Manhattan College

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MECHANICAL ENGINEERING DEPARTMENT

50-199

August 10, 1989

Theodore S. Michaels, Project Manager
Standardization and Non-Power Reactor Project Directorate
Division of Reactor Projects III, IV, V, and Special Projects
Office of Nuclear Reactor Regulations
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Response to Questions Regarding HEU/LEU Conversion at Manhattan College (your letter of July 10, 1989)

Dear Mr. Michaels:

Enclosed are the responses to questions raised in your letter of July 10, 1989 relative to the Safety Analysis Report we submitted regarding the HEU to LEU fuel conversion for the Manhattan College Zero Power Reactor. Please let me know if any additional information is required.

Sincerely,

Robert E. Berlin
Reactor Administrator

A020
1/1

MANHATTAN COLLEGE ZERO POWER REACTOR
QUESTIONS AND ANSWERS

(1) Q What is the calculated "just critical" mass?

A "Just Critical U-235 Mass" in the LEU core is calculated by the Argonne National Laboratory (ANL) as 235 x 15 + 27.4 x 1 = 3552.4 grams (total U-235), where 15 represents the number of full fuel elements and 1 represents partial fuel elements.

(2) Q What is the fuel element worth versus grid position for the LEU fuel?

A There was no calculated "Fuel element worth versus grid position" provided by the ANL for the LEU fuel, since there was no such information provided by AMF Atomics for the HEU fuel in 1965 for comparison purpose. However, detailed calculations on two control rods (1 regulating rod and 1 shim rod) have been made by the ANL, using both the Monte Carlo method and diffusion theory.[a]

(3) Q Are there provisions for any out of core fuel storage? Please explain what provisions have been made to safely store the HEU fuel elements in the event that shipment is not possible on the day they are removed from the core.

A There is onsite capability for storage of all the HEU elements after removal from the reactor. The elements would be placed individually into cylindrical sleeves, and then placed four to a container in the original SYLCOR shipping containers the fuel elements were received in. This temporary storage procedure has been used in the past during tank cleaning and maintenance, and is documented in MCZPR records and in the August, 1983 SAR.

(4) Q Please provide any HEU versus LEU comparisons of power distributions in the fuel elements, and any power distribution versus fuel loading information in the partially loaded element.

A The power distribution and nuclear power peaking factors that were calculated by the ANL for the existing HEU core and the LEU reference core with the shim and regulating rods fully-withdrawn are shown in Figure 1.[a] The power distributions show the power per fuel element (in milliwatts) and the power peaking factors show the absolute peak power density in each fuel element (computed at the edge of the mesh interval with the highest power) divided by the average power density in the core fuel.

The data in Figure 1 shows that the power distributions and total power peaking factors are nearly the same in the HEU and LEU cores. However, the limiting fuel element in the HEU core is located in grid position 33 and the limiting fuel element in the LEU core is located in grid position 34. This is because the location of one fuel element was changed in the LEU core (from position 46 to position 14) to increase the reactivity worth of the regulating rod.

Instead of power distribution versus partial fuel loading in the LEU reference core, ANL provided us with the changes of excess reactivity due to the presence of the partial fuel element, as shown in Table I.[a]

(a) J.E. Matos and K.E. Fresse, "*Analyses For Conversion of the Manhattan College Zero Power Reactor From HEU to LEU Fuel*", ANL, February, 1989.

(5) Q Neglecting bias, calculational uncertainty, etc., what are the Manhattan (best estimate) published values for regulatory and shim rod worth? Please give a technical justification for the values you choose. In reference to Table 3.2, page 19, please explain the meaning and operational implications of the two parenthetical statements about "biases".

A Our Manhattan published values for regulating rod worth is $-0.9\%\Delta k/k$, and for shim rod worth is $-2.5\%\Delta k/k$. These values were measured by AMF Atomics for a critical assembly of the PTR reactor which has the same core as our MCZPR.

"Biases" here means the deviations of rod worth between previously measured values (by AMF Atomics) and current simulation data (by ANL) on the same HEU core. The major causes of such rod worth biases are:

(a) methods used for calculation, such as the Monte Carlo method and diffusion theory,

(b) U-235 fissile loading variation (generally $\pm 2\%$), and

(c) sensitivity caused by ppm Boron equivalents in fuel plate cladding materials.

Since ANL has included all the "biases" possibilities in their analyses for the LEU reference core, safe operation can be expected as long as rod integrity is maintained.

(6) Q Please resubmit page 19 to show the deltas in Table 3.2 that are missing ($\%\Delta k/k$) and correcting an apparent typo in section 3.3.1, seventh line, viz. reversing vs. revising.

A These corrections have been made (see enclosure).

(7) Q ... Please provide NRC with the details of your Zero Power Physics Test program in this area. Additionally, ... you should prepare a fuel loading plan ... Please provide such a plan.

