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Dose rate determination of the highly-enriched uranium fuel at the University of Missouri-Rolla Reactor in preparation for transportation

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DOSE RATE DETERMINATION OF THE HIGHLY-ENRICHED

URANIUM FUEL AT THE UNIVERSITY OF

MISSOURI - ROLLA REACTOR

IN PREPARATION FOR TRANSPORTATION

by

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ABSTRACT

An analytical model was developed in order to accurately estimate the dose rates in air of the highly-enriched uranium (HEU) fuel at the University of Missouri - Rolla Reactor. Based upon these results, a method is proposed to remove the fuel from the reactor.

First thermoluminescent dosimeters (TLDs) were exposed under water to each of the fuel elements in the reactor pool, except for the four control rodded elements. A model was developed to use the in-water TLD readings to calculate the dose rate of the fuel elements in air at 1 foot and 3 feet. The fuel was modeled first as a cylindrical source; then as a line source. Since both models seemed to underestimate the source strength of the fuel, each of the fuel elements were then approximated as a line source with a cosine distribution along the line. Once the dose rates in air had been predicted, a single element was removed from the pool, and TLDs were exposed to the element to determine the actual in-air reading. The cosine distributed line source appeared to be somewhat of an overestimate; thus the results it gave were conservative in determining the strength of the source. Since the model yielded good results, it was adopted for all of the elements.

Fuel element F9 had the highest dose rate which was calculated to be 54.7 rem/hr +/- 10% at 1 foot and 18.1 rem/hr +/- 10% at 3 feet in air. Based on the high dose rate of this element, and several others, it was decided that the transporting of the elements could be broken down into two shipments with the least radioactive being taken on the first trip; then removing and shipping the others after storage in the spent fuel pool for a year or more.

ACKNOWLEDGEMENTS

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A very special thanks goes to Phil Simpkins, the author's fiance, whose encouragement and love motivated her to finish her thesis, also for Phil's understanding that research had to come first.

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I. INTRODUCTION

The United States Nuclear Regulatory Commission (USNRC) has mandated that the University of Missouri-Rolla (UMR) Reactor convert from highly-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel. Currently the UMR Reactor uses MTR plate-type elements that contain U_3O_8 in an aluminum matrix. The uranium enrichment is nominally 90% U-235. The LEU fuel will be less than 20% enriched. It will also have plate-type elements. Currently, there are 14 elements, 1 half-element and 4 control rod elements in the core and 8 fuel elements and 1 half element in the spent fuel pool. The HEU fuel that currently is in the UMR Reactor pool is owned by the Department of Energy (DOE).

Before the LEU fuel can be placed in the core grid plate, the HEU fuel must be removed. It can be temporarily stored in the spent fuel pool that lies at the end of the reactor pool; however, it must ultimately be removed from there and transported to a DOE facility. Before it can be removed, the activity needs to be carefully measured in order to determine the amount of shielding needed for transport. This study expands on previous work (1) and provides a more complete study analyzing all of the elements that are on site, except for the four control rodded elements that are in the core.

Several other research reactors in the country have already completed the process of converting their fuel and shipping the HEU fuel off site. The general procedure is to use a transfer cask to remove the fuel from the building where it is then placed in a shipping cask. A transfer cask is already in existence for this purpose. The cask is made of lead, is approximately 44 inches tall, has a diameter of 19 inches, and weighs 4650 lbs.

The University of Missouri - Rolla reactor facility was built in 1959 and has concrete floors in the bay that are 12 inches thick. Underneath the concrete is a bed of gravel, which has probably settled, making the concrete virtually free standing. Details regarding any reinforcement are not certain. With this in mind, the floor will not support the weight of a 4650 lb cask. A risk of cracking the floor cannot be taken because the crack could lead to the pool wall which would allow some, or possibly all, of the coolant to leak from around the pool.

A second idea for transport would be to lower a fifty-five gallon drum, with spacers in the middle to keep the fuel stationary into the pool, flood it with water, transfer the element into it under several feet of water, lift the drum out of the water, and move it to the shipping cask. This procedure, however, is not possible at the University of Missouri-Rolla Reactor because there is no overhead crane and the ceiling structure is not strong enough to support the weight of the fifty-five gallon drum flooded with water. The process could be done if a support (Such as a double A-frame) was built over the pool to handle this weight; however, this would take additional time and money.

The purpose of this thesis is to determine the dose rates of all the fuel elements as accurately as possible both in air and encased in a fifty-five gallon drum. Once this is done, a method will be proposed to remove the fuel from the building. This is a necessity for the Department of Energy to safely remove the fuel from the reactor facility.

II. ACTIVITY DETERMINATION

The method employed to experimentally determine the activity of the fuel was to use thermoluminescent dosimeters (TLDs). Previous work had been done using TLDs, and an apparatus has already been devised for exposing them to the elements under water.(2) TLDs were chosen because they work well under water when encapsulated, can be reused upon annealing, and yield reliable results for the expected dose rates.

A. PRINCIPLE OF THE THERMOLUMINESCENT DOSIMETRY(TLD)

When a TLD crystal is exposed to ionizing radiation, electrons leave the valence band and go to the conduction band. An electron, along with the hole that is left behind, will migrate through the crystal until it recombines or is trapped in metastable states.(3) These metastable states or "traps" are formed by foreign atoms (impurities, interstitial atoms), dislocations, vacancies and imperfections. As long as the temperature of the crystal remains constant or decreases, these traps stay in the same place. As the temperature of the crystal increases, the probability of an electron escaping from the trap increases. These electrons and the holes emit light as they are freed and return to the ground state. This emission of light is called thermoluminescence, thus the name TLD.(4)

Various TLD readers are commercially available. The system must be able to heat the TLD under specific conditions, contain a detector which is the sensitive to light emitted, and have a recording instrument. The lithium fluoride TLDs that were used in this experiment were heated at a constant rate of 8 °C per second after a preheat of 100 °C. They were then evaluated at a constant temperature of 240 °C. Once they have been read, they can be annealed at 400 °C for one hour, then at 100 °C for 2 hours. After this annealing procedure, they were ready to be reused.(5)

B. THERMOLUMINESCENCE DOSIMETER CALIBRATION

Before the results from exposing the TLDs to the fuel can be of any value, a calibration model must be developed. This was done by exposing the TLDs to a Cs-137 source of known strength for a known period of time at several known distances from the source. Since one knows the strength of the Cs-137 source when it was received, the current source strength can be determined using the exponential decay law. Once this is done, the source strength is divided by the distance from it, squared , to determine the dose rate at the various distances where the TLDs were exposed. A plot was made of TLD reading versus calculated dose rate and a least squares fit was applied to the data to determine the equation for the line.

Two different calibrations were done -- one low-range and one high-range. The low-range calibration contains TLD readings between 0 and 40 nanoCoulombs(nC). It is depicted in Figure 1. Because the error on this set of points was minimal, error bars were not shown. The equation for a least squares fit between these points yielded:

$$\text{Dose(mrem)} = 26.8*(\text{TLD reading,nC}) + 1.37 \qquad \text{(eq. 1)}$$

The high-range calibration can be used for TLD readings up to 1600.0 nC. The plot for it is shown in Figure 2. Because the error in the TLD readings on this graph was also minimal, error bars were not shown. The equation for the line was determined to be:

