ANCR-1077

N72 33651

NASA CR-120903

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GAMMA HEATING IN REFLECTOR HEAT SHIELD OF GAS CORE REACTOR

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Prepared for
NATIONAL AERONAUTICS AND SPACE ADMINISTRATION
Contract C-54558B

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DATE PUBLISHED–AUGUST 1972

Aerojet Nuclear Company
NATIONAL REACTOR TESTING STATION
Idaho Falls, Idaho 83401

Printed in the United States of America
Available from
National Technical Information Service
U. S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22151
Price: Printed Copy $3.00; Microfiche $0.95

LEGAL NOTICE

1. Report No.
NASA CR-120903

2. Government Accession No.

3. Recipient's Catalog No.

4. Title and Subtitle

GAMMA HEATING IN REFLECTOR HEAT SHIELD OF GAS CORE REACTOR

5. Report Date
August 1972

6. Performing Organization Code

7. Author(s)

J. H. Lofthouse, J. F, Kunze, T. E. Young, and R. C. Young

8. Performing Organization Report No.
ANCR-1077

10. Work Unit No.

9. Performing Organization Name and Address

Aerojet Nuclear Company
Idaho Falls, Idaho 83401

11. Contract or Grant No.
C-54558B

13. Type of Report and Period Covered
Contractor Report

12. Sponsoring Agency Name and Address

National Aeronautics and Space Administration
Washington, D. C. 20546

14. Sponsoring Agency Code

15. Supplementary Notes

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16. Abstract

Heating rate measurements made in a mock-up of a BeO heat shield for a gas core nuclear rocket engine yields results nominally a factor of two greater than calculated by two different methods. The disparity is thought to be caused by errors in neutron capture cross sections and gamma spectra from the low cross-section elements D, O, and Be.

17. Key Words (Suggested by Author(s))

Gamma Heating Rate
Gas Core Reactor
BeO Reflector

18. Distribution Statement

Unclassified - Unlimited

19. Security Classif. (of this report)	20. Security Classif. (of this page)	21. No. of Pages	22. Price*
Unclassified	Unclassified		$3.00

* For sale by the National Technical Information Service, Springfield, Virginia 22151

ACKNOWLEDGEMENTS

Acknowledgement is made for the important contributions of C. G. Cooper, P. L. Chase, R. R. Jones and D. H. Suckling for their support in reactor operations and data reduction. Also, acknowledgement is made to Mr. J. P. Cusimano of the AEC and his staff at the NRTS Health and Safety Laboratory for their contribution in calibrating and processing the thermoluminescent dosimeters and assisting in the interpretation of the data.

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1.0 SUMMARY

Gamma ray heating rates measured in a block of beryllium oxide located in the reflector of the Spherical Gas Core Critical Experiment gave results nominally a factor of two higher than those calculated. A transport code and a point kernel shielding code were used for the calculations. Fair agreement was obtained between the two calculational methods with the point kernel shielding code yielding results 8 to 10% higher than the transport code. Both calculations utilized nominally the same photon reaction cross section data in 18 energy group detail.

Heating rate distribution through the BeO block was measured with LiF thermoluminescent dosimeters (TLD) highly depleted in Li^6 such that the manufacturers quoted Li^6 contamination was nominally 0.005%. Even though the Li^6 contamination and the Li^7 and F thermal cross sections appear small, the TLD response from thermal neutrons is appreciable and a correction was made using thermal neutron flux values measured through the block. Use of a small beryllium Bragg-Gray chamber was utilized as a calibration check on the TLD's. The results agreed to within 11%. Thus two methods of measurement in fair agreement yield results nominally a factor of two larger than results from two methods of calculation, also in fair agreement.

2.0 INTRODUCTION

The gamma ray heating rate measurements described in this report were conducted as an extension of tests previously performed with the Spherical Cavity Reactor Critical Experiment. A detailed description of the previous experiments is given in Reference 1. All of these experiments have been performed in support of the gas core nuclear rocket concept. This concept envisions an optimized reflector-moderator system consisting of an arrangement of annuli of heavy water and beryllium oxide moderators as shown in Figure 2.1

The experiment reported in this document was performed with rectangular blocks (of various thicknesses) of BeO suspended in the D_2O reflector moderator tank of the Critical Experiment to "mock-up" a section of heat shield in the reflector.

Energy deposition rate measurements were made through the BeO slab with LiF thermoluminescent dosimeters (TLD). Beryllium Bragg-Gray chamber measurements were obtained in an aluminum tube which penetrates the block and these measurements were correlated with TLD measurements at the same location.