A A: MCZPR Physics Test Program on LEU Core Excess Reactivity

Part I: Normal Tank Water Temperature Reactivity Tests (60 - 80° F)

HEU core excess reactivity measurement under normal tank temperature has been a routine experiment of the MCZPR operational program since 1965 (see Attachment I). Although reactivity shows a slightly positive value in the current HEU core, it is much lower than the allowable peaking value of $0.44\%\Delta k/k$. The same experimental procedures will be followed for the LEU core excess reactivity test; however, it will be conducted each time with an increasing 5° F step-wise temperature change to ensure that maximum excess reactivity of $0.44\%\Delta k/k$ will not be exceeded within the range of 60 - 80° F.

Part II: High Tank Water Temperature Reactivity Tests (80° F and up)

Since maximum excess reactivity has been measured (not calculated) at 110.6° F for the HEU core, the same result may also occur during isothermal heating (based on ANL analysis). For safety reasons, we will repeat the reactivity test at each 3 - 5°F temperature increment a few times during a period of several days running, with the same procedure as in Part I to insure

a) uniform temperature in the tank,

b) consistent reactivity in the core, and

c) good agreement with ANL calculated data.

If measured excess reactivity of $0.44\%\Delta k/k$ is reached well below a temperature of 110.6° F, we will stop the tests and work with ANL to obtain further analysis and subsequent verification before our Technical Specifications revisions.

A B: <u>Fuel Loading</u>

Part I: Fuel Loading Plan

HEU fuel element removal and insertion has been performed at least every six years since 1965 for tank cleaning and component maintenance purposes. In order to avoid the abrupt changes of core reactivity as well as power level, we had proposed the fuel removal and insertion sequence, as shown in Table II, based on data recorded during the past 25 years. Table II shows that console meter readings were made during each 2-3 fuel element removal and insertion periods, for reactor power level (same as neutron multiplication) and Gamma radiation level checks. These standard procedures and the previous recorded data will be used as reference for HEU/LEU core conversion.

Part II: Fuel Loading Beyond Criticality

Based on "Reactor Period and Reactivity" experiment in the MCZPR (see Attachment I), reactor transient power has to be temporarily 25% over critical power in order to avoid an involuntary scram. This 125% power level (0.125 watt) was approved by the NRC during our license renewal in 1985.

As shown in Table 3-1 of Revision 4 of our Technical Specifications (see page 6), that both High Neutron Flux "count rate channel setting" and "linear channel setting" are allowed to reach 125% of full power rating. These safety systems (Table 3-1), which are particularly designed for reactor transient analysis would allow us to handle LEU fuel loading beyond criticality.

A C: <u>Reactor Power Level Determination and Rod Worth Calibration</u>

"Power Level Determination" and "Rod Worth Measurement" (includes Rod Worth Calibration processes) have been two important routine experiments in the MCZPR since 1965 (see Attachment II). Two methods will be employed for LEU core analysis in each of these two experiments. Recorded data from previous HEU core experiments will be used as reference for LEU core measurements.

(8) Q In the current LEU SAR, it is assumed that handling accidents where the fuel element is dropped results in no clad breach. Is this scenario the same as the handling accident discussed in your 1983 HEU SAR supporting the relicense?

A Yes, it is.

(9) Q Please explain how you will prepare the emergency shutdown rod for rapid use in the event of an error or other occurrence during fuel changes. (See section 4.2, SAR).

A The manual B_4C emergency shutdown rod, which is capable of shutting down the reactor in itself is located on the wall of the reactor facility within arms reach of a person standing on the platform. It can be rapidly lifted from its supports and manually inserted in the reactor core to accomplish immediate shutdown.

(10) Q Please resubmit only the revised sections of the Technical Specifications needed to accommodate the HEU to LEU conversion and any other change that you plan to make. Provide a brief rationale for the changes.

A The proposed revisions to the Technical Specifications and rationale are provided in Attachment III. This will supercede the proposed revisions 5 and 6 included in our May 8, 1989 submittal.

those which would appear in the proposed Technical Specifications for the LEU core (Rev. 6), are listed in table 3.2. The technical specifications for the LEU Core (Revision 6) are included as Attachment III to this Report.

TABLE 3.2 Parameters of HEU and LEU cores

Reactor Parameters	Core HEU	Core LEU	Page No. in Tech Spec (rev. 6)
Excess Reactivity, % $\Delta k/k$ (with -1.0% $\Delta k/k$ Bias to LEU Core)	0.32-0.40	1.1+0.4	3-1
Worth of Reg. Rod, % $\Delta k/k$ (with +0.3% $\Delta k/k$ Bias to LEU Core)	-0.9	-1.3	3-2
Shutdown Margin, % $\Delta k/k$ (with Shim Rod Stuck Out)	-0.5	-0.6	3-1
Worth of Shim Rod, % $\Delta k/k$	-2.5	-3.4	3-2

3.3 Description of Fuel Removal and Replacement

3.3.1 Steps in Removal and Replacement Processes

During the process of HEU/LEU core conversion, each HEU fuel element will be removed from the core and lowered into the fuel container (fuel cask) supplied by the EG&G Co. A sufficient number of containers will be obtained such that all 16 fuel elements (15 full and 1 partial fuel elements) can be sequentially removed from the core at one time, and then shipped to the DOE repository site. Immediately after the completion of HEU fuel removal, the new LEU fuel will be installed into the core, reversing the procedure for HEU fuel element removal. In order to avoid an abrupt change of reactivity in the reactor core and to prevent the fuel elements from obstructing each other, all the outer (circumferential) HEU fuel elements will be removed prior to that of the central elements and a reverse process of installing the LEU fuel elements will be carried out from the center of the reactor core. The detailed removal and replacement sequence are schematically shown in Table 3.3 and Figure 3-2.