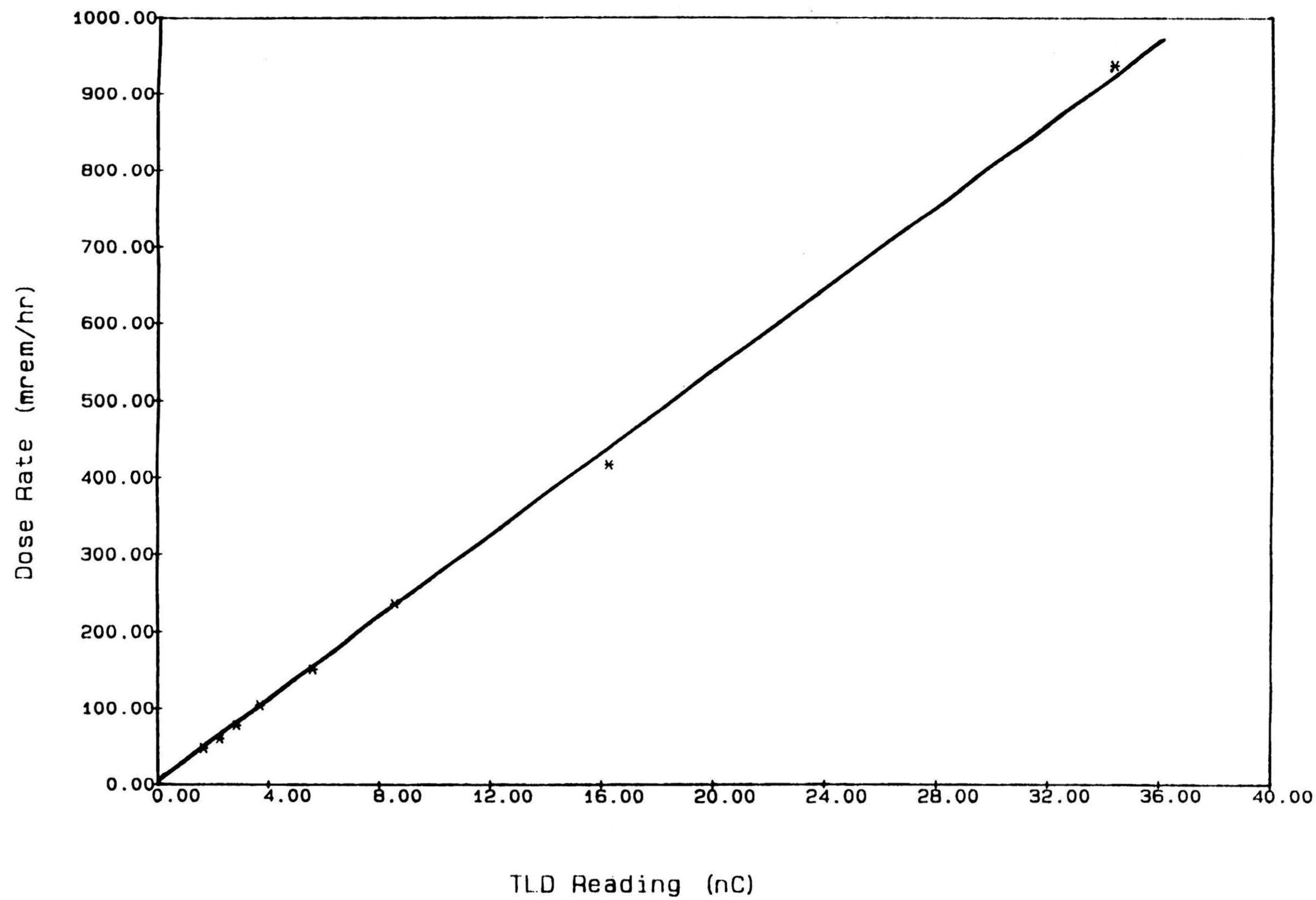

Figure 1. Low-Range Calibration Plot

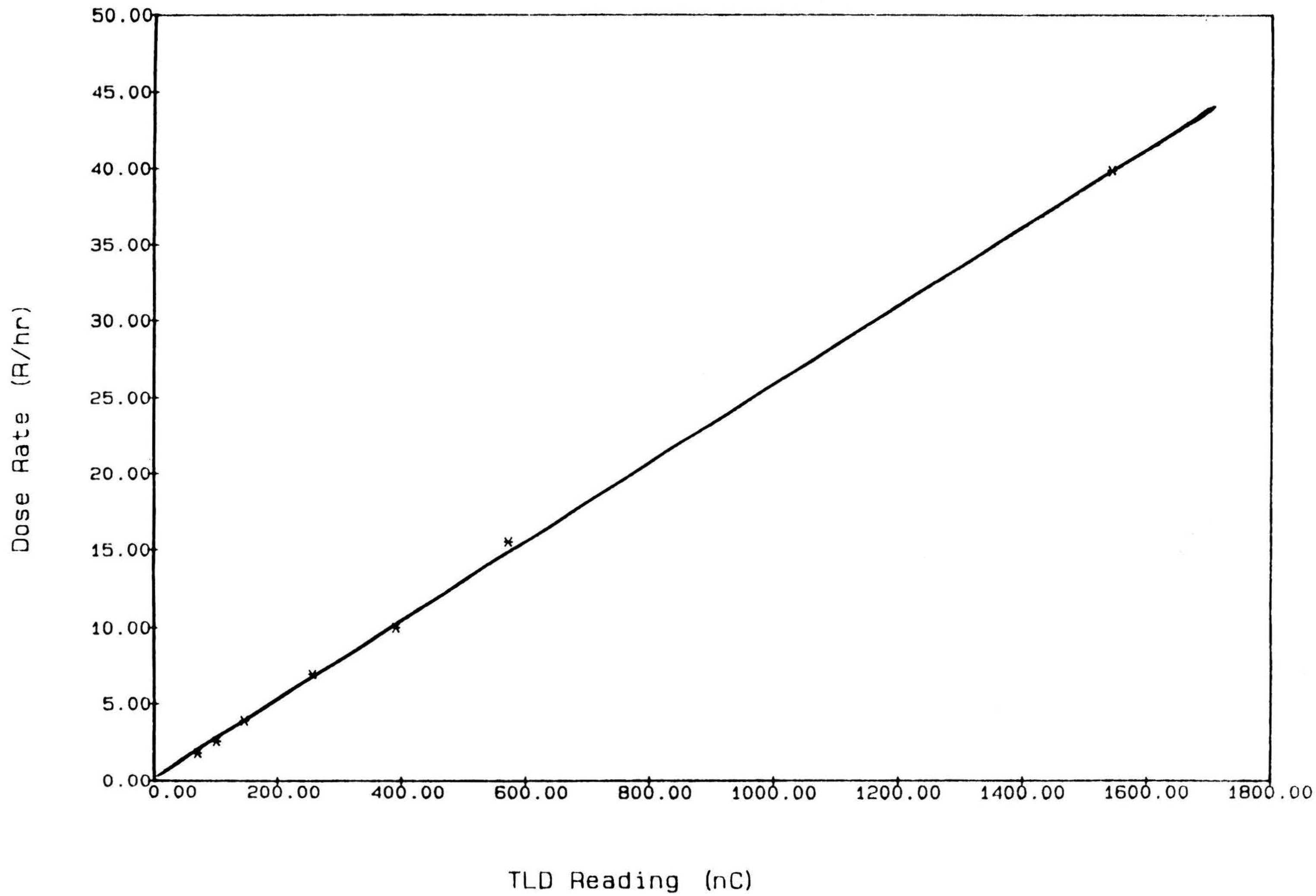

Figure 2. High-Range Calibration Plot

$$Dose(mrem) = 26.2*(TLD\ reading, nC) - 32.8 \quad \text{(eq. 2)}$$

A PASCAL program was written using a least squares fit to determine the equation for a straight line given a set of points. It is shown in Appendix A.

The error for the slope of this line was estimated using the following equation:

$$\sigma^2 = s^2/[\Sigma x_i^2 - (\Sigma x_i)^2/n] \quad \text{(eq. 3)}$$

where $s^2 = \Sigma(y_{actual} - y_{equation})^2/(n-2)$. Using the above equation, the error in the slope of 26.2 mrem/nC was found to be 0.32 mrem/nC.(6) Since the error was so low, the high-range calibration was used to determine the dose for all of the TLD measurements, except for those where the TLD readings were less than 10 nC; then the low-range model was used.

C. EXPERIMENTAL PROCEDURE

1. TLD Exposure in Water The gamma survey device used in this research was the same one used by J. Joel Smith. (7) Device design drawings are shown in Appendix B. This device is designed to give both a horizontal and vertical profile.

TLDs were exposed to all of the elements in the spent fuel pool and the core except for the control rodded elements. See Figure 3 for a diagram of the reactor pool. Figure 4 shows a more detailed view of where the elements were located in the core and spent fuel pool. The gamma survey device was placed in position R-1 (See Figure 4); then the fuel element to be measured was moved to position R-5. The TLDs were

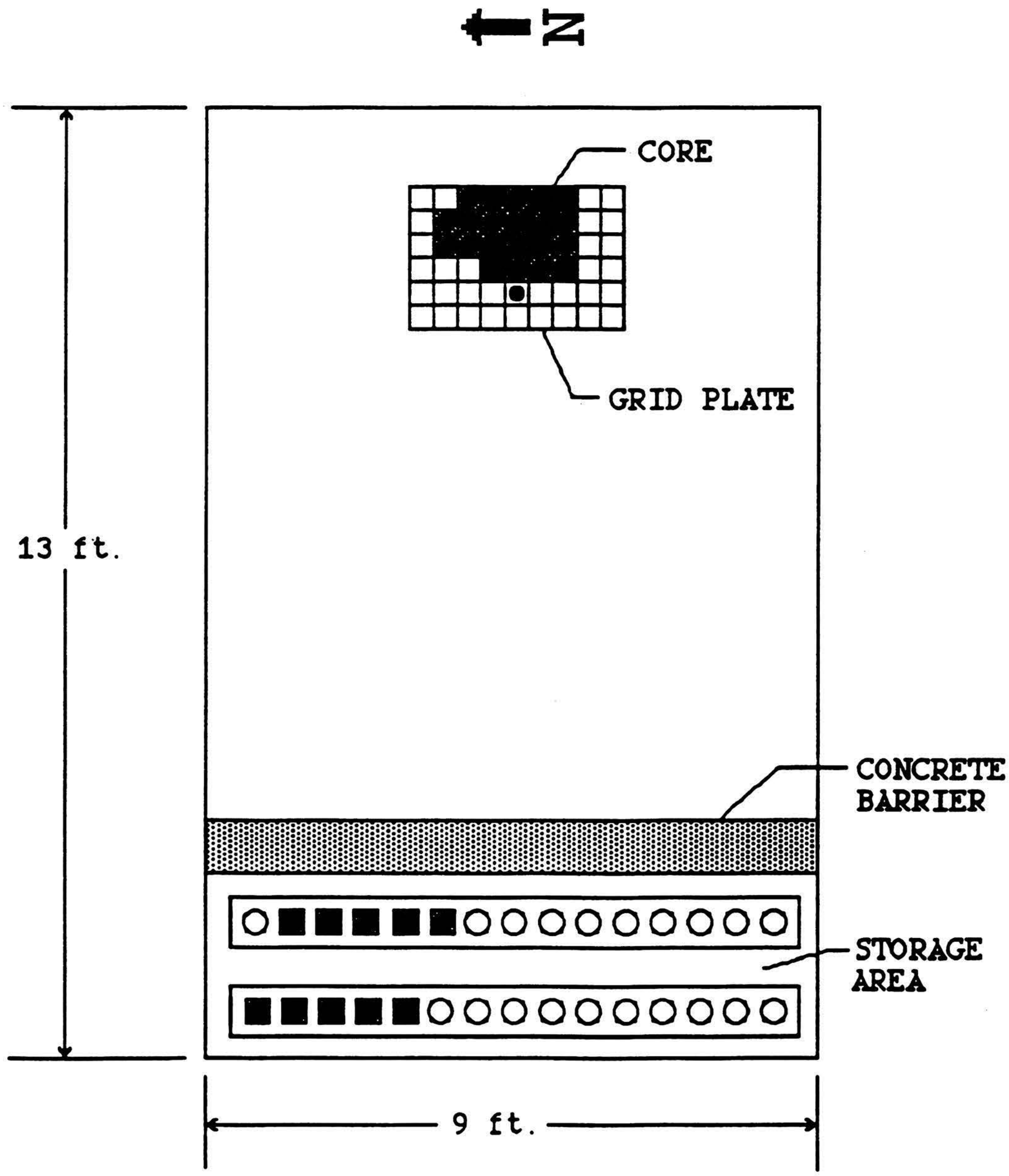

Figure 3. Reactor Pool Layout

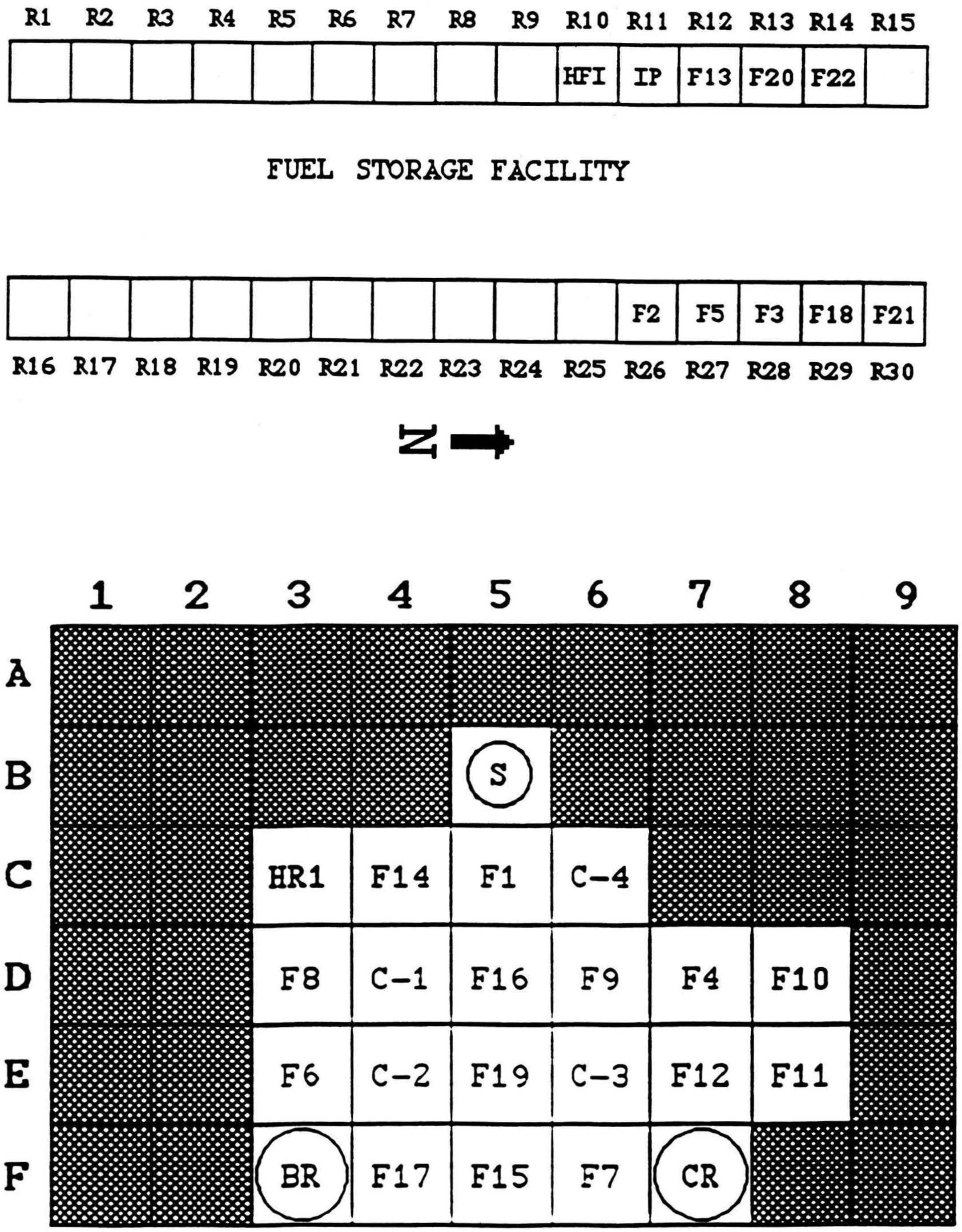

Figure 4. Diagram of Core and Fuel Storage

exposed for five minutes at distances of one, five, nine, and thirteen inches from the element. Three TLDs were placed at each position, and the average of the three was taken for calculations. Once the TLDs are read, equation 2 was used to determine the dose rates that correspond to the various distances in water.