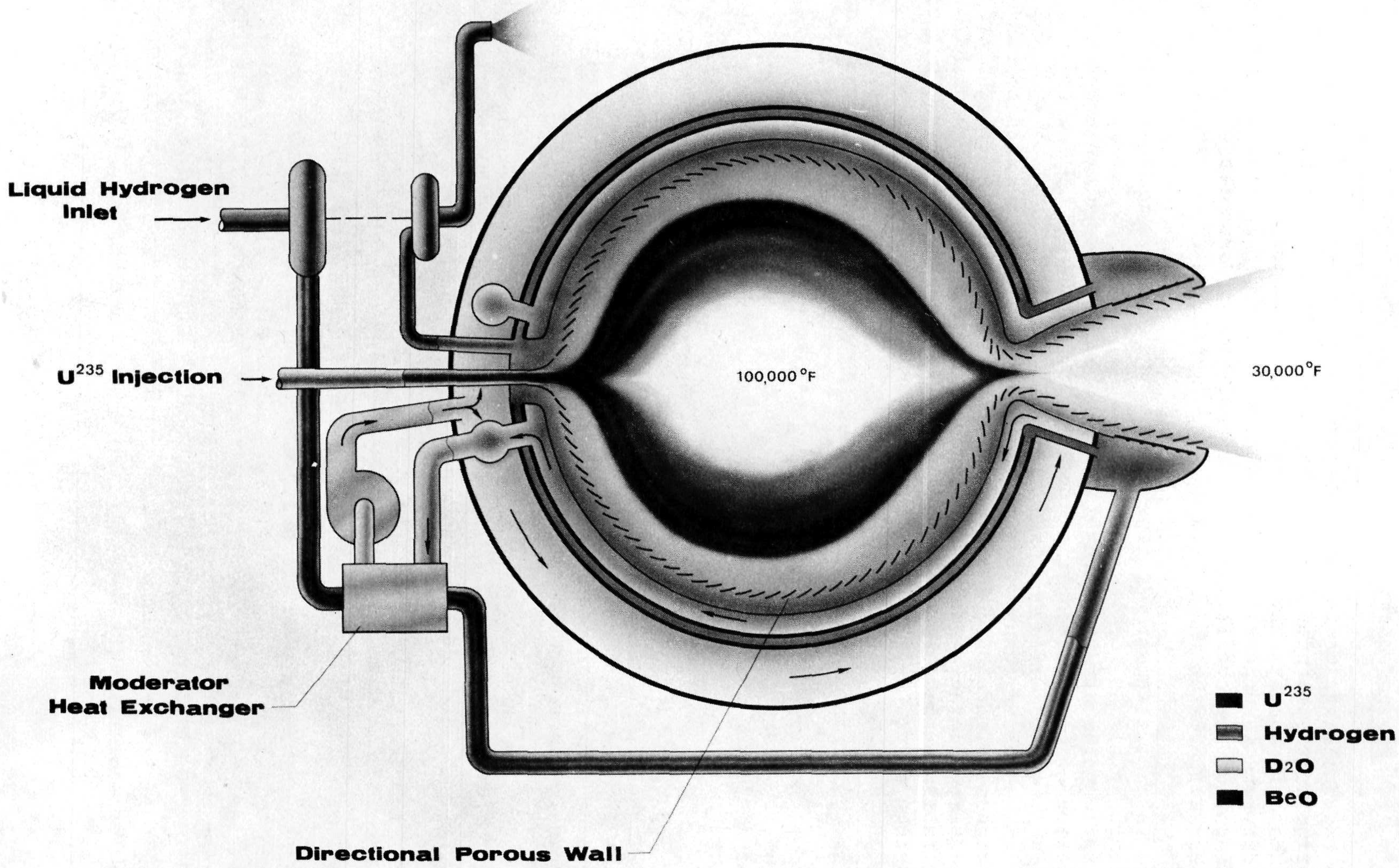

Figure 2.1 Gas Core Nuclear Rocket Concept

3.0 TEST CONFIGURATION AND APPARATUS

3.1 Reactor Configuration

The general reactor configuration is shown in Figure 3.1. An aluminum tray suspended from the lid of the reflector-moderator tank was used to hold an assembled block of beryllium oxide in the heavy water reflector tank. The block, a mockup of an annular BeO heat shield sector, was constructed from 1/2 hexagonal bars of BeO, nominally 4.76 cm across corners by 14.29 cm long. Measurements were made through three block thicknesses (12.42 cm, 22.77 cm, and 31.05 cm) and the rectangular dimensions in each case were 71.44 cm x 69.22 cm.

The aluminum tray in which the blocks were assembled was open at the top, with outside dimensions of 91.44 cm x 91.44 cm x 30.48 (height) cm with 0.633 cm thick walls and bottom. It was positioned 5.08 cm above the sphere forming the inner wall of the moderator tank as shown in Figure 3.1. A hole cut in the center of the tray allowed it to slip down over the sensor tube, a 2-inch schedule 40 aluminum pipe, which extends up from the cavity wall (inner wall of the D_2O tank). This tube served as an access hole for sensors into the reactor. The bulk of the gamma ray heating measurements were made vertically through the BeO slab at a distance of 30.48 cm from the center of the sensor well, using the small thermoluminescent dosimeters (TLD) inserted between the beryllium blocks. During the gamma ray heating measurements the sensor tube was plugged with beryllium oxide over the same axial distances covered by the main beryllium heat-shield block.

The reactor fuel loading was 14.631 kg of uranium, 93.2% U^{235} enriched, in the form of UF_6 maintained in the vapor state by a hot air heating system. A spherical annulus between the core heating system shrouds and the sphere forming the inner wall of the D_2O reflector-moderator tank contained 22.33 kg of foamed polystyrene and 14.23 kg of polyethylene sheet cut and distributed evenly throughout the volume.

A more detailed overall diagram of the Spherical Cavity Reactor Critical Experiment is shown in Figure 3.2 and Table 3.1 is a component code to identify the parts. Table 3.2 gives a description of the reactor components. The configuration is essentially similar to the configuration #2 described in References 1 and 16.

3.2 Description of Devices Used to Measure Heating Rate

The bulk of the heat rate measurements were made with small thermoluminescent dosimeters. These devices are LiF* chips nominally 0.3175 cm x 0.3175 cm x 0.889 cm formed by compacting LiF powder to an average measured density of 2.69 gms/cm^3. The TLD-700 series dosimeters used for these experiments are depleted in Li^6 so that the manufacturers

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* CaF dosimeters were also installed in the experiment, but the data proved to be most inconsistent and unusable.

quoted Li^6 content is nominally 0.005%, thus substantially reducing the thermal neutron sensitivity.

Calibration, data readout, and dose conversion were performed by the AEC Idaho Operations Health and Safety Laboratory. There, the fundamental calibration standard is a Cobalt-60 source with gamma rays at 1.17 and 1.33 Mev. However, absorbed dose depends both on the gamma ray energy spectrum and the absorbing material. Therefore, a beryllium Bragg-Gray chamber was exposed in the sensor well with each TLD traverse to give a direct calibration of the TLD reading. This chamber was fabricated by milling out a chamber volume in a 1/2 hexagon bar of beryllium and fitting with a polystyrene insulator and a central electrode to give a net chamber volume of 2.28 cm^3. The center of the chamber volume was located 8.1 cm above the bottom of the BeO block for all of the measurements where TLD traverses through the block were made. The TLD calibration was performed by placing several of the TLD's directly in the chamber cavity on a separate run from the one for which the cavity current was measured. Fundamentally, this calibration should be considered the only valid calibration, for it is not subject to the differences in gamma spectrum which result from the use of a Co-60 calibration. However, both results are reported for reference.