Figure 1. Power Distributions and Power Peaking Factors

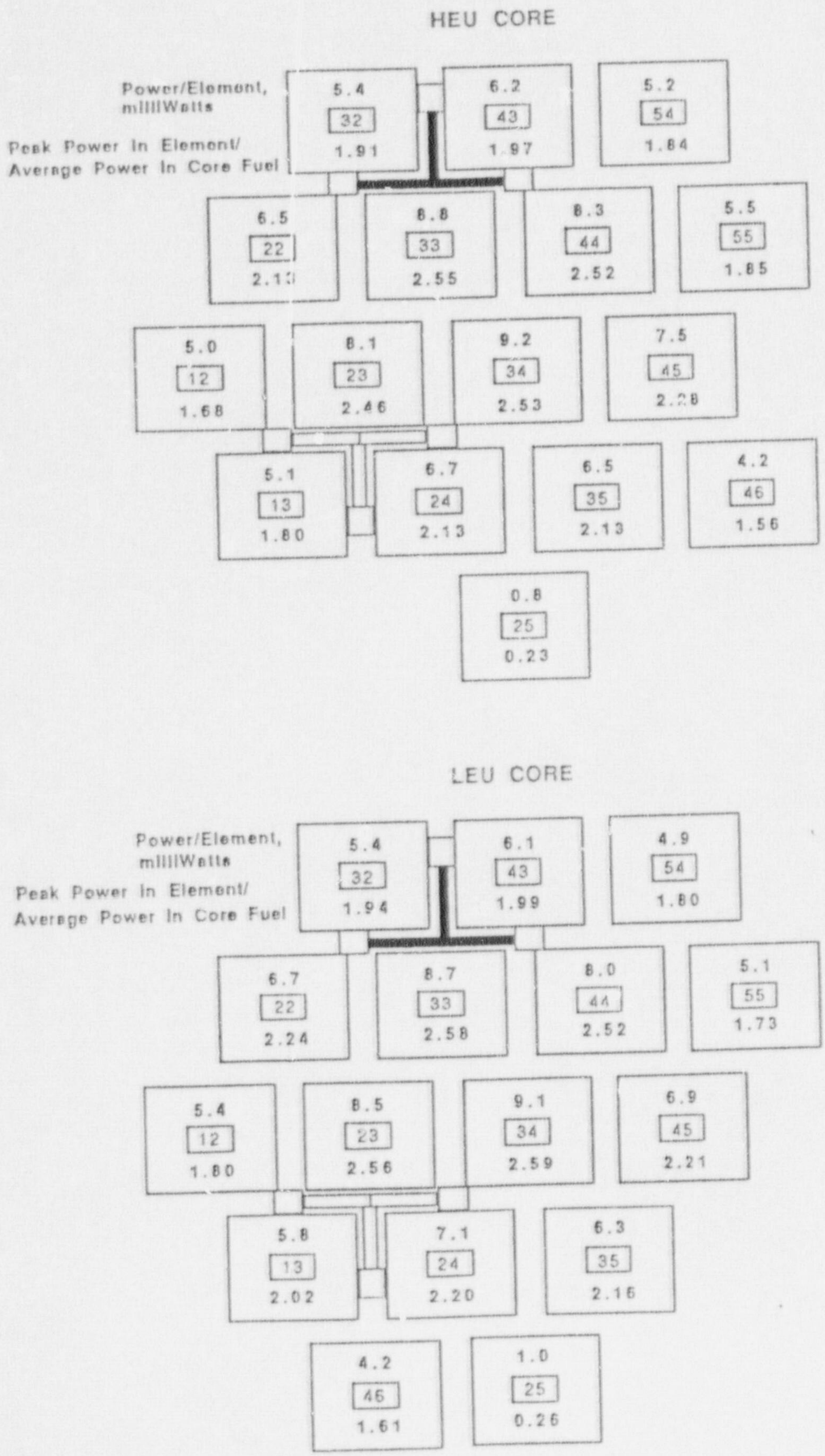

Table I. Sensitivity of LEU Core to the Number of Fueled Cylinders in the Partial Element

Case	Fuel in LEU Partial Element	Reactivity Change, $\% \Delta k/k$
1	Cylinder 2 Only	0.0 (Reference Core)
2	No Partial Element	- 0.33
3	Cylinder 4 Only	0.14
4	Cylinder 6 Only	0.26
5	Cylinders 2 and 4	0.37
6	Cylinders 2 and 6	0.47
7	Cylinders 4 and 6	0.57
8	Cylinders 2, 4, and 6	0.72

Table II. Removal and Replacement Sequence of HEU and LEU Fuel Elements

Fuel Element No.	HEU Fuel Removal Order	LEU Fuel Insertion Order	No. of Console Meter Readings
25 (partial)	1	16	1st
46	2	replaced by # 14	
14	empty	15	
13	3	14	2nd
12	4	13	
22	5	12	
32	6	11	3rd
43	7	10	
54	8	9	
55	9	8	4th
23	10	7	
24	11	6	
35	12	5	5th
45	13	4	
44	14	3	6th
33	15	2	
34	16	1	7th

ATTACHMENT I

MECHANICAL ENGINEERING DEPARTMENT

REACTOR PERIOD AND REACTIVITY

Objective

To determine the reactivity worth of a portion of the stainless steel regulating rod.

