the exterior increases reactivity.

If core reactivity measurements are to be made with some control rod inventory in the core (ie., not all control rods are full out), the exact positions of control rods need to be known and recorded as part of the measurement. To help ensure exact rod placement between measurements, it is recommended that the rods be driven in manually during the shutdown procedure instead of scramming the rods. Also, make the very last control rod movements in the same direction (in) prior to taking the measurement.

The placement of fuel into or out of the reactor shall be done with all control rods fully inserted and shall be conducted under the direction of a licensed senior operator. When not in use, fuel shall be stored in the fuel storage vault.

C. Auxiliary Equipment

1. Stopwatch, wristwatch or equivalent. (recommended)

2. Inhour curves.

D. Procedure

(Note: It is assumed that the reactor pre-startup checks have been successfully completed.)

1. Record the initial fuel loading and plac[illegible]
2. Perform a reactor startup, and establish a reasonable positive period. Record control rod positions. Measure and record the doubling-time, and comp[illegible]te and record reactor period. Calculate and record core reactivity from the inhour curves. Record reactor water temperature, and the location of the neutron source (either in-core or out).
3. Insert all control rods.
4. With all control rods fully inserted, remove the pin of interest and place in the fuel vault., or insert the pin of interest into the core.
5. Perform a reactor startup, and establish a reasonable positive period. Record control rod positions. Measure and record the doubling-time, and compute and record reactor period. Calculate and record core reactivity from the inhour curves. Record reactor water temperature and source placement.

6. Insert all control rods.
7. If other pin measurements are to be made, repeat steps 2-6 above, remembering to replace the prior measured pin if it was removed.
8. Calculate the difference in the base case core reactivity and the core reactivity with a pin removed or inserted, compensating for any difference in control rod inventory and changes in reactor water temperatures.

APPENDIX D

Method for Determining the Moderator Temperature Coefficient of Reactivity

A. Discussion

This procedure is intended to provide guidance on determining the moderator temperature coefficient of reactivity. There are two ways to perform the measurement. The first method begins with a critical reactor at some temperature. As the moderator heats up, negative reactivity is added and the reactor goes subcritical. A control rod is then pulled to establish slightly supercritical conditions. The heatup continues until the reactor returns to the point of criticality. The worth of the control rod removed from the core can be related to the difference in temperature between the two critical points to determine the temperature coefficient..

The second method utilizes the positive period method. The reactor is brought up to a stable positive period and moderator temperature, and core reactivity is calculated from this period at the corresponding temperature. At a later time after some degree of heatup, and at the same control rod position, the period is again measured to calculate core reactivity (at the new corresponding temperature). The difference in core reactivity divided by the change in moderator temperatures gives the moderator temperature coefficient. The process is repeated for the next step.

B. Notes and Cautions

When the measurement is complete, ensure the immersion heaters are off.

C. Auxiliary Equipment

1. Calibrated control rod curve (method 1).
2. Wristwatch (or equivalent).
3. Inhour curves (method 2).

D. Procedure
(Note: It is assumed that the reactor pre-startup checks have been successfully performed.

D1. Method 1

1. Record fuel loading and placement, and initial moderator temperature readings from all applicable thermocouples. Fully withdraw three control rods, leaving the calibrated control rod full in.
2. Withdraw the calibrated control rod until the reactor goes critical. Record control rod position, moderator

temperature readings.

3. Energize reactor tank heaters and propeller-type mixer, if desired. Record time. Continue recording moderator temperature readings and time every few minutes.
4. As the moderator heats up, the reactor should turn subcritical due to the added negative reactivity. After power has turned, withdraw the calibrated control rod to achieve slightly supercritical conditions. Record control rod position. Continue recording moderator temperature readings and time every few minutes.
5. As the moderator continues to heat up, the reactor should turn from supercritical to critical, and on to subcritical due to the added negative reactivity. The point of criticality is known as the "turning point". At this point, and at all further turning points, record moderator temperature readings.
6. After power has turned downward through criticality, withdraw the calibrated control rod to achieve slightly supercritical conditions. Record control rod position. Continue recording moderator temperature readings and time every few minutes.

7. At the next turning point, record moderator temperature readings.

8. Continue in a similar fashion until the moderator temperature has reached the desired point.

9. By knowing the temperature difference and reactivity added from the control rod withdrawal from turning point to turning point, calculate the temperature coefficient between each consecutive set of turning points. Each coefficient is valid at the average of the two turning point temperatures. Determine heatup rate.

D2. Method 2

1. Record fuel loading and placement, and initial moderator temperature readings from all applicable thermocouples. Fully withdraw three control rods, leaving one control rod full in.

2. Withdraw the fourth control rod until the reactor goes supercritical and establish a reasonable positive period. Measure and record reactor period. Record control rod position, moderator temperature readings, and core reactivity calculated from the inhour curves.