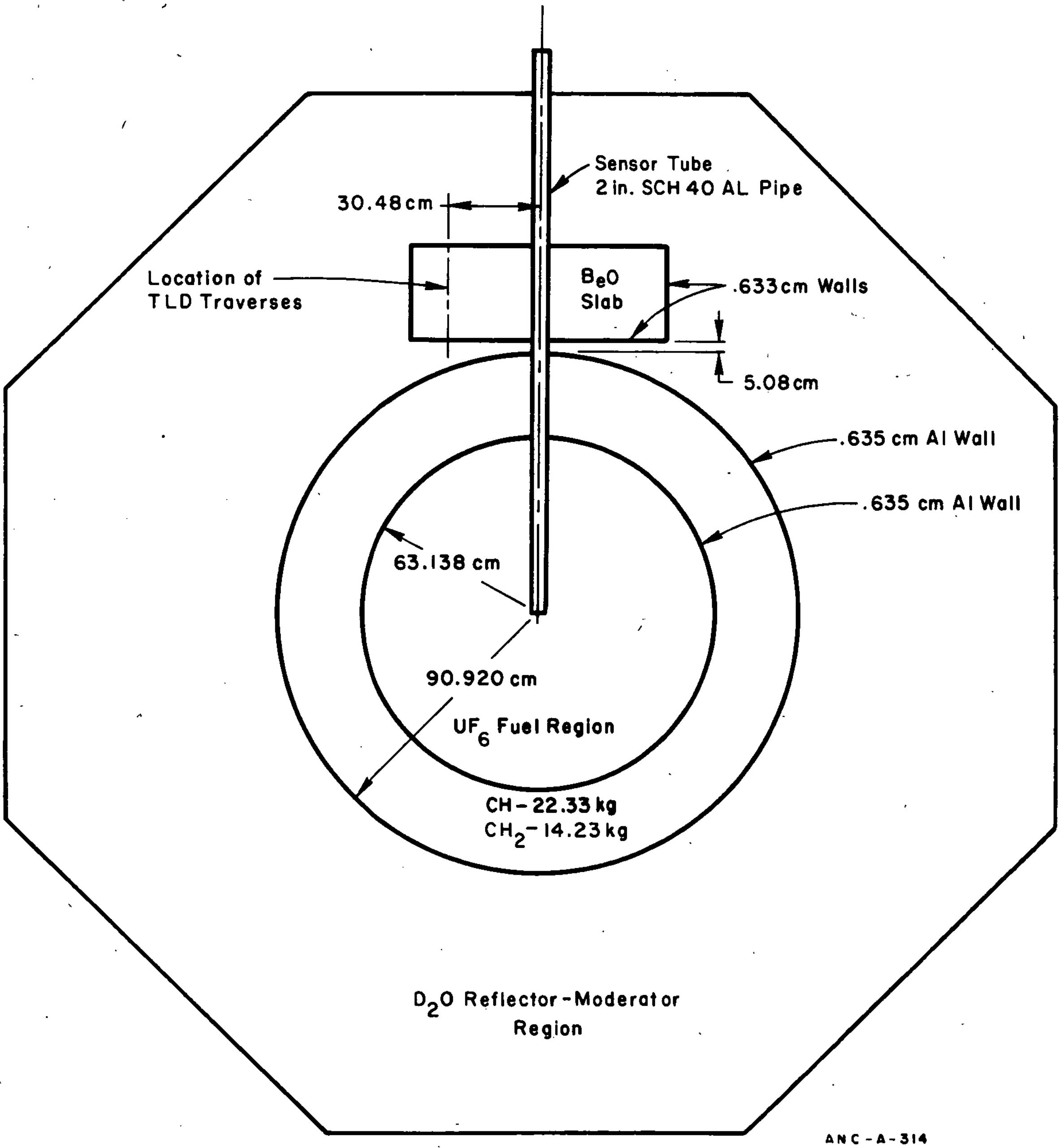

Figure 3.1 Diagram of Spherical Gas Core Reactor showing gamma heating measurement detail.