References

"Introduction to Nuclear Reactor Theory" by John R. Lamarsh, pp. 420-428, 437-439 and 441-442.

"Nuclear Reactor Physics" by Raymond L. Murray, pp. 156-160.

"Introduction to Nuclear Engineering" (Second edition) by Raymond L. Murray, pp. 131-138.

Theory

The theory involved in this experiment is explained adequately in the three references given above. The inhour equation is given on the accompanying pages.

Procedure

1. The reactor will be made critical with the Reg. Rod about 89% withdrawn and the picoammeter reading about 2 on the upper scale when the scale selector is set for 3×10^{-8} amperes. Introduce a step δk by moving the Reg. Rod to 100%. Obtain doubling times by clocking the time elapsed for the needle to move from 3 to 6 and from 4 to 8 on the upper scale. As soon as you have taken the reading at 8 switch the scale selector to 10×10^{-8} amperes in order to avoid an involuntary scram.

2. The reactor will now be made critical with the Reg. Rod about 84% withdrawn and the picoammeter reading about 1.5 on the upper scale when the scale selector is set for 3×10^{-8} amperes. Repeat the remainder of step #1.

Procedure (continued)

3. The reactor will now be made critical with the Reg. Rod about 75% withdrawn and the picoammeter reading about 2 on the 10×10^{-9} ampere scale. Introduce a step δk by moving the Reg. Rod to 100%. Turn the scale selector to 3×10^{-8} amperes when the needle reaches 7 on the 10×10^{-9} ampere scale. If the "Up Limit" switch clicks and the "Up Limit" light goes on, follow the remainder of the procedure in step #1. Otherwise it may be necessary to take the first doubling time from 3.5 to 7 or to omit it altogether.

Results Required

In each of the three steps of the Procedure, average the two doubling times and determine the reactor period. From the accompanying tables determine the reactivity input for each step in terms of both per cent and cents.

EXPERIMENT #20 - POWER LEVEL DETERMINATION

Introduction:

In high power reactors, it is possible to determine a given power level by calorimetric methods; core and coolant temperature measurements, coolant flow rates, etc. Low power reactors do not have this advantage since the heat generated during normal operation is small. Therefore, the approach used to measure a given power level of the Manhattan ZPR will not involve the product of nuclear fission, heat, but the initiator of the reaction, neutrons. Each fission which occurs in the reactor core releases an average energy of 200 mev. The rate at which energy is being released, the power level, is proportional to the fission rate. A measurement of the fission rate will therefore constitute a power level determination.

Two methods will be employed to determine the power level of the reactor and thereby calibrate the Log N and Linear channels. The first method will be a subcritical one while the second will employ the use of gold foil as an activation detector. An absolute thermal flux measurement will be made at the core center with the standard gold foil. This foil will be counted on an end-window G-M counter whose efficiency for the standard gold foil has been determined from a previous standard pile irradiation. The results of the measurement will yield the average thermal flux in the reactor core. A cadmium ratio measurement of the gold foil will also be made in order to determine the fraction of the total fission rate due to epithermal neutrons.

Theory:

Assuming an all thermal homogeneous reactor model, the following expression describes the total fission rate occurring when the reactor is in a steady state condition:

$$R_f \frac{\text{fissions}}{\text{sec.}} = \Sigma_f \int_{\text{vol.}} \phi \, dV \qquad (1)$$

where ϕ = the thermal neutron flux, the total neutron density times the most probable value of the thermal neutron velocity distribution at standard temperature.

Σ_f = macroscopic fission cross section of the fuel at the most probable velocity corrected if necessary for a non-1/v behavior.

The neutron flux will have a certain spatial distribution in the core depending on the geometry of the system. If the average thermal neutron flux can be determined, equation (20. 1) will simplify to

$$R_f = \Sigma_f \bar{\phi} V_{score}$$

with $\phi = \int_V \phi \, dV / \int_V dV$, the spatial average of the thermal flux.