2. TLD Exposure in Air. Next, an in-air TLD exposure was done for a single element. The half element (HF1) was chosen because it was determined to be the least active after doing the in-water TLD exposures. First, the element was raised just to the surface of the pool to see how high the dose rate was above the water and through the 12 inches of concrete on the side of the pool. Through calculations a dose rate of approximately 500 mrem/hr at 1 ft was determined and by measuring the dose rate through the concrete, and doing some hand calculations, we justified these assumptions.

Considering that a fuel element has not been taken out of the UMRR pool, it took some time to devise a method to do it. Due to the length of the fuel handling tool, it cannot be used to take an element out of the pool because the tool would hit the ceiling when it is held straight, and if it is held at an angle the connection that holds the element is not reliable. It was decided that the fuel element would be lifted by placing it on a rope with a hook at the end. Since the dose rate at one foot for this element was expected to be 500 mrem/hr, minimum time was spent near the element, and all work was done as far away from the element and as quickly as possible. Figures 5 and 6 show the setup for the in-air measurement.

The TLD holder was placed on a 12 foot pole so that its handling

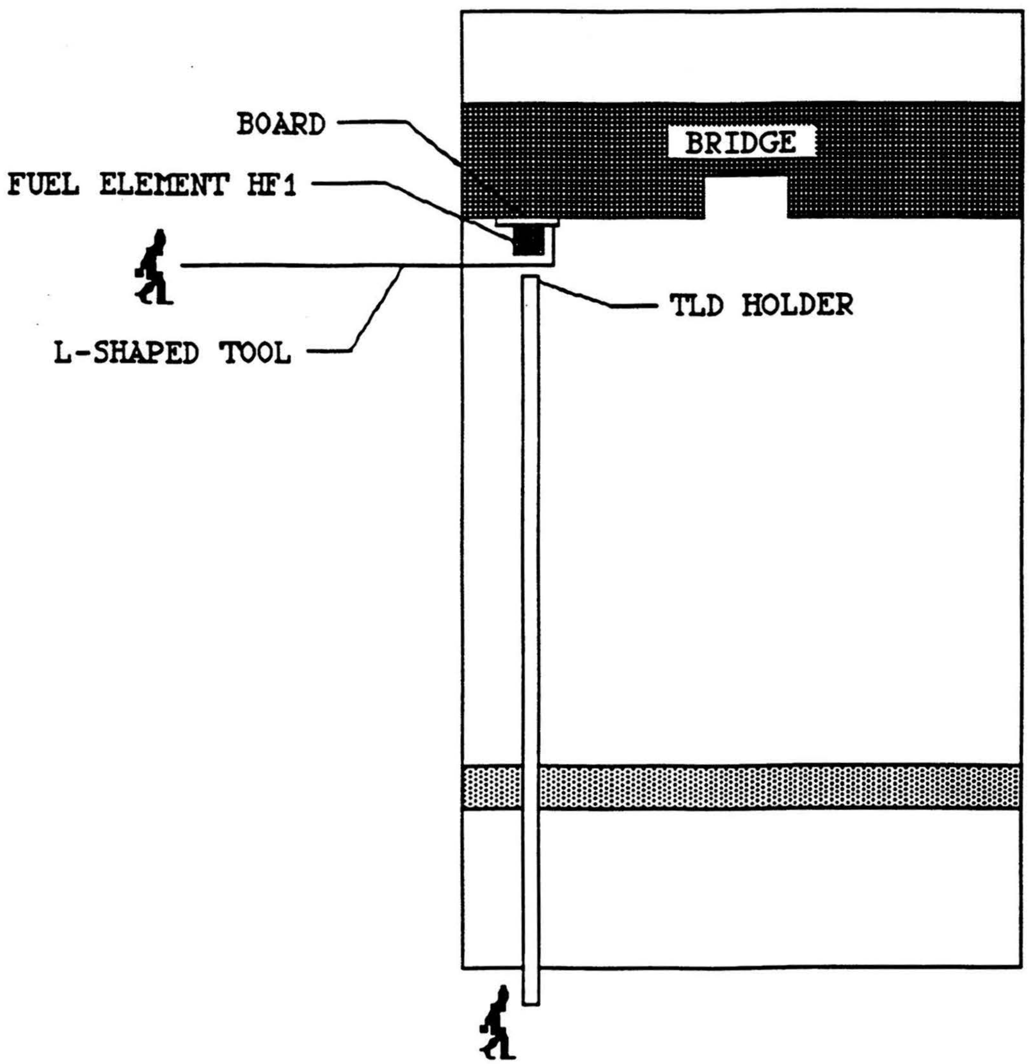

Figure 5. Setup for in-Air Measurement (Top View)

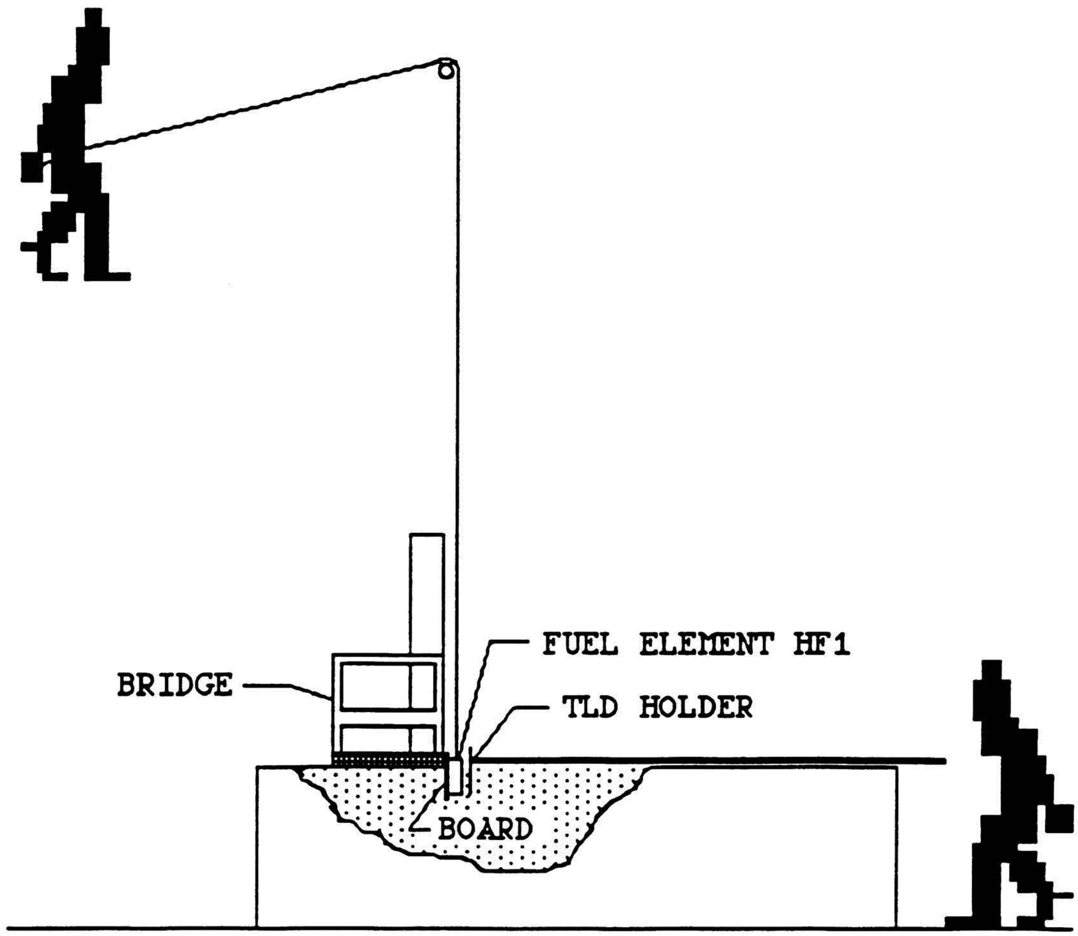

Figure 6. Setup for in-Air Measurement (Side View)

was done remotely. The fuel element was setup on a pulley system to enable it to be moved from three stories up. The person that had to be closest to the fuel was the one that positioned the fuel in line with the TLD holder at a length of approximately six feet from the fuel using an L shaped tool. This took the person only about 5 seconds. The element was taken to the desired height; then it was stabilized against a board with the L-shaped tool. The TLD holder had one inch foam padded spacers affixed to its end so one could just push the TLD holder until the spacers just touched the element. A five minute irradiation was done. The maximum dose received by any person was 0.2 mrem.

For this setup both a horizontal and vertical measurement were done. The dimensions for the distances on the gamma survey device can be seen in Appendix B.

D. SOURCE GEOMETRY MODELS CONSIDERED

Several geometry models were considered in order to determine the dose rate of the element in air at 1 foot and 3 feet. To determine which model is best, the calculated results were compared with the in-air measurements.

1. Cylindrical Geometry Model The first model approximated the fuel as a cylinder with a uniform distribution. The radius of the cylinder was calculated by setting the cross-sectional area of the cylinder equal to the cross-sectional area of the fuel and solving for r. The equation used for this was:(8)

$$\phi = \frac{B\, S_v\, R^2}{2(a + z)} F(\theta,\ b_2) \qquad \text{(eq. 4)}$$

The geometry is depicted in Figure 7 where point P is in the center of the element so $\theta_1 = \theta_2$. There is no intervening shield for this model.

In the above equation the variables are:

ϕ = flux at point P

S_v = the source strength (γ/cm^3-s)

B = buildup factor

R = radius of the cylinder (cm)

a = distance from cylinder to point P

z = self absorption distance

F = secant function(9)

θ = angle (radians)

$b_2 = \mu_s z + \mu_w a$

Using the known dose rates in water, equation 4 is then manipulated to solve for S_v. Once the source strength is known, the same equation above can be used to determine the dose rate in air at 1 foot and 3 feet. Sample calculations of this are shown in Appendix C.

Using the cylindrical model underestimated the strength of the source. For instance, when the dose rate measured in water at 5 inches was used, a given source strength was calculated. This source strength should be greater than the source strength at an inch from the fuel; however, this was not true for most of the elements; therefore, the model was not suitable.