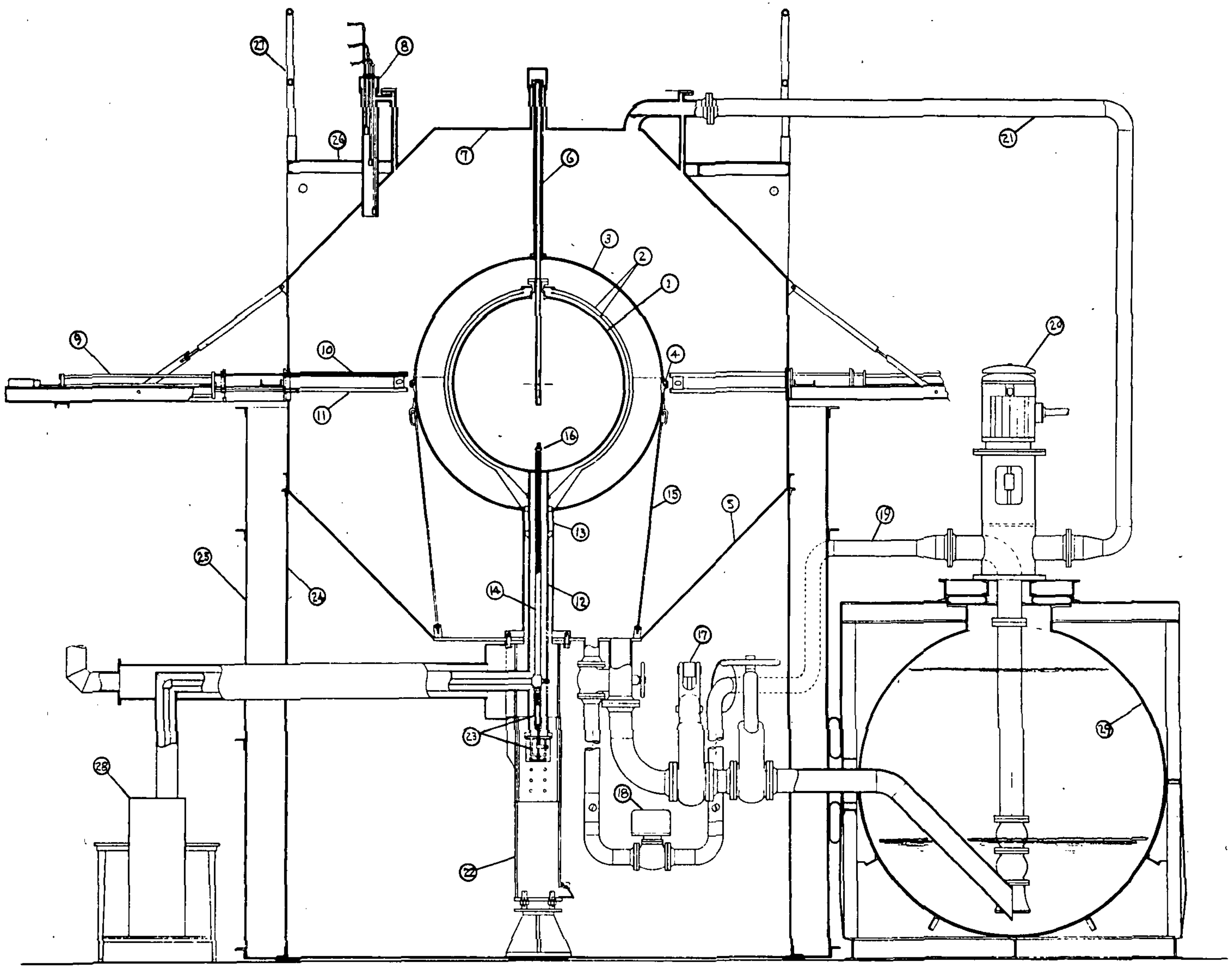

Fig. 3.2 Overall diagram of the Spherical Cavity Reactor Configuration

TABLE 3.1

Spherical Cavity Reactor Configuration Code

Code No.	Description
1	Core sphere tank
2	Air flow guide baffles
3	Cavity sphere tank
4	Stainless steel V-band connector
5	D_2O tank
6	Sensor well
7	Removable D_2O tank lid
8	D_2O level sensing and fill level limit switches
9	Typical of 8 symmetrical control rod actuators and support
10	Control rod poison tip (cluster of three per actuator)
11	Control rod guide tube
12	Core tank support column
13	Cavity tank support column
14	UF_6 fuel line
15	Cavity tank hold down rod (typical of 8 symmetrical)
16	Core tank valve
17	Air operated D_2O quick dump valve
18	Motor operated D_2O inlet control valve
19	D_2O fill line
20	D_2O pump
21	D_2O overflow and cover gas return line
22	Main support column
23	Valve bellows and valve actuating mechanism
24	D_2O tank support column
25	Emergency D_2O catch tank
26	Work platform
27	Hand rail
28	UF_6 transfer and core heating system
29	D_2O storage tank

TABLE 3.2

Reactor Component Description

Component	General Description	Weight kg	Dimensions	Material
Core tank	Two spun hemispheres welded together. Designed for internal pressure of 50 psig at 230° F.	90.72	127.55 cm O.D. x 0.635 cm wall nominal	5052 Al
Core tank sensor well flange	2-inch pipe flange and neck extends above core tank 12.86 cm	2 (calc.)		6061 Al
UF_6 inlet line	1-inch schedule 40 pipe		3.34 cm O.D. x 0.338 cm wall	5052 Al
Core tank support column	5-inch schedule 40 pipe		14.12 cm wall x 0.655 cm wall	5052 Al
Cavity tank wall	Two spun hemispheres held together with a SS V-band clamp and sealed with a butyl rubber O-ring	204.0	183.10 cm O.D. x 0.635 cm wall nominal	5052 Al
Cavity tank support column	8-inch schedule 40 pipe		21.9 cm O.D. x 0.818 cm wall	5052 Al
V band connecter and bolts	Marmon V-band clamp	8.62		301 SS
Sensor well guide tube	2-inch schedule 40 pipe			1100 Al

TABLE 3.2

(Continued)

Component	General Description	Weight kg	Dimensions	Material
Control rod guide tubes	3/4-inch schedule 40 pipe sealed at one end with welded flat plate			5086 Al
Control rod guide tube end spacers	Triangular piece - one on the end of each gang of 3 guide tubes	.325 ea.		1100 Al
Startup neutron source guide tube	1 1/2-inch schedule 40 pipe - penetrates reflector-moderator tank above midplane and extends downward at 30° angle to cavity tank wall near midplane			5052 Al
Cavity tank hold down rods	8 - 9/16-inch diam. rods equally spaced around periphery of cavity tank and extending downward to the floor of the reflector moderator tank			6061 Al
Air flow circulation baffles	2 concentric octagonal spheres with a transition to cylindrical at the bottom- constructed from 20 gauge aluminum sheet	29.94		1100 Al
Core tank valve push rod	1/2-inch diameter rod			1100 Al

4.0 TEST PROCEDURES

The measurements required six reactor runs. These runs are summarized in Table 4.1 where the purpose and test conditions are given for each run. The concluding run of the series, run 115, was made with the sensor well tube which penetrates the center of the BeO block, packed with small 5.08 cm long 1/2 hex pieces of BeO with both gold foils and TLD's at each interface. Two of the gold foils (one at the bottom of the BeO block and the other near the top of the 30.48 cm thick block) were cadmium covered and a thermal neutron distribution through the block was calculated using the measured total gold response and a linear interpolation of the cadmium ratios determined at the two measured positions. All of the TLD-700 data were corrected for the thermal neutron response.