3. Energize reactor tank heaters and propeller-type mixer, if desired. Record time. Continue recording moderator temperature readings and time every few minutes.
4. After some degree of heatup, measure and record reactor period, core reactivity calculated from the inhour curves, and moderator temperatures. Continue recording moderator temperature readings and time every few minutes.
5. After some additional degree of heatup, measure and record reactor period, core reactivity calculated from the inhour curves, and moderator temperatures. Continue recording moderator temperature readings and time every few minutes.
6. Continue in a similar fashion until the present control rod position can no longer sustain supercriticality. At this point, withdraw the control rod to a new position and establish a reasonable positive period. Measure and record reactor period, core reactivity calculated from the inhour curves, and moderator temperatures. Continue recording moderator temperature readings and time every few minutes.

7. Continue in a similar fashion until the desired moderator temperature is reached.

8. By knowing the difference in core reactivity and moderator temperatures between consecutive sets of readings at a constant control rod position, calculate the temperature coefficient. This coefficient is valid at the average temperature between the two readings. Calculate the heatup rate.

APPENDIX E

Method for Determining the Void Coefficient of Reactivity

A. Discussion

This procedure is intended to provide guidance on determining the core void coefficient of reactivity. The reactivity of a known clean (unvoided) core configuration is first measured by the positive period method. A polystyrene foam sheet is then wrapped around a fuel pin located at some radial position in the core. The core reactivity is again determined. Knowing the volume of the sheet, and the difference in reactivity between the voided and unvoided cores allows computation of the void coefficient.

A similar process is repeated for each of several radial locations of the polystyrene sheet placement. The void coefficients are then averaged over the volume of the core to provide the core average void coefficient of reactivity.

B. Notes and Cautions

This experiment requires core disassembly and reassembly several times to allow access to the fuel pin to be wrapped with the foam sheet, since a wrapped fuel pin cannot be inserted into the core from above. Care must be taken when handling the fuel, especially following irradiation. Monitor core activity during this process. Also, try to limit core power to the lowest levels possible to minimize radiation dose during fuel handling. All fuel handling shall be supervised by a licensed reactor operator. When wrapping and unwrapping the fuel pin in place, be careful not to rip the foam since the same foam sheet piece will be used for all measurements. If the sheet becomes damaged between measurements and a replacement sheet used, calculations will need to be corrected for any change in volume between the sheets.

C. Auxiliary Equipment

1. 36-inch long, thin polystyrene foam sheet(s), (of known density and volume), cut to fit around the outside of a fuel pin.
2. Inhour curves.
3. Stopwatch, wristwatch or equivalent (recommended).
4. Tape.

D. Procedure

(Note: It is assumed that the reactor pre-startup checks have been successfully completed.)

1. Note and record fuel loading and configuration. Record moderator temperature. Perform a reactor startup. Measure and record the core reactivity using the positive period method, with all rods out (if possible) and the source out. Record control rod positions at time of measurement. Shutdown the reactor.
2. After waiting for core reactivity to decay to a low level, disassemble the core to allow insertion of the polystyrene sheet. Wrap the selected pin with the sheet and tape. Note and record radial position of the sheet. It is desired to measure the void effect along a core radius of pins (from the center pin outward to an edge pin), but the sequential measurements along the radius may take any order. If the core loading is changed, record the new core loading and fuel placement.
3. After core reassembly has been verified, perform a rector startup and measure and record the core reactivity using the positive period method, with all control rods (if possible) out and the source out.. Record control rod positions at the time of measurement and moderator temperature. Shutdown the reactor.

4. After waiting for core reactivity to decay to a low level, disassemble the core to allow insertion of the polystyrene sheet to a different fuel pin. Note and record radial position of the sheet, and core loading and fuel placement.

5. After core reassembly has been verified, perform a rector startup, and measure and record the core reactivity using the positive period method, with all control rods out (if possible) and the source out. Record control rod positions at the time of measurement and moderator temperature. Shutdown the reactor.

6. Repeat the above procedure at all desired fuel pin locations. Calculate the core average reactivity due to the void. Given the reactivity of the unvoided core, determine the average change in reactivity per unit volume of void,ie.,the average core void coefficient (cents/cm^3 of void). (Be sure to include any reactivity effects from control rods if measurements were not taken at the same control rod position for all measurements, or if fuel additions or deletions from the base core configuration were required during void

measurements.) Shutdown and secure the reactor. Be sure to remove the void sheet from the core before any other operations occur.

APPENDIX F

Method for Determining Relative Power Shape

A. Discussion

This procedure is intended to provide guidance on determining the relative axial power, although the same principle applies to determining radial variations in power. The reactor is started up and run for a short period, and then shutdown. After a brief waiting period, selected pins are removed from the core and counted on a radioactive particle counting system. The counts are corrected for background,and decay, and then fitted to a cosine function (for axial measurements).