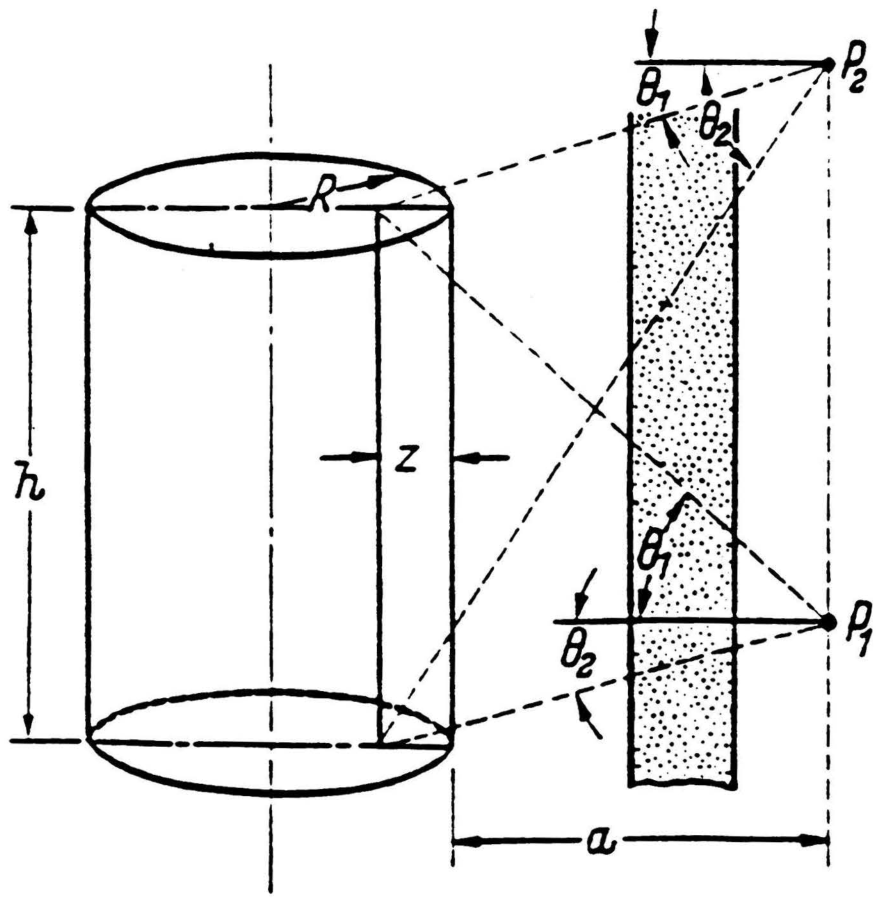

Figure 7. Cylindrical Geometry Model

2. Line Source Model. Next the fuel was approximated as a line source. The equation for approximating the fuel as a line source is as follows:

$$\phi = \frac{B\ S_L}{2\pi h} F(\theta,\ \mu h) \qquad \text{(eq. 5)}$$

The geometry is depicted in Figure 8. Since point P was in the center of the line, the two angles are equal to 45 degrees. In the above equation the variables are:

ϕ = flux at point P

S_L = the source strength (γ/cm-s)

B = buildup factor

h = distance from line to point P

F = secant function(10)

μ = linear attenuation coefficient

t = distance gammas are attenuated a shield(in this case = h in air)

θ = angle (radians)(11)

Using the line source led to the same problem as the cylindrical model. It underestimated the source strength even greater than the cylindrical model.

3. Line Source with a Cosine Distribution Model Finally a line source with a cosine distribution was used. The geometry for this is the same as Figure 7. Point P lies in the center of the line. The equation is as follows:

$$\phi = \frac{S_L}{4\pi h} \int_{-L/2}^{L/2} \cos \frac{\pi x}{L}\ e^{-\mu r}\ (1+a\mu r e^{-\mu b r})dx \qquad \text{(eq. 6)}$$

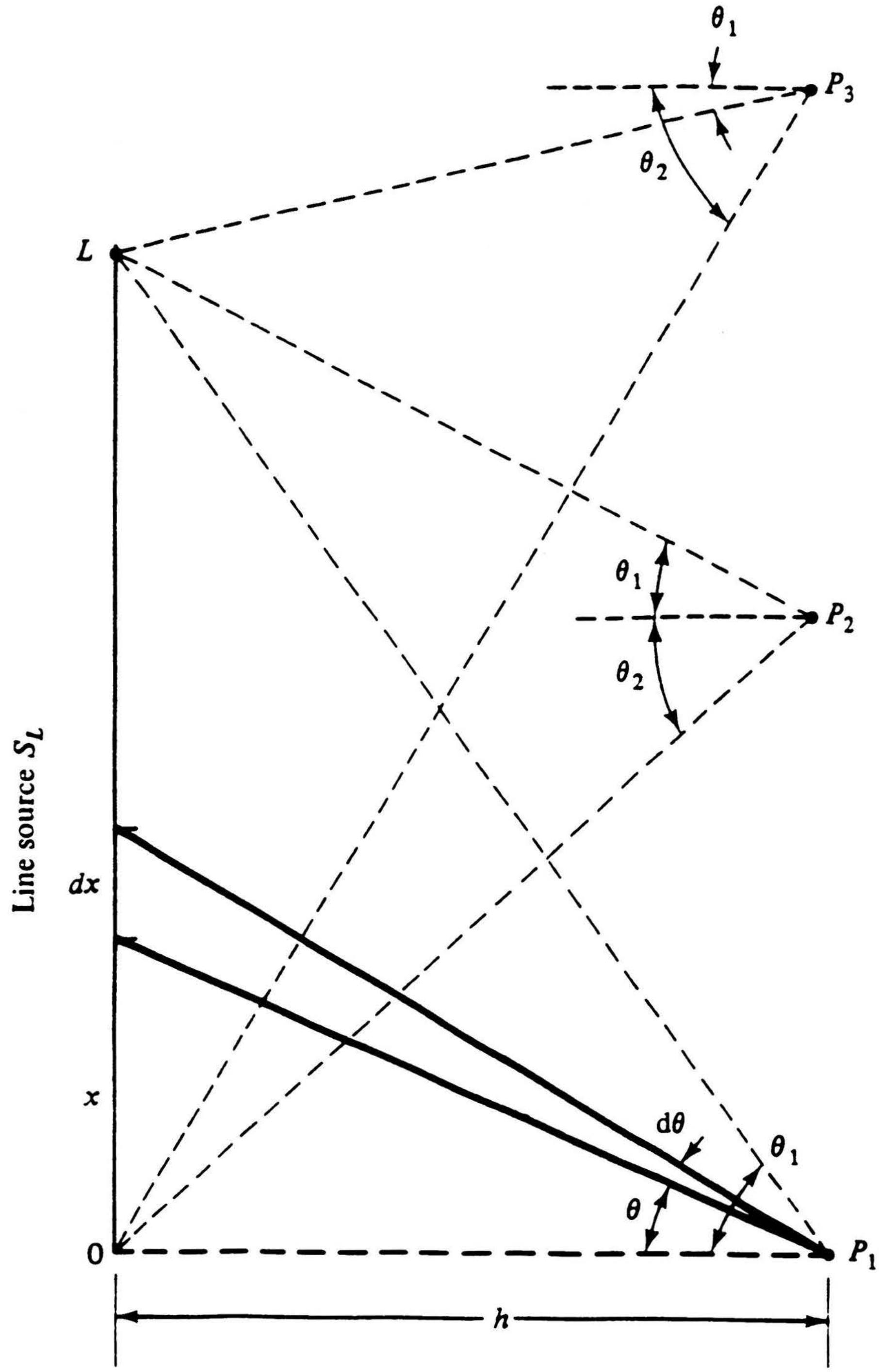

Figure 8. Line Source Geometry

where

$r = h/\cos\theta$

S_v = source strength (γ/cm^2-s)

h = distance from the line to the point of interest

L = the length of the line

x = distance along the line

u = attenuation coefficient (cm^{-1})

a,b = constants for a Berger Buildup Factor (12)

When using a cosine distribution, the ends of the line source are assumed to be zero; however, this is not true in the case of the fuel element. Therefore, the line source is assumed to be longer than the fueled portion of a fuel element actually is, in order to provide a tail for the cosine distribution. Figure 9 shows the measured TLD readings at 7 points along the fuel. The error bars were determined by considering several factors of which will be discussed in more detail in the first section of the results. The active fuel is between 0-24 inches, and the curve is extrapolated to the axis in order to determine the effective length of the fuel. This curve was drawn in with a french curve. (Two centimeters were added to each end.)

Because the integral in equation 6 cannot be easily integrated, a numerical integration technique was used to determine its value. The trapezoidal rule was used.(13)