The principal thermal neutron thermoluminescent detector response in LiF originates from the reactions listed in the following table and a calculated correction factor is listed for each reaction with the total being the correction factor applied to the data. The Li^6 contribution is based on 0.005% of total Lithium present.

Reaction	Cross Section	Correction Factor (RADS/n/cm^2)
Li^6 (n, 4.8 mev α)	970 b	9.46×10^{-11}
Li^7 (n, 3 mev α + 13 mev β)	36 mb	4.16×10^{-11} (α)
		1.46×10^{-11} (β)
F^{19} (n, 5.4 mev β)	10 mb	$.5 \times 10^{-11}$
		1.56×10^{-10}

The calculated correction factors assume that all of the energy release is absorbed in the TLD chip and the β energy absorption is estimated from the differential energy loss for electrons in water[12].

A better method for obtaining an effective correction factor would be by direct measurement of the effect. An experiment was performed to make this measurement but was unsuccessful within the accuracy available. The reason is discussed in section 7.0.

Heating rate distribution information was obtained by placing several TLD-700 dosimeters encased in small sealed polyethylene satchets, approximately 1 cm square, at the interfaces of the hexagonal BeO bars composing the heat shield mockup. Thus a relatively fine spacial distribution was obtained. The data represents the average reading from two to four TLD's at a specific depth in the block. All distributions were obtained vertically through the BeO block at a distance of 30.48 cm from the center (sensor well) as shown in Figure 3.1. Thus, if any gammas streamed up the sensor well, the 30 cm distance was far enough away that the TLD's would not have been affected.

The raw TLD data were obtained in units of Rads dose to the dosimeter based on Co-60 calibration. One Rad equals 100 ergs of energy deposited per gram of material. All of the data was then normalized by converting it to units of Rads per watt-hr of reactor operation. Reactor power was determined by exposing catcher foils (aluminum foils placed against bare enriched uranium foils to catch the recoil fission products whose activity is proportional to the fission rate) in the sensor well to obtain a fission rate distribution through the core region. Since the core is homogeneous UF_6, these data were then volume weighted and by knowing the U-235 core loading, a reactor power was calculated for the run. Gold foils placed in the same location on the moderator tank during each run were used to normalize the power on those runs where catcher foil distributions were not obtained.

TABLE 4.1

Summary of Reactor Runs

Run No.	Purpose	Test Conditions
110	To determine the reactor power required to give useful data. A minimum of TLD's exposed in a 12.42 cm thick BeO slab along with catcher foils and gold normalizers. No useful heating rate data obtained.	
111	TLD heating rate measurement traverse through a 12.42 cm thick BeO slab.	Reactor power = 75.6 watts TLD exposure time = 1.333 hrs. TLD watt-hr of exposure = 100.8. Catcher foil and gold normalizer exposure time = 20 min. Be Bragg-Gray Chamber in sensor well at 104.7 cm from center of reactor (8.10 cm from bottom of BeO slab). Bragg-Gray chamber temperature 60°C to 77°C with atmospheric pressure at 642.8 mm Hg. Three TLD's attached to outside of B-G chamber. Four TLD chips at each traverse location in the BeO slab.
112	TLD heating rate measurement traverse through a 22.77 cm thick BeO slab.	Reactor power = 37.8 watts. TLD exposure time = 1.0 hrs. TLD watt-hr of exposure = 37.8. No catcher foils exposed. Gold normalizers exposed 1.0 hrs. Be Bragg-Gray chamber in sensor well at 104.7 cm from center of reactor (8.10 cm from bottom of BeO slab). Bragg-Gray chamber temperature nominally 60°C to 77°C with atmospheric pressure at 647.4 mm Hg. Three TLD chips attached to outside of B-G chamber. Four TLD chips at each traverse location in the BeO slab.

TABLE 4.1

(Continued)

Summary of Reactor Runs

Run No.	Purpose	Test Conditions
113	TLD heating rate measurement traverse through a 31.05 cm thick BeO slab.	Reactor power = 37.8 watts. TLD exposure time = 1.0 hours. TLD watt-hrs exposure = 37.8. Catcher foil and gold normalizer exposure time = 20 min. Be Bragg-Gray chamber in sensor well at 104.7 cm from center of reactor (8.10 cm from bottom of BeO slab). Bragg-Gray chamber temperature and pressure nominally the same as for Run 112. Three TLD chips attached to outside of B-G chamber. Three TLD chips at each traverse location in the BeO slab.
114	Expose TLD's inside the Bragg-Gray chamber.	Reactor power = 85.8 watts. TLD exposure time = 20 min. TLD watt-hr exposure = 28.6. Catcher foil and gold normalizer exposure time = 20 min. Be Bragg-Gray chamber in same location as Runs 111, 112 and 113. TLD chips inside of chamber at bottom, middle and top. TLD chips at three locations on the 1/2 hex bar containing the B-G chamber.
115	Measure the neutron flux in the 31.05 cm thick BeO slab.	Reactor power = 85.8 watts. TLD exposure time = 20 min. TLD watt-hr exposure = 28.5. Gold normalizer exposure time = 20 min. The sensor well through the BeO block was packed with BeO hex pieces 5.08 cm long. A TLD and a gold foil was exposed at each 5.08 cm interval through the block starting at the bottom.

TABLE 4.1

(Continued)

Summary of Reactor Runs

Run No.	Purpose	Test Conditions
115 (con't)		A cadmium covered gold foil was exposed at the bottom and near the top of the BeO block. A small thimble-shaped Be Bragg-Gray chamber (volume = 1.69 cm^3) was exposed near the top of the block in the sensor well. TLD's were attached to the top, middle, and bottom of the B-G chamber.

5.0 TEST RESULTS

The measured gamma ray heating rate distribution through the beryllium oxide block is tabulated in Tables 5.1, 5.2 and 5.3 for the three block thicknesses, 12.42, 22.77 and 31.05 cm respectively. The corrected Rads/watt-hr is obtained by multiplying the measured thermal neutron fluence per watt-hr times the calculated thermal neutron response 1.56×10^{-10} Rads/n/cm^2 and subtracting this value from the total TLD measured Rads/watt-hr. The corrected values are plotted in Figure 5.1 for each block thickness.