The power level of the reactor corresponding to the average thermal flux is then equal to the total fission rate divided by the number of fissions per second required to produce one watt of power. For a reactor utilizing U-235 fuel,

$$\text{Power} = \frac{R_f}{3.1 \times 10^{10}} = \frac{\Sigma_f \bar{\phi} V}{3.1 \times 10^{10}} \quad \text{Watts} \qquad (2)$$

Although equation (20. 2) has been derived for a simplified model, its accuracy when used for the Manhattan ZPR power calibration will depend primarily on the average thermal flux measurement and to a lesser extent on a correction made for non-thermal fission. The energy distribution of the neutrons in the core actually covers a wide range, from fission energies down to the thermal region. Before the neutrons are slowed down into the thermal region, some will be captured in U-235 and U-238 and hence cause additional fissions. The fission occurring in U-238 is caused only by neutrons of high energy. The threshold reaction's contributions to the total fission rate can be assumed small for the Manhattan ZPR reactor since its moderator to uranium volume ratio is appreciable and its fuel is [illegible]iched with the U-235 isotope. Very fast fission is normally accounted for in the four factor formula by the factor ϵ the number of neutrons produced by all fissions divided by the number produced by thermal fission. In the Manhattan ZPR non-thermal fission is predominately resonance fission since U-235 has finite fission cross sections at all energies. The amount of epithermal fission can be determined by a simple cadmium ratio measurement of Manhattan ZPR type fuel. The fission product activity of a bare and cadmium covered fuel sample can be counted on a proportional counter after two similar irradiations in the reactor core. Their ratio will yield the amount of non-thermal fission to the total fission after proper corrections for sample weight differences, irradiation times and power level differences have been made. The final power level expression then becomes

$$P = \frac{\Sigma_f \phi V}{3.1 \times 10^{10}} \left[\frac{CR}{CR-1}\right]_{\text{fuel}} \quad \text{Watts} \qquad (3)$$

where CR = fuel sample's cadmium ratio.

The average spatial thermal flux is actually the total neutron density times a velocity of 2200 m/sec. This is true because the absolute thermal flux is measured with a 1/v absorber using its cross section at this standard velocity. The macroscopic fission cross section of U-235 is that of the same velocity. Actually, since this cross section does not have a 1/v dependence, a correction can be made to make it an equivalent 1/v cross section, which will give the correct fission rate in equation (20. 3). This correction is based on a Maxwellian neutron distribution being present and is a function of the neutron temperature. The temperature of the neutrons in the Manhattan ZPR core is approximately ^{o}K which corresponds to a correction factor of 0.

The absolute flux measurements at the core center with a standard gold foil requires the knowledge of the fundamental activation analysis. The activity of the foil immediately after the irradiation can be expressed as

$$A_o \frac{\text{dis}}{\text{sec}} = N_T \sigma_{act} \overline{\varphi} K (1 - e^{-\lambda t}) \qquad (4)$$

where N_T = total number of detector atoms.

$\sigma_{act.}$ = thermal neutron activation cross section at 2200 meters/sec.

ϕ = thermal neutron flux (nv_{2200})

K = thermal flux depression, self shielding factor

λ = detector's decay constant

t = irradiation time.

If the detector is counted on a system whose inverse efficiency, E(dis/sec/count/sec), is known at a time t_c after irradiation, the detector's activity becomes

$$EC = A_o e^{-\lambda t_c} \qquad (5)$$

where C = counts/sec at t_c.

Solving (20. 4) and (20. 5) for the neutron flux

$$\phi = \frac{EC\, e^{\lambda t_c}}{N_T \sigma_{act} K (1 - e^{-\lambda t}} \qquad (6)$$

The resulting neutron flux is the proper flux to be used in equation (20. 3) only if the activity is caused by thermal neutrons alone. Since the cross section of gold has a strong resonance activation peak, a correction must be made to separate the resonance activation from the thermal activation. This can be done by making a cadmium ratio measurement with the standard gold foil. If CR is the measured value, the thermal activity of the detector becomes

$$A_{o\,th} = A_o \left[\frac{CR-1}{CR}\right] A_o \text{ Foil}$$

Hence, the flux, in(20. 6) must be multiplied by the factor in the brackets to yield the correct thermal flux.

Procedure:

Method 1 of determining the core power level will be a subcritical method. Upon completing the initial critical for the Manhattan ZPR, the regulating rod will be calibrated using period measurements. After determining the worth of the regulating rod, the core will be held at criticality with the source in and all indicating instrumentation readings will be recorded. After all readings at criticality are completed, the calibrated regulating rod is inserted until the reactor is subcritical by a known amount. The reactor power level will decrease and level at a new lower level which can be calculated using the following relationship:

$$\text{Power (watts)} = \frac{0.2\, S_o}{(1 - K_{eff})\; 7.55 \times 10^{10}}$$

S_o is estimated from the Manhattan ZPR source strength (1.8×10^6 n/sec)

k_{eff} is known from the rod calibration.

Method 11 of determining the core power level will be done using gold foil irradiation data. A bare gold foil and a cadmium covered gold foil will be placed on an aluminum holder so that each foil will be located a distance 6. 5 in. above and below the center of the active core length in fuel element #20 in core position 33. This distance will place the two foils at the position calculated to see the average axial flux in this fuel element. The max/avg. of the core radially will be 1. 44.