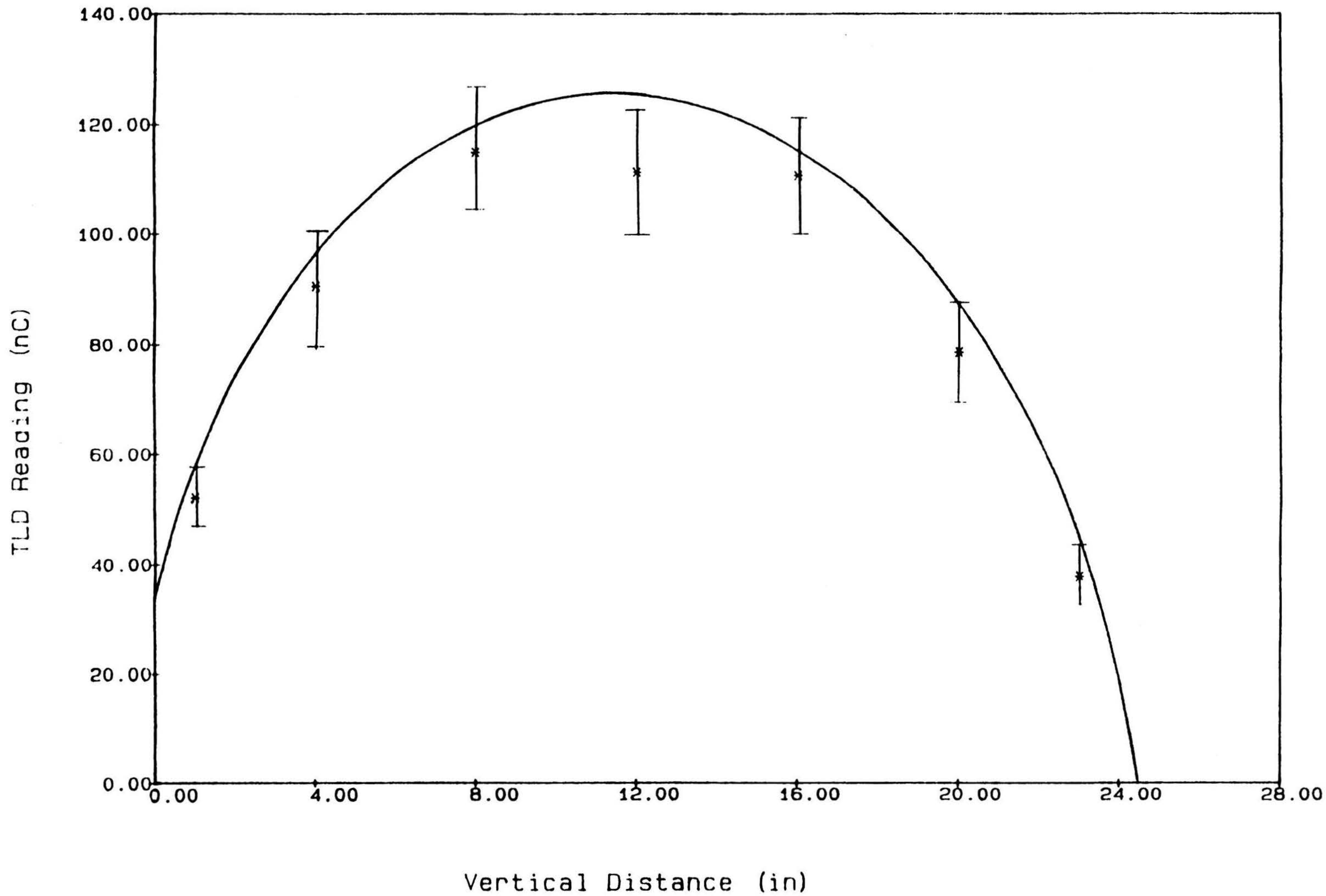

Figure 9. Vertical Flux Profile

III. RESULTS

A. THERMOLUMINESCENT DOSIMETER IN-WATER MEASUREMENTS.

The results from the cylindrical model are shown in Appendix D. Appendix E shows the raw data for all of the experiments. Special notes have been made on every run for any parameters that may effect the outcome of the calculations. The 10% error that is shown was estimated considering various factors. First, the variation in the time that it took to place the element in the right location next to the TLD device took as long as 15 sec at times. This alone contributes a 5% error. Another factor that could contribute to the error is how well the TLD device was lined up with the element. The TLD device is designed so that when it is placed in position R-1 a one inch gap should remain between it and the element. If either the device or element was tilted this distance varied. The tilting was kept to a minimum by careful placing of the TLD holder. Finally, another source of error is the accuracy of the TLDs themselves and the instrument that was used to read them. After considering all these factors an error of 10% was approximated.

Using the line source model with a cosine distribution that was discussed earlier in section IID3, the dose rates for all of the elements except for the rodded elements were calculated. The PASCAL program that was used to calculate the dose rates is given in Appendix F. Table I shows the dose rates calculated at 1 foot and 3 feet in air for each of the elements. The top portion of the table lists the dose rates of elements that were in the fuel storage. The bottom portion lists dose rates in air of the elements that are still in the core.

The lowest dose rate in air was for element HF1 which was

Table I.

Dose Rates Calculated in Air

FUEL ELEMENT NUMBER	DOSE RATE IN AIR AT 1 FT (rem/hr)	DOSE RATE IN AIR AT 3 FT (rem/hr)
	Spent Fuel Elements	
HF1	0.268 +/- 10%	0.088 +/- 10%
HF1 (repeat)	0.268	0.088
F2	1.95	0.648
F2 (repeat)	1.65	0.546
F3	3.43	1.87
F5	2.87	0.953
F13	4.40	1.46
F13 (repeat)	5.32	1.77
F18	2.26	0.75
F20	1.19	0.394
F21	3.18	1.05
F22	2.87	0.953
	In Core Elements	
HR1	9.00 +/- 10%	2.99 +/- 10%
F1	32.1	10.7
F4	41.3	13.7
F6	12.5	4.15
F7	24.0	7.97
F8	32.4	10.8
F8 (repeat)	38.9	12.9
F9	54.7	18.1
F10	28.3	9.39
F11	14.1	4.66
F12	11.1	3.65
F14	32.4	10.8
F15	26.0	8.63
F16	19.9	6.59
F17	21.0	6.95
F19	29.4	9.74

experimentally determined to be 0.268 +/- 0.027 rem/hr at 1 foot. The highest dose rate in air was element F9 which was calculated to be 54.7 +/- 5.47 rem/hr at 1 foot.

Table II shows a comparison of results from this experiment and work done by J. Joel Smith in 1988(14). The ones that are left blank in the second column are the ones Smith had not included. Smith used a cylindrical model similar to the one discussed in section IID1.

For the spare fuel elements only, the results are quite similar. It has been three years since Smith did his measurements, so the current results should be lower because of the time allowed for them to decay. Given the causes of error and the fact that the numbers are quite small to begin with, the similarity of the two sets of data is quite good.

For the elements in the core, a big difference is seen when comparing them to Smith's results. This is probably due to a difference in experimental procedure. Smith irradiated his TLDs with elements from the core by placing the gamma survey device in a corner position in the grid plate of the core and then moving the element adjacent to it. Considering the size of the grid plate, there is no place to put the gamma survey device where sufficient distance can be kept from the rest of the elements and the activated grid plate. It is apparent that Smith's results contained a large contribution from the rest of the core and the grid plate. In comparing some of these results Smith's results are a factor of ten higher.

B. THERMOLUMINESCENT DOSIMETER IN-AIR MEASUREMENTS

Next, an in-air TLD measurement was made to determine how close the adopted model was to the actual in-air readings. As seen in

Table II.

Comparison of Dose Rate Calculations in Air from Netzer & Smith

FUEL ELEMENT NUMBER	DOSE RATE IN AIR AT 1FT A. A. Netzer (R/hr)	DOSE RATE IN AIR AT 1FT J. J. Smith (R/hr)	
	Spent Fuel Elements		
HF1	0.268 +/- 10%	0.495 +/-	0.078
HF1	0.268		
F2	1.95	5.053	0.159
F2	1.65		
F3	3.43	4.997	0.035
F5	2.87	5.436	0.037
F13	4.40	5.876	0.007
F13	5.32		
F18	2.26	3.934	0.045
F20	1.19	2.416	0.056
F21	3.18	2.473	0.056
F22	2.87	3.904	0.056
	In Core Elements		
HR1	9.00 +/- 10%	10.00	N/A
F1	32.1	39.203	1.265
F4	41.3	90.867	0.376
F6	12.5	152.0	5.0
F7	24.0		
F8	32.4	48.157	0.431
F8	38.9		
F9	54.7	61.993	0.929
F10	28.3	34.0	0.20
F11	14.1	122.198	2.542
F12	11.1	120.0	2.0
F14	32.4	33.741	0.174
F15	26.0	212.010	6.668
F16	19.9	247.453	10.571
F17	21.0	152.399	4.811
F19	29.4	426.630	25.548

Table I, the dose rate of element HF1 in air at 3 ft was expected to be 110 mrem/hr according to calculations. When the fuel element was brought up, a portable ion chamber (PIC) was held three feet from the element and it read approximately 100 mrem/hr.

Table III shows the TLD results from the five minute exposure in air. Figure 10 graphically depicts this data. The error bars are shown only on the points where they could be seen based upon the scale of the graph.

Table III.

Measured Dose Rate in Air For HF1.

Distance From Source(in)	Dose Rate (R/hr)
1.0	19.2 +/- 10%
5.0	1.24
9.0	0.351
13.0	0.046

A least squares program was written to determine the straight line that is formed on semi-log paper. The program is shown in Appendix A. The equation determined was:

$$\ln[\text{Dose Rate(rem/hr)}] = -0.348[\text{Distance(in)}] + 2.09 \qquad \text{(eq.7)}$$

At twelve inches from the source the above equation gives a dose rate of 0.124 rem/hr. Using the line source model with a cosine distribution gives a dose rate of 0.320 rem/hr, showing that for this particular element, the model chosen overestimated the dose rate by a factor of two and a half. This agreement is still much better than the

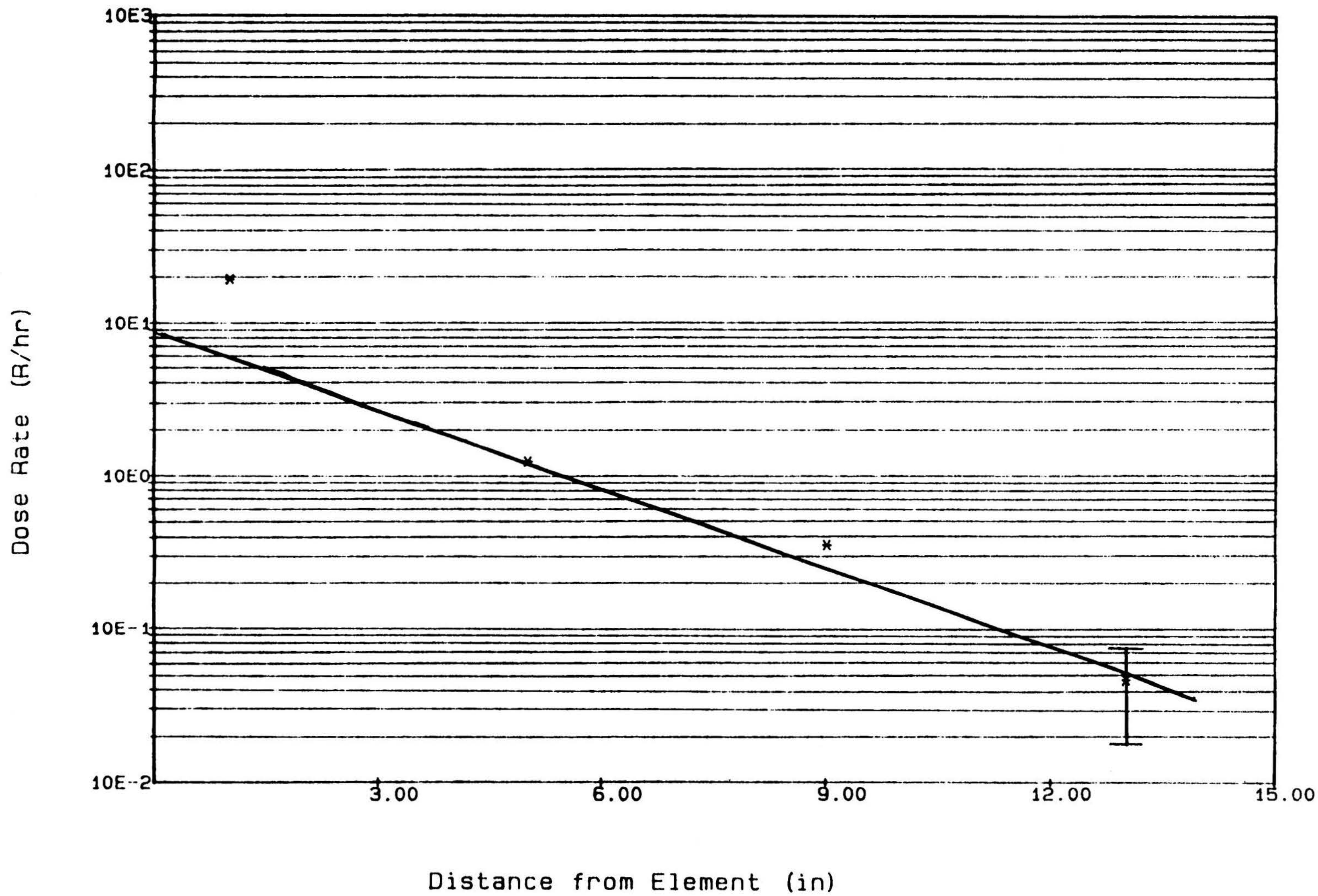

Figure 10. Dose Rate in-Air for Fuel Element HF1

cylindrical model which underestimates each of the dose rates.