B. Notes and Cautions

Because the fuel pins are most radioactive directly following irradiation, the following are recommended to minimize personnel exposure:

1. Determine irradiation time such that counting statistics are good, but dead time losses are minimal, but at the lowest possible power;

2. Following irradiation, delay fuel handling for a brief period to allow short-lived isotopes to decay;

3. When handling irradiated fuel, monitor often with radiation detection instruments;

4. Prior to the first count, practice a dry-run with the counting system setup to minimize fuel handling time;

5. Keep removed fuel pins as shielded as possible.

All fuel movements are to supervised by the reactor operator. Fuel movements into or out of the reactor shall be performed with all control rods fully inserted.

C. Auxiliary Equipment

1. Radioactive particle counting system.
2. Wristwatch, clock or equivalent.

D. Procedure

1. Perform background counts on all pins to be counted at the 18" position (Normally 60-100 second counts).
2. Place the fuel pins into the core at desired locations.
3. Perform a reactor startup and irradiate the fuel pins for a short period. Shut down the reactor.
4. After waiting a brief time, remove the fuel pins to be counted.
5. Start the counting process. Record the time and counts

from each counting step. The first pin to be counted will become the reference pin. Counts from this pin will be made throughout the entire counting process so that these counts can be used to determine the decay corrections for all other counts.

6. Count (normally 60-100 seconds) the 18" position on the reference pin. Record time, counts, and pin number.
7. Count at an axial location on one of the pins. Record time, counts, axial location and pin number.
8. Continue making counts at various axial locations, and at the 18" location of the reference pin. Record time, counts, axial location and pin number from each count.
9. Upon completion of counts, correct axial counts for background and for decay. Plot the counts as a function of position.
10. For radial measurements, perform the applicable sections from above. Counts on all pins are made at the 18" position. After correcting for background and decay, plot the counts as a function of position.

APPENDIX G

Method For Determining Absolute Core Power

A. Discussion

This procedure is intended to provide guidance on determining the absolute core power. Once determined, this power is then used as calibration information for the reactor power indicators.

Gold foils are evenly spaced along the length of the center fuel pin. Some of the foils will be completely covered with cadmium. The pin is placed back into the reactor and irradiated for a period of time. After the irradiation period, all foils are counted with an appropriate counting system.

Gold is used because of its favorable attributes. It has a large enough neutron absorption cross-section which can give good counting statistics, and it has a long enough decay half-life that allows time for counting after irradiation.

When a bare gold foil (Ag^{197}) is irradiated, both fast and thermal neutrons are absorbed to create Ag^{198}, which then decays by beta and gamma emission. A gold foil covered with cadmium however, is only effected by fast neutrons since the cadmium will absorb all thermal neutrons (also creates Ag^{198}).

Because the core power is dependent on thermal neutrons to create fission within the fuel, the thermal neutron flux is the item of interest. Since the bare gold foils can provide information on the fast + thermal flux, and the cadmium covered foils can provide information on the fast flux, the difference between the two can provide information on the thermal flux.

Once the thermal flux is determined, average core power can be found through calculation, given the macroscopic fission cross-section of the fuel, the core volume, and the energy released per fission.

B. Notes and Cautions

The irradiation time of the foils must be long enough to provide good counting statistics, but short enough so that dead time counting losses are minimal. Because this procedure requires handling of irradiated fuel, good work practices must be maintained to minimize personnel exposure to radiation.

The placement of bare and covered foils on the fuel pins should be arranged so that the thermal neutron flux depression caused by the cadmium cover does not effect the thermal flux on a nearby bare foil. For this reason, the minimum spacing between bare and covered foils should be about two inches. Also, when covering a foil with cadmium, make sure it completely covers the foil.

C. Auxiliary Equipment

1. Gold foils.
2. Cadmium covers.
3. Calibrated gamma or beta counting system.
4. Cellophane tape.
5. Wristwatch or clock.

D. Procedure

(Note: It is assumed that the reactor pre-startup checks have been successfully performed.

1. Remove the center pin and tape the gold foils along its axis at about two inches apart or more. Cover every other foil completely with cadmium,securing it with tape. Record the mass of each foil and placement of each bare and covered foil on the fuel pin.
2. Place the pin in the core center position. Record core loading and placement. Record moderator temperature.
3. Perform a reactor startup, and operate for a given length

of time, then shutdown the reactor. Note and record the operating history (ie., reactor period and length of time. and steady state power and length of time), and time of reactor shutdown.

4. After a short waiting, remove the center pin to allow removal of the foils.
5. Count each foil on the radioactive particle counting system (usually 60 - 100 second counts will suffice depending on irradiation time and power). Record counts from each foil and time of each count. Determine counting system counting efficiency and the background counts of the system.
6. After correcting the counts for decay and background, plot the counts as a function of axial position for both the bare and covered foils. Calculate the thermal flux as a function of position. With this, and the radial distribution found before, calculate the core average thermal flux. Calculate the average core power and compare to actual recorded power.