Place the foil holder into fuel element #20 between fuel plates #3 and #4 and bring the reactor to critical leveling at same instrument readings used for Method 1. Remove the start-up source maintaining criticality. Irradiate the foils for 15 minutes. After 15 minutes irradiation, shut down reactor and remove foils. The foils will be counted on the Manhattan G-M Counter that has been standardized for the gold foil. Count foils and correct count rate measured back to zero time.

The average thermal flux seen by the gold foil is determined as follows:

$$\varphi^{Au}_{th} = \frac{A_B(t_d)}{\sigma^{Au}_{act}} \; \frac{(CdR-1)}{CdR} \; \frac{At.\,Wt.}{0.623} \; \frac{e^{\lambda t_d}}{1-e^{-\lambda t_e}}$$

A_B = absolute activity per gram of bare foil at time t_d (dis/sec gm)

$C_dR = \frac{AB}{A_{cd}}$ At. Wt. = 19.7, λ - $2.96 \times 10^{-6} sec^{-1}$, σ_{act}^{Au} = 85 barns.

t_d = delay time between exposure and counting of foil (sec)

t_e = duration time of exposure (sec)

The power level of the core is calculated using the following relationship:

$$\text{Power (watts)} = 3.54 \times 10^{-11} \times M \times \bar{\varphi} \text{ core } \frac{P_T}{P_{Th}}$$

where M = Mass of U-235 in core at time gold foils are irradiated in gms.

φ core = $\varphi_{th}^{Au} \times 0.695$

$\frac{P_T}{P_{Th}}$ = Ratio of total power to power resulting only from thermal fission = 1.20

Prerequisites

A. Normal start-up instrumentation must be operative

B. Initial critical loading must have been completed

C. Equipment required:

1. Two gold foils with known weights
2. One cadmium cover
3. Foil holders
4. Manhattan counting room equipment
5. Calibration data for one safety rod

Precautions:

Normal operating procedures will be followed. Power level must be held steady during irradiation of gold foils, and as low as practical for the performance of the experiment. A power level of a hundred mw is desirable. Foil holder will be placed into fuel element #1 holddown rod with core in shutdown condition.

Reactor Conditions:

The reactor during this measurement will have the initial critical loading All instrumentation will be operative

DATA SHEET 1 - 4

INITIAL POWER LEVEL DETERMINATION

METHOD 1

Core Loading # _______________

U-235 Content _______________

Pool temperature _______________

Start-up Source Type __________ Strength __________ Location __________

Critical Rod Positions (Source in core)

Reg. Rod _______________

Shim Rod _______________

Critical Rod Positions (Source removed)

Reg. Rod #1 __________ #2 __________ #3 __________ #4 __________ #5 __________

Shim Rod #1 __________ #2 __________ #3 __________ #4 __________ #5 __________

Period Measurement		Trial #1	Trial #2
Reg. Rod	Critical (source out)	__________	__________
Position	Super critical (source out)	__________	__________

Trial #1 Period __________ sec

Trial #2 Period __________ sec

Use in-hour curve to determine Δ k required to give above periods

Differential worth of Reg. Rod Trial #1 __________ Trial #2 __________

Insert start-up source (note core must be made subcritical before inserting source)

Return reactor to critical, adjust all rods in positions of previous critical with source in.

Record all instrument readings at criticality

Start-up channel______________Linear ______________% ______________Range

Start-up channel______________Log N ______________

Insert Reg. Rod until the K_{eff} of core is subcritical a known amount.
(use above differential worth to determine K_{eff})

K_{eff} (Subcritical ______________

Reg. Rod ______________

Allow power level to drop until it levels off. After a steady state subcritical condition is reached; record all instrument readings.

Record all instrument readings at level subcritical position

Start-up channel #1______________ Linear______________% ______________Range

Start-up channel #2______________ Log N ______________

DATA SHEET 1 - 5

INITIAL POWER LEVEL DETERMINATION

METHOD 11

Core Loading #______________ Date______________

U-235 Content ______________

Pool Temp. ______________

Start-up Source Type____________ Strength____________ Location__________.

Prepare foil holder with two gold foils so that when placed in core the gold foils will be located 9.5 inches above and below the core midplane. One gold foil will be cadmium covered and the other will be bare.

Foil Data

Gold Foil #1 (Bare) Gold Foil #2 (cd covered)

Id. #______________ ______________

Wt. ______________ ______________

With the core shut down, insert the foil holder containing the foils into the hold down rod of the fuel element in #33 position.

Bring the core to criticality and allow power to rise one decade above previous criticality instrument readings. Level the power and remove the start-up source.

Record time Power Level attained__________________

Critical Rod Positions (source out)

Reg. Rod ______________

Shim Rod ______________

Instrument Readings

Start-up Channel #____________ Log N______________

Start-up Channel #____________ Linear__________% ________________Range

After ~15 minutes shut down core and remove foil holder

Time of shut down ____________________

$t_{exposure} = t_e =$ __________ seconds

Foil Data

	Gold Foil #1	Gold Foil #2
Time Counted	________	________
td (decay time)	________	________
Total counts	________	________
Activity (dis/sec. gm)	________	________
Counter efficiency	________	________

$$\text{Cd Ratio} = \frac{\text{A Bare}}{\text{A Cd covered}} = \_\_\_\_\_\_\_\_$$

(PERIMENT #27 - ROD WORTH MEASUREMENTS

:st Purpose:

To determine reactivity worths of the cadmium control rod and the stainles: eel regulating rod.

st Summary:

Rod worths will be determined using the rod drop and positive period ethods. By reference to the Core Diagram, the regulating rod is between core positions , 23 and 24, while the cadmium safety rod is located between core positions 32, 33 and .

erequisite Operations:

General:

The reactor core loading shall be limited to the initial critical loading : the start of these tests.