Figure 11 shows a comparison between the normalized dose rates in-air that were calculated and the actual measured ones. It was normalized by taking each set of data and dividing it by the dose rate at 5 inches from the element. The circled points are the measured values. They are both quite different for the first point but they were very close for the others. This further shows the validity of the model at the distances that were under consideration.

C. <u>SUGGESTED METHOD OF TRANSPORT OUT OF THE REACTOR.</u>

After determining the dose rates of all the elements at one and three feet in air, it becomes apparent that special care must be taken to transport the fuel out of the reactor pool and building. The following procedure is suggested. First, a person standing on 20 feet high scaffolding will transfer a fuel element in air from the pool to a fifty-five gallon drum that is on a fork lift at the edge of the pool. This fifty-five gallon drum will already be partially filled with water. Use the fork lift to transport the drum to the door where it will be loaded into the shipping cask by use of scaffolding. A fifty-five gallon drum full of water weighs less than 600 lbs. This number is considerably less than the transfer cask with a weight of 4650 lbs. By using a fork lift the weight will be evenly distributed by the wheels of the fork lift.

Table IV shows the dose rate on the outside of a fifty-five gallon drum flooded with water for each of the elements. Fuel element F-9 is the highest at 28.6 rem/hr. This is somewhat higher than a radiation worker should handle unless absolutely necessary. If, however, the 12 elements with the lowest dose rates were removed at the time of fuel

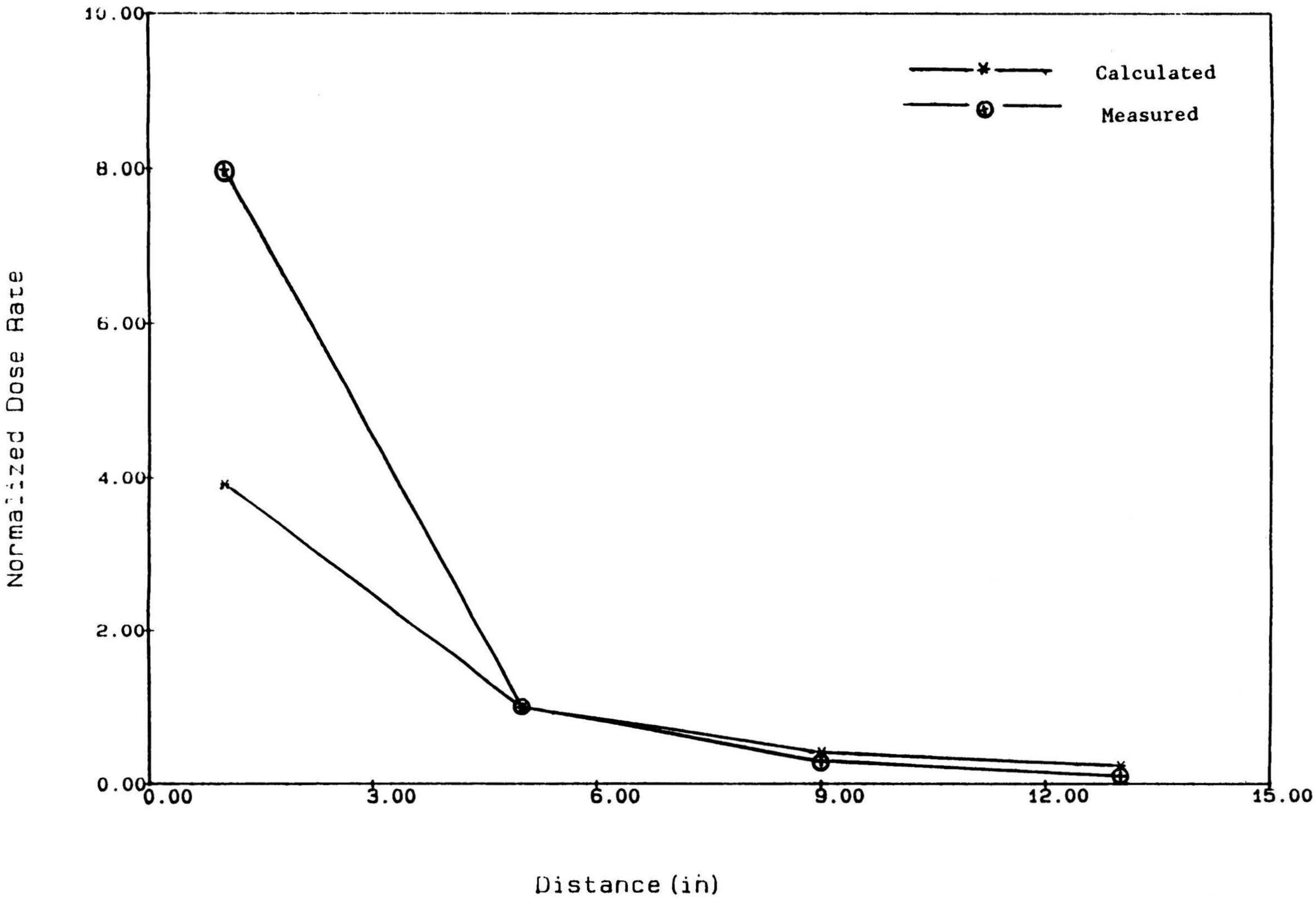

Figure 11. Measured and Calculated Normalized Dose Rates in air for FE HF1

Table IV.

Dose Rates on the Surface of a 55 Gal Drum Filled with Water.

FUEL ELEMENT NUMBER	DOSE RATE AT 1 FOOT IN WATER (R/hr)
HF1	0.140 +/- 10%
HF1	0.140
F2	1.02
F2	0.861
F3	1.82
F5	1.50
F13	2.30
F13	2.78
F18	1.18
F20	0.621
F21	1.66
F22	1.50
HR1	4.71
F1	1.68
F4	21.6
F6	6.55
F7	12.6
F8	17.0
F8	20.3
F9	28.6
F10	14.8
F11	7.35
F12	5.75
F14	17.0
F15	13.6
F16	10.4
F17	11.0
F19	15.4

conversion, the highest dose rate of these 12 would be F-12 which would have a dose rate of 5.75 rem/hr on the outside of the fifty-five gallon drum. If a person was transferring it on scaffolding that was twenty feet high, he/she would receive only 0.524 rem/hr. No one should be in the bay when this element is moved except for the person on the scaffolding, the one driving the fork lift, and the health physicist. Even then extra shielding should be placed between them and the source.

The other 16 elements can be stored in the fuel storage area and checked with TLDs after one year to determine if enough decay has occurred to make them safe for transport. Considering that the shipping cask will be limited in size and that it will probably have to broken down into two or three shipments anyway, this seems like a solution that would save time ,money and radiation exposure. This procedure would also leave ample room in the fuel storage pool because there would still be 18 empty slots.

One suggestion for further study would be to run the ORIGEN code to determine what the dose rates would be after the fuel elements have decayed for one year in the fuel storage pool.

IV. CONCLUSIONS

The dose rates of all of the fuel elements in the University of Missouri- Rolla Reactor were calculated at one foot and three feet in air based upon experimental measurements taken in water. The element with the highest dose rate is fuel element F9 which registers 54.7 rem/hr +/- 10% at one foot and 18.1 rem/hr +/- 10% at three feet. With this knowledge, it is recommended that the fuel be removed in two separate shipments: one at the time of fuel conversion and another after a year or more to allow the radioactive fuel elements to decay. They could be rechecked with the TLDs.

This research will be of great assistance to the reactor staff because prior to it, they thought that the element with the highest dose rate was 426.630 rem/hr at one foot; however, this research shows that the highest one at one foot is 54.7 rem/hr. Since a computer model has been set up it is possible to determine the dose rate of the fuel at any time given a TLD reading at a known distance. The same computer model can also be used for the low-enriched uranium fuel, if some of the parameters are properly changed.

APPENDIX A

LEAST SQUARES PROGRAM

APPENDIX A

LEAST SQUARES PROGRAM

```
Type

        svector= array[1..20] of real;



Var

         i,n:  integer;

         yint,slope,d,sumw,sumwxx,sumwx,sumwy,sumwxy:  real;

         x,y,w,yy: svector;

Begin

Writeln ('How many data points do you have?');

Readln (n);

writeln ('Enter X(i) and then Y(i) for all of your points now?');


sumw:= 0.0;

sumwxx:= 0.0;

sumwx:= 0.0;

sumwy:= n;

sumwxy:= 0.0;


For i:= 1 to n do

Begin

  readln (x[i]);

  readln (y[i]);

  w[i]:= 1/y[i];

  sumw:= sumw + w[i];
```

```
  sumwxx:= sumwxx + (w[i]*x[i]*x[i]);
  sumwx:= sumwx + w[i]*x[i];
  sumwxy:= sumwxy + w[i]*x[i]*y[i];
End; {for }


d:=sumw*sumwxx - sumwx*sumwx;
yint:= (sumwy*sumwxx - sumwxy*sumwx)/d;
slope:= (sumw*sumwxy - sumwx*sumwy)/d;
writeln ('The equation for this set of points is:');
writeln (' ');
yint:=(yint);
writeln ('y=',slope,'x','+',yint);
{writeln ('y=',yint,'e**',slope,'x');}
end.
```