Table 5.4 summarizes the results of the beryllium Bragg-Gray chamber exposed 8.1 cm above the bottom of the BeO block while exposing TLD's for each of the block thicknesses noted above. The average dose rate measured for the three reactor runs is 8.56 ± 0.27 Rads/watt-hr. TLD's were exposed on the outside of the chamber during each of these runs but these TLD's were subject to some exposure from core gammas streaming up the sensor tube and as noted in the table appear to be high. TLD's were exposed inside the chamber and outside the chamber sandwiched between two 1/2 hex beryllium pieces (one 1/2 hex contained the chamber). The data from TLD's inside the chamber (Run 114) is in very good agreement with the Bragg-Gray chamber data. The TLD data from run 115 is not reported because it was obviously influenced by the presence of the cadmium buckets used with the gold foils exposed on this run. Thermal neutron absorption in cadmium produces a 9 mev gamma ray.

TABLE 5.1

Gamma Ray Heating Rates

in a 12.42 cm thick BeO Block (Run No. 111)

Distance into Block (cm)	Total Measured* Rads/watt-hr	Measured Thermal Neutron Fluence per watt-hr x 10^{-10} (n/cm^2-watt-hr)	Corrected Rads/watt-hr
0	11.8	1.16	9.97
2.07	9.93	1.28	7.93
4.14	8.13	1.33	6.06
6.21	6.58	1.36	4.46
8.28	5.73	1.36	3.61
10.40	5.19	1.33	3.12

*Direct integrated light reading converted to Co-60 equivalent dose

TABLE 5.2

Gamma Ray Heating Rates

in a 22.77 cm thick BeO Block (Run No. 112)

Distance into Block (cm)	Total Measured* Rads/watt-hr	Measured Thermal Neutron Fluence per watt-hr x 10^{-10} (n/cm^2-watt-hr)	Corrected Rads/watt-hr
0	12.60	1.16	10.80
2.07	9.93	1.28	7.93
4.14	8.41	1.33	6.34
6.21	7.24	1.36	5.11
8.28	6.33	1.36	4.21
10.40	5.50	1.33	3.42
12.40	4.62	1.30	2.59
14.50	4.23	1.25	2.28
16.60	3.67	1.20	1.80
18.60	3.42	1.14	1.64
20.70	2.84	1.07	1.17

*Direct integrated light reading converted to Co-60 equivalent dose

TABLE 5.3

Gamma Ray Heating Rates

in a 31.05 cm thick BeO Block (Run No. 113)

Distance into Block (cm)	Total Measured* Rads/watt-hr	Measured Thermal Neutron Fluence per watt-hr x 10^{-10} (n/cm^2-watt-hr)	Corrected Rads/watt-hr
0	13.33	1.16	11.50
2.07	10.30	1.28	8.30
4.14	8.72	1.33	6.65
6.21	7.21	1.36	5.09
8.28	6.28	1.36	4.16
10.40	- -	1.33	- -
12.40	4.77	1.30	2.74
14.20	4.23	1.25	2.28
16.60	3.72	1.20	1.85
18.60	3.42	1.14	1.64
20.70	3.23	1.07	1.56
22.80	2.81	1.00	1.24
24.80	2.57	0.92	1.13
26.90	2.32	0.83	1.03
29.00	1.96	0.75	0.79

*Direct integrated light reading converted to Co-60 equivalent dose

TABLE 5.4

Summary of Bragg-Gray - TLD Calibration
(2.28 cm^3 beryllium thick-wall chamber)

Run	Chamber Current (amps)	Pressure (mm Hg)	Temperature (^oC)	RAD's/watt-hr Chamber	RAD's/watt-hr TLD[1]
111	1.31×10^{-10}	643	71.1	8.76	13.7
112	0.62×10^{-10}	647	71.1	8.25	10.7
113	0.66×10^{-10}	647	71.1	8.68	11.7
114					8.36 (8.04)[2]

(1) The TLD's were located on the outside of the beryllium 1/2 hex block containing the chamber for runs 111, 112 and 113.

(2) On run 114, TLD's were located inside the chamber at the top, middle and bottom to give an average of 8.36 Rads/watt-hr which can be compared directly with the dose rate measured with the chamber on the previous three runs. TLD's sandwiched between the two 1/2 hex beryllium bars at chamber height gave the value 8.04 noted in the table.