Instrumentation:

All instrumentation shall be checked out and in proper working order. recheck shall be made of the initial criticality data for the rods 50% withdrawn.

ecautions:

Rod Withdrawal

Do not introduce a period less than 20 seconds.

Bypass Switches

Bypass the same scram circuits as for the "Criticality Test."

itial Plant Conditions:

1. H_2O at normal operating level.
2. Initial criticality core loading.
3. Source in position as indicated in Core Diagram.
4. Reactor ventilating system operating.
5. Reactor water purificating system operable.

Procedures:

Rod Drop Method

1. Using the start-up channel, remove source and measure any extraneous background level.

2. Insert the source to the original position.

3. Bring the reactor to critical and adjust the neutron flux level to a decade or two above the neutron flux level due to the source with all rods in, and operate for a few minutes to allow most of the delayed neutrons to stabilize.

4. Remove the source.

5. Record the rod positions as shown on the fine position indicator.

6. Take three one (1) minute counts on the start-up channel counting circuit. Record these counts.

7. Drop both rods by pressing manual scram button after operation for 10 minutes.

8. Measure total counts on the start-up channel between 30 and 90 seconds after the instant of scram. This operation may be done manually by use of a stop watch but it is recommended that an automatic timer be employed to start and stop the BF_3 channel scaler for the desired time interval.

9. Repeat steps (1) through (6) except position rod to be tested full out.

10. Drop the desired rod by reducing the magnet current.

11. Repeat step (8).

12. Repeat steps (9) through (12) for the second rod.

13. Calculate the following ratio for each rod tested.

$$R = \frac{\text{(Count Rate at Power (cps))}}{\text{Integral Counts (30 - 90s)}}$$

14. Determine the worth of the rod tested by reading the negative reactivity effect, i.e. % excess k, from the graph of -% excess k versus R using the measured value of R. (see accompanying graph).

Positive Period Method

1. Bring the reactor critical with the neutron flux level a few decades above the shutdown source level. Withdraw the shim safety rod fully while keeping the flux level stabilized by adjusting the regulating rod critical position.

2. With the flux level stabilized, withdraw the regulating rod a predetermined distance from its critical position and measure the resultant period on the Linear and Log-N circuits. Period measurements should be between 30 and 50 seconds in order to obtain reasonable data. Record initial and final positions of the regulating rod.

3. Insert the shim safety rod to stabilize the flux at the level measured prior to the period test.

4. Repeat steps (2) and (3) until regulating rod is calibrated over its complete range.

5. From the period measurements and the reactivity versus period curve, plot the worth of the regulating rod versus the distance above the lowest critical level.

6. Repeat steps (1) through (5) for the shim safety rod.

NOTE: Since either rod can shut the reactor down, shim and regulating can be interchanged.

Data Required:

1. Differential and integral rod worth curves for each rod as a function of rod position.

2. Rod worth measurement data for rod drop method.

3. Core conditions under which measurements were made.

Attachment III

Proposed Revisions To Technical Specifications

The following proposed revisions to the Technical Specifications are divided into two (2) groupings; Part A are revisions to correct existing terminology; & Part B are the revisions to accommodate the HEU to LEU conversion

A. Corrections to Existing Technical Specifications:

(1) Page 1-1, delayed neutron fraction

Revise to read "when converting between absolute and dollar value reactivity units, a beta effective delayed neutron fraction of 0.0078 is used".

Rationale: A delayed neutron fraction of 0.00645 for converting between absolute-and-dollar-value reactivity units is not correct. A delayed neutron fraction of 0.00645 is the value for pure U-235. It is the number of delayed neutrons divided by the total number of neutrons emitted when an atom of U-235 is fissioned and is independent of the uranium enrichment and the geometry of the reactor. The correct conversion unit is the effective delayed neutron fraction, which depends on the reactor geometry and the diffusion properties of the medium (see ANL-5800, 2nd Edition, Reactor Physics Constants, July 1963, pp. 441-444). A beta effective of 0.0078 for both the HEU and LEU cases has been computed and suggested by the Argonne National Laboratory.

(2) Page 1-2, reactivity limits

Delete the sentence "For the MCZPR the reactivity limits are 0.44% $\Delta K/K$ (0.68$) at 110.6° F".

Rationale: The reactivity limit is specified in paragraph 3.1.3 on page 3-1. Since individual reactor parameters are not generally included in the definition, and the sentence is repetitious of the paragraph 3.1.3 wording, consistency would suggest it be deleted.