APPENDIX B

GAMMA SURVEY DEVICE DRAWINGS

APPENDIX B

GAMMA SURVEY DEVICE DRAWINGS

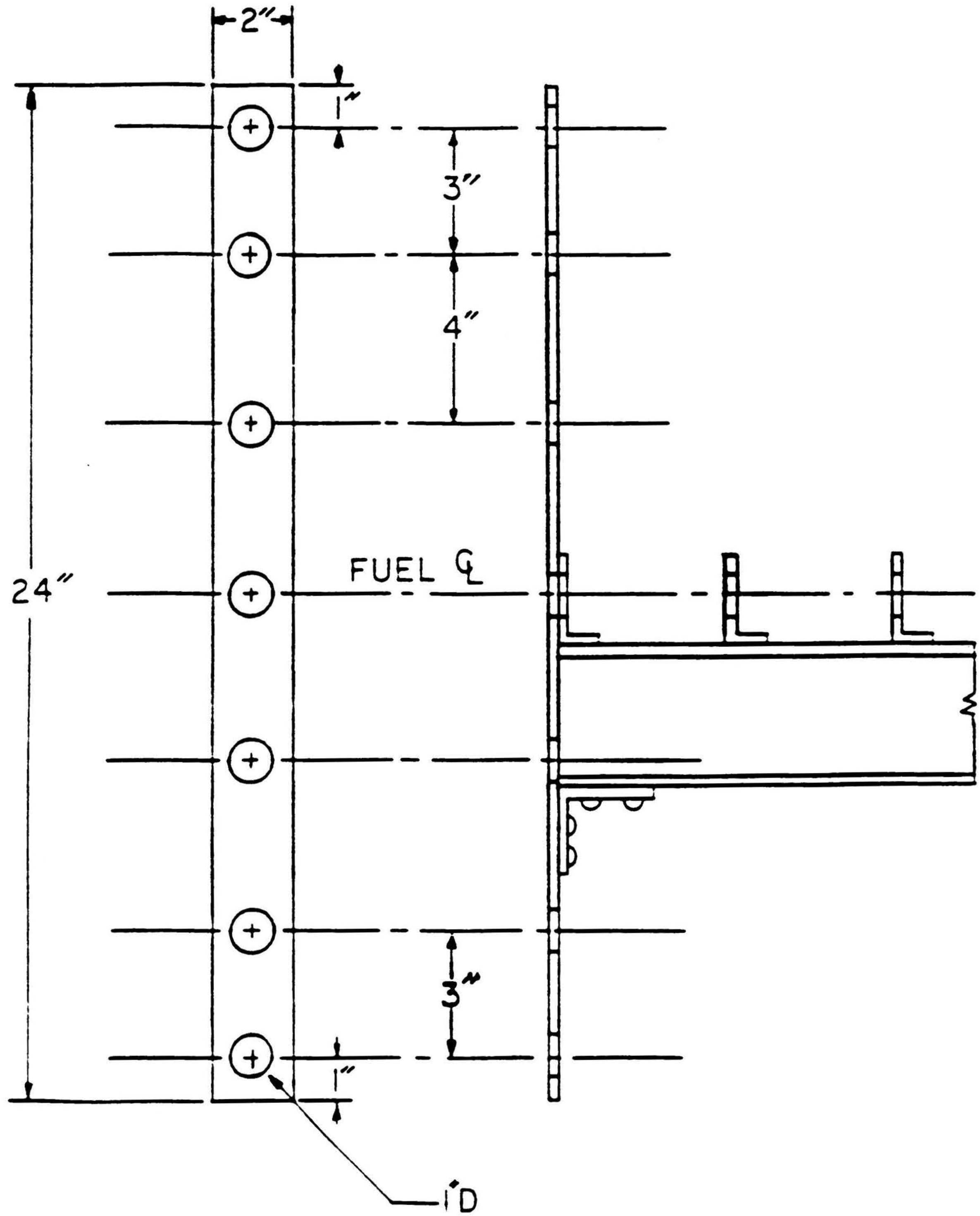

Figure B1. Front and Side View of Fuel Survey Device

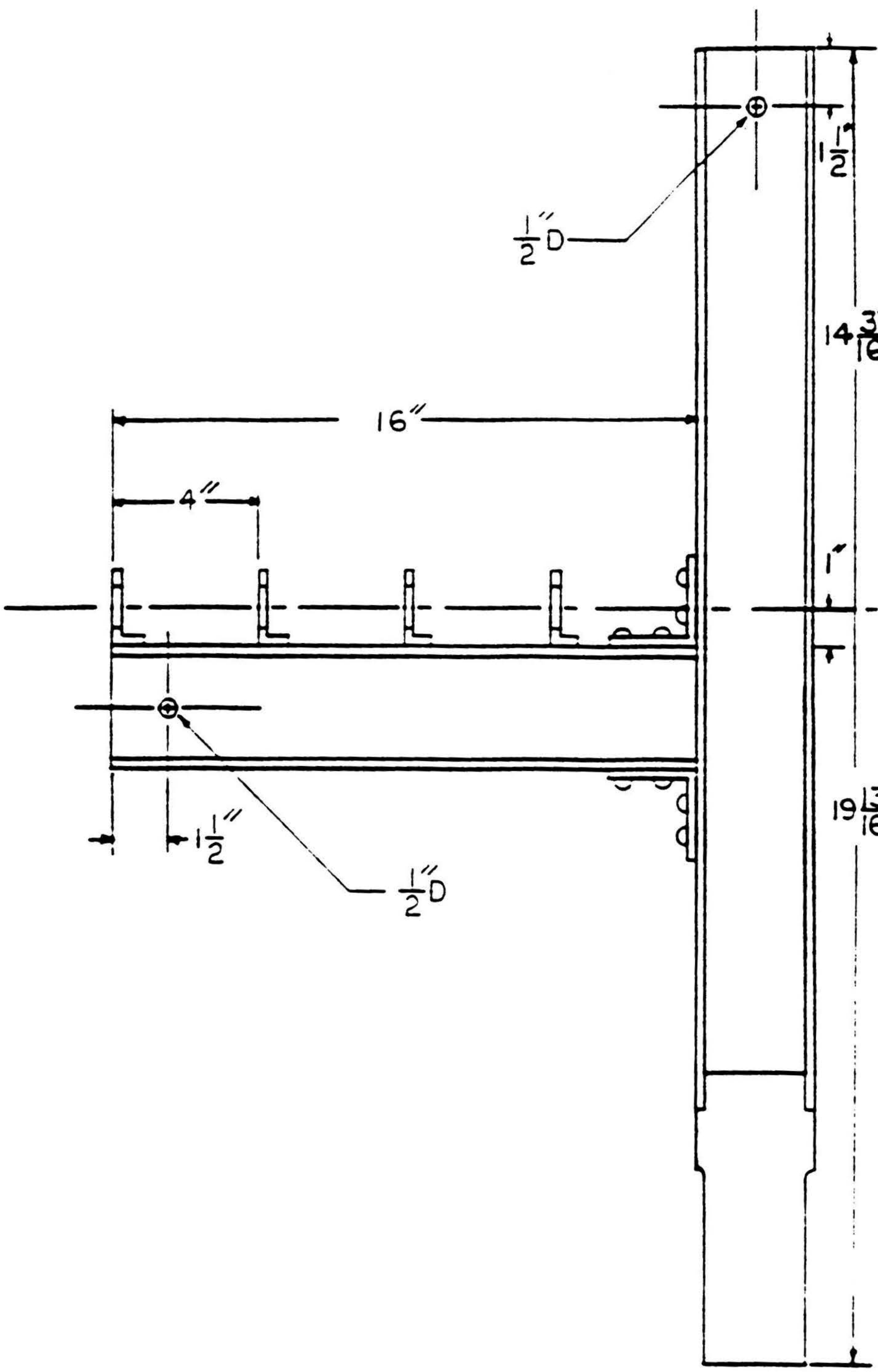

Figure B2. Side View of Fuel Survey Device

APPENDIX C

SAMPLE CALCULATIONS USING CYLINDRICAL GEOMETRY

APPENDIX C

SAMPLE CALCULATIONS USING CYLINDRICAL GEOMETRY

The general equation is as follows:

$$\phi = \frac{B\ S_v\ R^2}{2(a + z)}\ F(\theta,\ b_2)$$

Find b_2 wh
ere $b_2 = \mu_s z + b_1$

a = 5 inches or 12.7 cm

b_1 = shield attenuation length = $\mu_w a = 0.0706*12.7 = 0.90$

μ_s = source attenuation

The source is primarily made up of a mixture of water and aluminum. The following shows how to calculate the linear attenuation coefficient for the mixture.

$$(\mu/\rho)_{mix} = \text{weight }\%(\mu/\rho)_{Al} + \text{weight }\%(\mu o/\rho)_{water}$$

$$\mu/1.407 = 0.514*0.0614 + 0.486 * 0.0706$$

$$\mu_s = 0.0922\ cm^{-1}$$

The effective radius turns out to be 4.39 cm by setting the cross-sectional area of the element equal to the cross-sectional area of the cylinder. The length of the element is kept constant.

$$a/R = 12.7/4.39 = 2.89$$

Therefore, from Figure 6.4-12a in the Engineering Compendium

$$\mu_s z/m = 1.05$$

$$\mu_s(R + a) = 1.58$$

From Figure 6.4-12b m = 0.3

From the above equation z = 3.42

$b_2 = 0.0922*3.42 + 0.90 = 1.2$

From p. 414 Chilton

$F(\theta, b_2) = F(1.176, 1.215) = 0.2463$

B= 1.958 from p.446 Chilton

$$S_v = \frac{\varphi 2(a + z)}{BR^2 F(\theta, b_2)}$$

$$S_v = \frac{\varphi 2(12.7 + 4.38)}{1.958(1.73*2.54)^2\ 0.246}$$

$S_v = 3.67\varphi$ where

φ = Dose Rate (R/hr)/Conversion Factor

Conversion Factor = 1.98×10^{-6} (Rem/hr)/(γ/cm^2-s)

Using the surface source strength that was calculated, a similar calculation is done to determine the dose rate at 1 ft and 3 ft in air.

$$\phi_{air} = \frac{B\ S_v\ R^2}{2(a + z)}\ F(\theta, b_2)$$

Find b_2 where $b_2 = \mu_s + b_1$

The values are determined as they were above and

$b_2 = 0.3044$

$F(\theta, b_2) = 0.588$

$\varphi_{air} = 0.16\ S_v$ at 1 foot from the source in air.

The calculations are done the same for three feet and

$\varphi_{air} = 0.02389\ S_v$ at 3feet from the source in air.

APPENDIX D

RESULTS FROM CYLINDRICAL GEOMETRY MODEL

APPENDIX D

RESULTS FROM CYLINDRICAL GEOMETRY MODEL

Table V.

Dose Rate Calculations Using Cylindrical Model

FUEL ELEMENT NUMBER	DOSE RATE AT 1 FT (R/hr)	DOSE RATE AT 3 FT (R/hr)
	Spare Fuel Elements	
HF1	0.478 +/- 10%	0.0713 +/- 10%
HF1	0.332	0.0496
F2	2.46	0.368
F2	2.00	0.299
F3	4.29	0.641
F5	3.52	0.525
F13	5.34	7.98
F13	6.42	0.958
F18	2.72	0.407
F20	1.55	2.31
F21	3.84	5.74
F22	3.54	0.529
	In Core Elements	
HR1	11.0 +/- 10%	1.64 +/- 10%
F1	38.8	5.80
F4	49.9	7.45
F6	15.2	2.27
F7	28.9	4.32
F8	46.9	7.00
F8	39.2	5.86
F9	66.0	9.85
F10	34.2	5.10
F11	16.9	2.53
F12	13.3	1.99
F14	39.2	5.85
F15	31.5	4.70
F16	2.40	3.58
F17	25.4	3.79
F19	35.5	5.30

APPENDIX E

RAW DATA

APPENDIX E

RAW DATA

The following is all of the raw data that was determined from experimentation. The distances from the elements are in inches. R1, R2, and R3 are the three TLD readings at each point. They are all in nanoCoulombs.