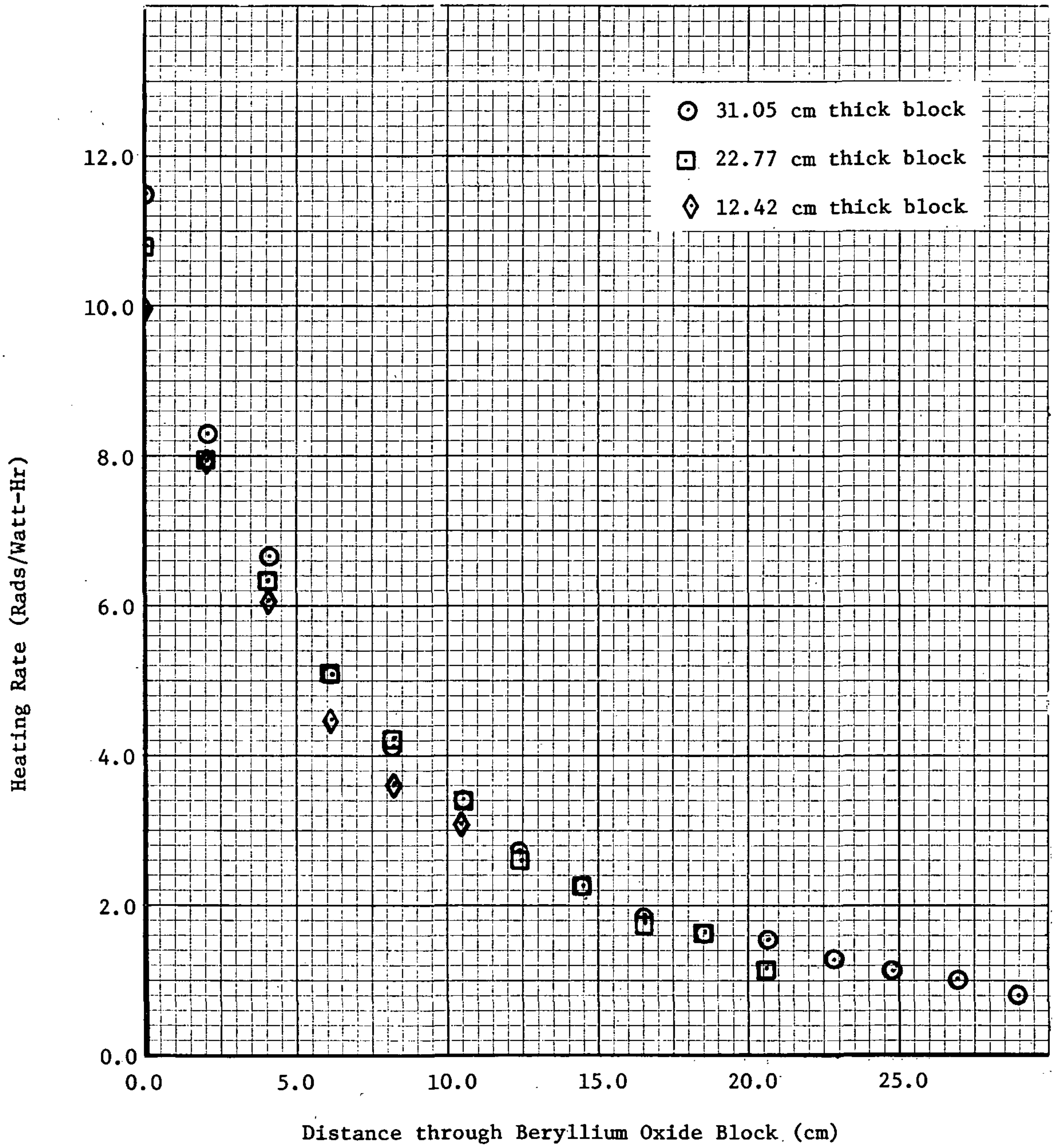

Figure 5.1 Heating Rate in Three Thicknesses of Beryllium Oxide - as Calibrated with Co-60 Spectrum

6.0 ANALYTICAL CALCULATIONS AND COMPARISON WITH MEASURED RESULTS

Calculations were performed using computer codes operable on the NRTS 360/75. Both point kernel shielding and Sn transport theory (for both neutrons and photons) calculations were made and the results are compared with the values measured in the critical experiment.

6.1 Calculational Model and Computer Codes

A calculational model is given in Table 6.1 and is the model used for both the point kernel and the transport code calculation of heating rate in a beryllium oxide reflector annulus of the critical experiment. The calculations were performed using the 18 energy group structure given in Table 6.2.

The QAD-P5A multi-energy group, point kernel shielding code[2] determines the dose rate by calculating the uncollided flux and applying appropriate buildup factors along line-of-sight paths from source volume elements to receptor points. Both direct core gamma sources and capture gamma sources were included. Gamma ray attenuation coefficients and buildup factors are included in the code library. Light element buildup was employed, since only one type can be used in a given problem. The gamma ray coefficients are determined by linear interpolation of data at 20 energy levels[3,4]. Buildup factors are calculated from polynomial fit to buildup data[5,6]. Neutron induced gamma ray source data (n,γ reactions) were generated using measured 2200 m/sec equivalent thermal neutron flux data from the critical experiment (see Table 5.3) and 2200 meter/sec (n,γ) cross sections from BNL-325 shown in Table 6.3. Core fission gamma ray distributions were obtained from Reference 7. These data are listed in Table 6.4. The contributions from the various sources are listed in Table 6.5. The direct fission gammas from the core represent the major contribution to the dose at the front part of the heat shield. However, these gamma rays have a nominal relaxation length of 8 cm in the BeO, such that at a 15 cm depth, capture gamma rays represent half of the dose. At the 30 cm depth, 80% of the dose is the result of capture gammas.

Transport calculations were performed with the versatile multi-energy group code SCAMP[8]. Nominally a neutron flux-eigenvalue code was adapted to perform these photon transport calculations. First a 19 energy group neutron problem using the SCAMP Sn code[8], with cross sections generated by the PHROG and INCITE codes[9,10] was performed to provide a neutron distribution through a spherical reactor with an annulus of BeO reflector*. The 19-group neutron data was then coalesced

* Note: The calculated thermal neutron flux in the BeO region agreed to within a few percent of the values measured in the critical experiment.

to four groups. With this data as input to the GAMSOR code[11], and using, fission and (η,γ) reaction cross sections from Reference 7 (with some minor adjustments in energy group assignments), an 18 energy group set of region gamma rays volume sources was generated. These region gamma ray volume sources then formed the input to SCAMP, along with photon transport kernels calculated by the GAMLEG Code[13], to calculate the dose rate distribution through the BeO annulus. The photon absorption cross section data used as input to GAMLEG are from Reference 14. Most of the calculations were performed in S_4 angular detail, however one case was run in S_8 detail with but a barely noticeable change in results. Thus, it was concluded that S_4 detail was adequate for the gamma transport problem, as it had been found to be for the neutron problem.

6.2 Comparison of Measured and Calculated Results

The data from Table 5.3 for the 31.05 cm thick beryllium oxide block in the critical experiment are plotted in Figure 6.1 along with the results of the transport code calculation and the point kernel shielding calculations. The point kernel code gave heating rate results that are about 8 to 10% higher than the transport code, but these results are only about half the values measured in the critical experiment. A beryllium Bragg-Gray chamber measurement in the sensor tube that penetrated through the beryllium oxide block gave results that were 3% (±8%) larger than the heating rate measured with TLD's placed inside the same chamber (on a separate reactor run).