(3) Page 3-1, 3.1.3 Specifications
A. Change (0.68$) to (0.56$)
B. Change (0.72$) to (0.59$)

Page 3-1, 3.1.4 Bases
Change (0.68$) to (0.56$)

Page 3-2, 3.2.3 Specifications
B. Change (1.40$) to (1.15$), and (3.88$) to (3.21$)
D. Change (0.154$) to (0.128$)

Rationale: The proposed revisions are consistent with using a beta effective of 0.0078 to convert absolute to dollar value reactivity units.

B. Proposed Revisions to Technical Specifications to Accommodate HEU to LEU Conversion

(1) Page 2-1, 2.1.3 Specifications

Revise to read: "The safety limit shall be on the temperature of the fuel element cladding, which shall be less than 1080° F".

Rationale: The cladding of the current HEU fuel elements is composed of 1100 (or 2S) aluminum which has a melting temperature of 1220°F. The cladding of the new LEU fuel elements will be 6061 aluminum, which has a melting temperature of 1080° F.

(2) Page 2-1, 2.1.4 Bases

Revise to Read: "The melting temperature of the aluminum used as cladding on the fuel elements is 1080°F. Therefore, in order to maintain fuel element integrity, the cladding temperature must not exceed 1080°F. As reported in Section 6.1.2 of "Analyses for Conversion of the Manhattan College Zero Power Reactor From HEU to LEU Fuel" by J. Matos, and K. Freese of Argonne National Laboratory (Reference 1). The maximum cladding temperature that can ever be reached is only 239°F (115°C) and reaches this level only during the Maximum Hypothetical Accident. The specification, therefore, provides assurance on the integrity of the fuel within the cladding".

Comment: Based on the same reason as in (1) above

(3) Page 2-2, 2.2.4 Bases

Revise to Read "Since there is no forced circulation cooling, the reactor core is cooled by the water surrounding the reactor core. Therefore, the only parameter which could be used as a limit for the fuel cladding temperature is the reactor power. The analysis in Reference 1 shows that even for the Maximum Hypothetical Accident (a reactor power excursion of 183 Kilowatts), the maximum cladding temperature reaches only 239°F (115°C). This temperature is much lower that the temperature (1080°F) at which cladding damage could occur. Therefore, a large safety margin exists between the safety system set point and the cladding safety limit."

The revised parameters are based on the ANL accident analysis (Reference 1) (Note in Section 6.1.2 (p.23) of Reference 1 that ANL personnel computed a peak power of 221 kilowatts and a peak cladding temperature of 241°F (116°C) instead of the peak power of 147 kilowatts and a peak cladding temperature of 221°F that is quoted for the HEU core in our Technical Specifications and SAR. None of our conclusions for the HEU core change because of this difference).

(4) Page 3-1, 3.1.3 Specifications, Paragraphs A and B., and 3.1.4 Bases

Comment: The reactivity limit for the LEU case has been assumed to be the same 0.44% $\Delta K/K$ as for the HEU case. If the 1966 isothermal heating experiment were redone for the LEU case, the maximum excess reactivity may not be exactly 0.44% $\Delta K/K$ at a pool water temperature of exactly 110.6°F.

With regards Paragraph B,

The actual minimum shutdown margin of the LEU core will not be known until the reactivity worth of the regulating rod and the maximum excess reactivity are actually measured. In Section 5.4.4 of Reference 1, the minimum shutdown margin in the LEU core is estimated to be 0.56% $\Delta K/K$ larger than in the HEU core because the fuel element in position 46 of the HEU core was moved to position 14 of the LEU core in order to increase the reactivity worth of the regulating rod.

After completion of testing of the LEU core, revisions to the paramters in Paragraphs A&B will be proposed, as well as the parameters in the first sentence of section 3.1.4.

(5) Page 5-3, 5.3.2 Reactor Fuel

Revise the first four sentences to read:

"The fuel portion of the elements consists of six concentric cylinders formed by mechanically joining and positioning eighteen curved fuel plates wihin grooves of three spacer webs. The cylindrical fuel plate consists of 0.020 inch-thick U_3Si_2 - Al fuel meat containing uranium enriched to 19.75 ± 0.2% in U-235 and clad on both sides with 0.015 inch of aluminum, making the total fuel plate thickness 0.05 inch. The nominal U-235 content of each full fuel element is 235 grams. The inner diameter of the innermost cylinder is about 1.25 inches and the spacing between adjacent cylinders (water channel width) is 0.118 inch".

Rationale: The revisions are consistent with the parameters of the new LEU case. The last 2 sentences relating to the exact number of fuel elements in the LEU case will be revised after a critical core satisfying all the Limiting Conditions for Operation is assembled.

In this regard, the nine fuel plates in cylinders 2,4, and 6 of the partial fuel element are removable. The nominal U-235 content of the three fuel plates in each of the three full cylinder are 27.4, 43.7, and 58.4 grams, respectively (see Reference 1, Table 1, p.5). If the partial fuel element is needed, the minimum loading is 9.1 grams U-235 in one fuel plate in cylinder 2 and the maximum loading is 129.5 grams.