Date of Irradiation: 4-26-91
Approximate Seat Time: 15 sec
Fuel Element: F13

Distance from Element	R1	R2	R3
1.0	119.0	125.0	122.0
5.0	29.2	29.8	31.6
9.0	10.0	10.1	9.902
13.0	3.927	3.821	3.853

Date of Irradiation: 4-26-91
Approximate Seat Time: 5 sec
Fuel Element: F20

Distance from Element	R1	R2	R3
1.0	40.1	41.0	42.4
5.0	9.350	9.591	9.937
9.0	3.756	3.771	3.266
13.0	1.915	1.711	1.707

Date of Irradiation: 4-26-91
Approximate Seat Time: 30 sec
Fuel Element: F21

Distance from Element	R1	R2	R3
1.0	98.2	100.4	100.9
5.0	21.9	21.8	22.5
9.0	7.427	7.339	7.700
13.0	2.945	3.062	3.009

Date of Irradiation: 4-26-91
Approximate Seat Time: 5 sec
Fuel Element: F13

Distance from Element	R1	R2	R3
1.0	157.0	166.4	171.5
5.0	35.6	36.9	35.5
9.0	11.7	11.9	11.3
13.0	4.312	4.455	4.624

Date of Irradiation: 6-04-91
Approximate Seat Time: 5 - 10 sec
Fuel Element: HR1

Distance from Element	R1	R2	R3
1.0	229.3	223.5	223.1
5.0	60.8	62.7	58.8
9.0	21.6	22.5	21.5
13.0	7.877	8.554	8.696

Date of Irradiation: 6-04-91
Approximate Seat Time: 10 - 20 sec
Fuel Element: F14

Distance from Element	R1	R2	R3
1.0	885.3	909.6	866.9
5.0	217.3	209.7	213.6
9.0	68.3	71.2	70.5
13.0	27.9	25.0	26.2

Date of Irradiation: 6-04-91
Approximate Seat Time: <5 sec
Fuel Element: F1

Distance from Element	R1	R2	R3
1.0	947.5	911.7	901.6
5.0	214.5	197.7	222.7
9.0	73.2	72.7	72.6
13.0	23.8	28.1	28.0

Date of Irradiation: 6-04-91
Approximate Seat Time: 5 - 10 sec
Fuel Element: F8

Distance from Element	R1	R2	R3
1.0	848.6	825.2	887.1
5.0	210.5	212.5	218.0
9.0	71.7	75.0	73.7
13.0	26.6	27.3	26.2

Date of Irradiation: 6-06-91
Approximate Seat Time: 5 -10 sec
Fuel Element: F9

Distance from Element	R1	R2	R3
1.0	1170.0	1290.0	1280.0
5.0	358.3	359.2	366.8
9.0	131.7	132.2	135.6
13.0	55.1	59.1	---

Date of Irradiation: 6-06-91
Approximate Seat Time: 5 - 10 sec
Fuel Element: F4

Distance from Element	R1	R2	R3
1.0	1090.0	1100.0	1130.0
5.0	267.0	268.7	279.0
9.0	95.1	84.1	89.4
13.0	38.3	32.7	30.8

Date of Irradiation: 6-06-91
Approximate Seat Time: 5 - 10 sec
Fuel Element: F10

Distance from Element	R1	R2	R3
1.0	722.7	712.1	758.8
5.0	191.4	182.9	184.9
9.0	56.6	54.9	63.3
13.0	23.9	21.5	21.7

Date of Irradiation: 6-06-91
Approximate Seat Time: 5 - 10 sec
Fuel Element: F8

Distance from Element	R1	R2	R3
1.0	847.5	947.2	817.4
5.0	293.7	254.9	217.2
9.0	88.3	78.8	72.9
13.0	25.7	25.1	26.7

Date of Irradiation: 6-24-91
Approximate Seat Time: 3 sec
Fuel Element: HF1

Distance from Element	R1	R2	R3
1.0	9.316	9.295	8.500
5.0	2.396	2.344	2.271
9.0	0.908	0.937	0.950
13.0	0.568	0.479	0.503

Date of Irradiation: 6-24-91
Approximate Seat Time: 5 sec
Fuel Element: F22

Distance from Element	R1	R2	R3
1.0	84.0	79.9	84.4
5.0	20.4	20.2	20.7
9.0	6.855	6.618	6.681
13.0	2.654	2.591	2.469

Date of Irradiation: 6-24-91
Approximate Seat Time: 5 sec
Fuel Element: F2

Distance from Element	R1	R2	R3
1.0	60.7	60.7	52.4
5.0	13.7	15.9	14.3
9.0	4.428	5.289	5.506
13.0	1.737	2.009	---

Date of Irradiation: 6-24-91
Approximate Seat Time: 5 sec
Fuel Element: F5

Distance from Element	R1	R2	R3
1.0	69.3	81.1	70.7
5.0	21.0	19.9	19.9
9.0	6.700	6.795	6.618
13.0	2.637	2.519	2.607

Date of Irradiation: 6-26-91
Approximate Seat Time: <5 sec
Fuel Element: F3

Distance from Element	R1	R2	R3
1.0	104.9	99.7	103.1
5.0	23.5	24.5	25.6
9.0	7.772	7.787	8.175
13.0	2.942	2.984	3.047

Date of Irradiation: 6-26-91
Approximate Seat Time: <5 sec
Fuel Element: F18

Distance from Element	R1	R2	R3
1.0	71.8	75.7	72.4
5.0	15.4	16.4	16.2
9.0	4.884	5.351	5.254
13.0	2.155	2.089	1.991

Date of Irradiation: 6-26-91
Approximate Seat Time: 5 - 10 sec
Fuel Element: F2

Distance from Element	R1	R2	R3
1.0	40.4	54.2	43.5
5.0	11.8	12.1	12.4
9.0	4.031	4.390	3.899
13.0	1.204	1.780	1.595

Date of Irradiation: 6-26-91
Approximate Seat Time: 5 sec
Fuel Element: HF1

Distance from Element	R1	R2	R3
1.0	6.522	7.963	6.248
5.0	1.885	1.647	1.905
9.0	0.722	0.795	0.786
13.0	0.466	0.454	0.459

Date of Irradiation: 6-28-91
Approximate Seat Time: 10 sec
Fuel Element: F11

Distance from Element	R1	R2	R3
1.0	406.8	428.5	408.5
5.0	94.9	94.8	89.4
9.0	31.8	31.9	31.4
13.0	11.8	12.0	12.0

Date of Irradiation: 6-28-91
Approximate Seat Time: 5 sec
Fuel Element: F12

Distance from Element	R1	R2	R3
1.0	488.9	290.4	337.7
5.0	71.3	80.2	69.1
9.0	24.7	24.3	35.1
13.0	13.2	10.4	12.6

Date of Irradiation: 6-28-91
Approximate Seat Time: 5 sec
Fuel Element: F6

Distance from Element	R1	R2	R3
1.0	360.3	353.2	354.8
5.0	88.7	81.1	81.0
9.0	27.3	27.6	28.3
13.0	9.823	10.7	9.782

Date of Irradiation: 6-28-91
Approximate Seat Time: 2 - 3 sec
Fuel Element: F16

Distance from Element	R1	R2	R3
1.0	526.4	537.2	539.7
5.0	127.2	133.4	133.2
9.0	42.2	47.6	42.4
13.0	17.0	16.4	16.2

Date of Irradiation: 7-08-91
Approximate Seat Time: 5 sec
Fuel Element: F7

Distance from Element	R1	R2	R3
1.0	677.0	685.0	674.7
5.0	155.4	162.0	156.7
9.0	52.5	52.2	57.0
13.0	19.2	18.8	20.8

Date of Irradiation: 7-08-91
Approximate Seat Time: 4 sec
Fuel Element: F15

Distance from Element	R1	R2	R3
1.0	720.3	706.9	684.2
5.0	168.7	173.2	173.3
9.0	50.1	54.0	55.5
13.0	21.5	19.5	20.8

Date of Irradiation: 7-08-91
Approximate Seat Time: 5 sec
Fuel Element: F19

Distance from Element	R1	R2	R3
1.0	723.7	785.1	784.0
5.0	197.6	188.2	195.4
9.0	67.5	64.6	61.8
13.0	25.3	25.5	24.7

Date of Irradiation: 7-08-91
Approximate Seat Time: <5 sec
Fuel Element: F17

Distance from Element	R1	R2	R3
1.0	610.1	603.6	594.8
5.0	146.0	133.0	137.7
9.0	45.8	45.1	46.1
13.0	17.1	17.6	15.8

The following is the in-air measurements.

Date of Irradiation: 7-16-91
Fuel Element: HF1

Distance from Element	R1	R2	R3
1.0	56.1	64.8	66.3
5.0	5.327	4.825	5.412
9.0	2.272	2.443	2.389
13.0	1.449	1.334	1.417

APPENDIX F

PROGRAM CALCULATING DOSE RATES IN AIR

APPENDIX F

PROGRAM CALCULATING DOSE RATES IN AIR

```
Program Trapezoidal(input,output);


{ This program will calculate dose rates of fuel elements in air when
given the TLD reading at a given distance in water. The Trapezoidal
Rule is used to solve the integral numerically.}


Const   mu=0.0706;{linear attenuation coefficient in water-1MeV}
        mua=8.2234e-5;{ linear attenuation coefficient in air-1MeV}
            l=60.96;{the length of the fuel element}



Type     Vector= array[1..1000]  of real;
         sivector= array[1..100] of integer;



Var     y,x,yair,phi,s,dose:  vector;
        fen,tld: sivector;
        i,n,c,j:  integer;
        int,intair,ssumair,ssum,sum,sumair,ya,yaair,yb,ybair,
        h,inc,a,b,d,doseair,din,hin:   real;
```

```
{***************************INPUT*******************************}
Begin
writeln ('How many different elements do you wish to determine');
writeln ('the dose rate on?');
readln  (c);
writeln ('Enter the fuel element number,return, and then');
Writeln ('the TLD reading in nC.');
For j:=1 to c do
begin
   readln (fen[j]);
   Readln (tld[j]);
   phi[j]:=12*((26.2*tld[j])-32.8)/1000/1.98e-6;
end;
Writeln ('How far is P from the line source in water(in)?');
Readln (hin);
h:=hin*2.54;


a:= -(1+6)/2;
b:=(1+6)/2;


Writeln ('Enter the number increments.');
Readln (n);
```

```
Writeln ('How far from the line source do you wish to calculate');
Writeln ('the in air reading?(in)');
Readln (din);
d:=din*2.54;

{*********************NUMERICAL INTEGRATION******************}
ssum:= 0.0;
inc:= (b-a)/n;

For i:=1 to n-1 do
Begin
   x[i]:=a + i*inc;
   y[i]:= cos(3.1416*x[i]/l)*exp(-mu*sqrt(x[i]*x[i] + h*h))*
             (1+1.51*mu*h*exp(0.035*mu*h));
   ssum:= ssum + 2*y[i];
   yair[i]:= cos(3.1416*x[i]/l)*exp(-mua*sqrt(x[i]*x[i] + d*d))*
                (1+1.51*mua*d*exp(0.035*mua*d));
   ssumair:= ssumair + 2*yair[i];
End;

ya:= cos(3.1416*a/l)*exp(-mu*sqrt(a*a + h*h));
yb:= cos(3.1416*b/l)*exp(-mu*sqrt(b*b + h*h));
sum:= ssum + ya + yb;
yaair:= cos(3.1416*a/l)*exp(-mua*sqrt(a*a + d*d));
ybair:= cos(3.1416*b/l)*exp(-mua*sqrt(b*b + d*d));
sumair:= ssumair + yaair + ybair;
int:=  (inc/2)*sum;{the integral is equal to this}
```

```
intair:= (inc/2)*sumair;
{****************************OUTPUT*******************************}
writeln ('The following numbers are for ',din,' inches from the source.');
   writeln ('Fuel El #      Surface Dose(R/hr)    Dose in Air(r/hr)');
for j:= 1 to c do
begin
   s[j]:=phi[j]*4.0*3.14159*h/int;
   {writeln ('The source strength is',s[j],'#/cm3');}
   dose[j]:= s[j]*1.98e-6;
   doseair:=intair*dose[j]/(4.0*3.14159*d);
   writeln ( fen[j],'           ',dose[j],'              ',doseair);
end;


end.
```

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VITA

Alice Ann Netzer was born April 23,1967 in Springfield, Missouri. She received her primary and secondary education in Everton and Springfield, Missouri. In the fall of 1985, she began her college education at Southwest Missouri State University which she attended for two years enrolled in the Pre-Engineering program. In the fall of 1987 she transferred to the University of Missouri- Rolla. In May of 1989 she was awarded a Bachelor of Science degree in Nuclear Engineering.

She has been enrolled in graduate school at the University of Missouri - Rolla since August of 1989. While enrolled, Ms. Netzer was a Power Fellow and assisted with the senior reactor laboratory courses. Ms. Netzer is a member of Alpha Nu Sigma and Tau Beta Pi honor societies.

She was married to Philip Simpkins on August 10,1991.