Were it not for the fair agreement between the two methods of measurement, especially since the Bragg-Gray ionization chamber method is relatively simple and straightforward, one would be inclined to cast doubt on the validity of the results from the TLD measurements. Furthermore, the Bragg-Gray measurement is fundamentally the correct way to calibrate the TLD's for the gamma spectrum of the particular environment. In light of the extensive checking and cross checking of TLD calibrations and reactor power calibrations the experimenters are more prone to question the accuracy of the calculations with photon cross section data the most open to suspicion. The relatively close agreement between the Bragg-Gray direct results and the TLD results based on Co-60 spectrum calibration is not unexpected. Both LiF and BeO consist of light elements with similar response to all gamma ray energies. Furthermore, in the 1 to 4 Mev region of most of the gamma rays encountered, compton scattering is by far the most predominant mechanism for energy loss in light elements, and it gives the same response for both LiF and BeO on an electron for electron basis (equal energy mass absorption coefficients).

TABLE 6.1

Calculational Model of Spherical BeO Heating Rate Experiment

Inner Radius (cm)	Outer Radius (cm)	Region Volume (cm^3)	Material	Mass (kg)
0.0	63.109	1.0528×10^6	UF_6	14.631
63.109	63.773	3.3583×10^4	Aluminum	90.72
63.773	90.830	2.0525×10^6	CH CH_2	22.33 14.23
90.830	91.555	7.5765×10^4	Aluminum	204.0
91.555	96.635	5.6534×10^5	D_2O	624.7
96.635	127.115	4.8236×10^6	BeO	156.99
127.115	188.14	1.9292×10^7	D_2O	21318.0

TABLE 6.2

Gamma Ray Energy Group Structure used for Calculating Heating Rates

Group	Energy Limits (MeV)	Group	Energy Limits (MeV)
1	10.00-8.00	10	2.00-1.60
2	8.00-7.00	11	1.60-1.20
3	7.00-6.00	12	1.20-0.90
4	6.00-5.00	13	0.90-0.60
5	5.00-4.00	14	0.60-0.40
6	4.00-3.50	15	0.40-0.21
7	3.50-3.00	16	0.21-0.12
8	3.00-2.50	17	0.12-0.07
9	2.50-2.00	18	0.07-0.01

TABLE 6.3

Thermal Neutron (η,γ) Cross Sections and Gamma Yields per Capture for the Important Reactor Materials

	F	H	Be	O	D	Al
$^{*}\sigma(\eta,\gamma)$	0.0098 (±0.0007)	0.332 (±0.002)	0.0095 (±0.001)	0.000178 (±0.000025)	0.0005 (±0.0001)	0.235 (±0.005)
Gamma Energy Group						
1	0.0	0.0	0.0	0.0	0.0	0.0
2	0.0	0.0	0.0	0.0	0.0	0.351
3	1.0	0.0	0.0	0.0	0.950	0.084
4	0.0	0.0	0.784	0.0	0.0	0.099
5	0.0	0.0	0.0	0.0	0.0	0.553
6	0.0	0.0	0.0	0.296	0.0	0.157
7	0.0	0.0	0.487	0.296	0.0	0.157
8	0.0	0.0	0.0	0.276	0.0	0.168
9	0.0	0.990	0.0	0.276	0.0	0.168
10	0.33	0.0	0.0	0.140	0.0	0.088
11	0.33	0.0	0.0	0.179	0.0	0.113
12	0.33	0.0	0.0	0.0	0.0	0.113
13	0.0	0.0	0.0	0.0	0.0	0.0
14	0.0	0.0	0.0	0.0	0.0	0.0
15	0.0	0.0	0.0	0.0	0.0	0.0
16	0.0	0.0	0.0	0.0	0.0	0.0
17	0.0	0.0	0.0	0.0	0.0	0.0
18	0.0	0.0	0.0	0.0	0.0	0.0

* Thermal neutron (η,γ) reaction cross sections from BNL-325

TABLE 6.4

Prompt Fission and Fission-Product

Gamma Yield Spectra (γ/fission)

Group No.	Energy Limits (MeV)	Prompt Fission	Fission-Product
1	10.00-8.00	0.0	0.0
2	8.00-7.00	0.0005	0.0
3	7.00-6.00	0.0093	0.0
4	6.00-5.00	0.0197	0.0
5	5.00-4.00	0.0586	0.0
6	4.00-3.50	0.0805	0.0
7	3.50-3.00	0.0805	0.0
8	3.00-2.50	0.2480	0.2554
9	2.50-2.00	0.2480	0.2554
10	2.00-1.60	0.4997	0.4455
11	1.60-1.20	0.6425	0.5728
12	1.20-0.90	0.6425	0.5728
13	0.90-0.60	1.1120	1.3783
14	0.60-0.40	1.6685	2.0681
15	0.40-0.21	2.7528	3.4120
16	0.21-0.12	0.0	0.0
17	0.12-0.07	0.0	0.0
18	0.07-0.01	0.0	0.0

TABLE 6.5

Contributions to the Dose Rates in BeO Heat Shield (Rads/watt hour)

Source Description	Position in Heat Shield					
	From Reactor Center (cm)	97.9	105.1	113.1	119.1	125.1
	From inner edge (cm) of BeO	0.635	7.835	15.835	21.835	27.835
(1) Core fission gammas		5.083	2.086	0.674	0.292	0.131
(2) Capture gamma sources						
Core tank and heater		0.132	0.072	0.035	0.021	0.013
Misc. structure		0.225	0.106	0.043	0.026	0.016
Hydrogen region		0.069	0.027	0.009	0.004	0.002
Cavity tank		0.505	0.198	0.082	0.044	0.025
Interior D_2O		0.002	0.001	0.0	0.0	0.0
BeO tray (Al)		0.851	0.303	0.113	0.061	0.035
Exterior D_2O		0.033	0.043	0.081	0.132	0.253
BeO		0.167	0.204	0.206	0.186	0.142
Sub Total of Sources other than Core		1.984	0.954	0.569	0.474	0.486
Total		7.07	3.04	1.24	0.766	0.617

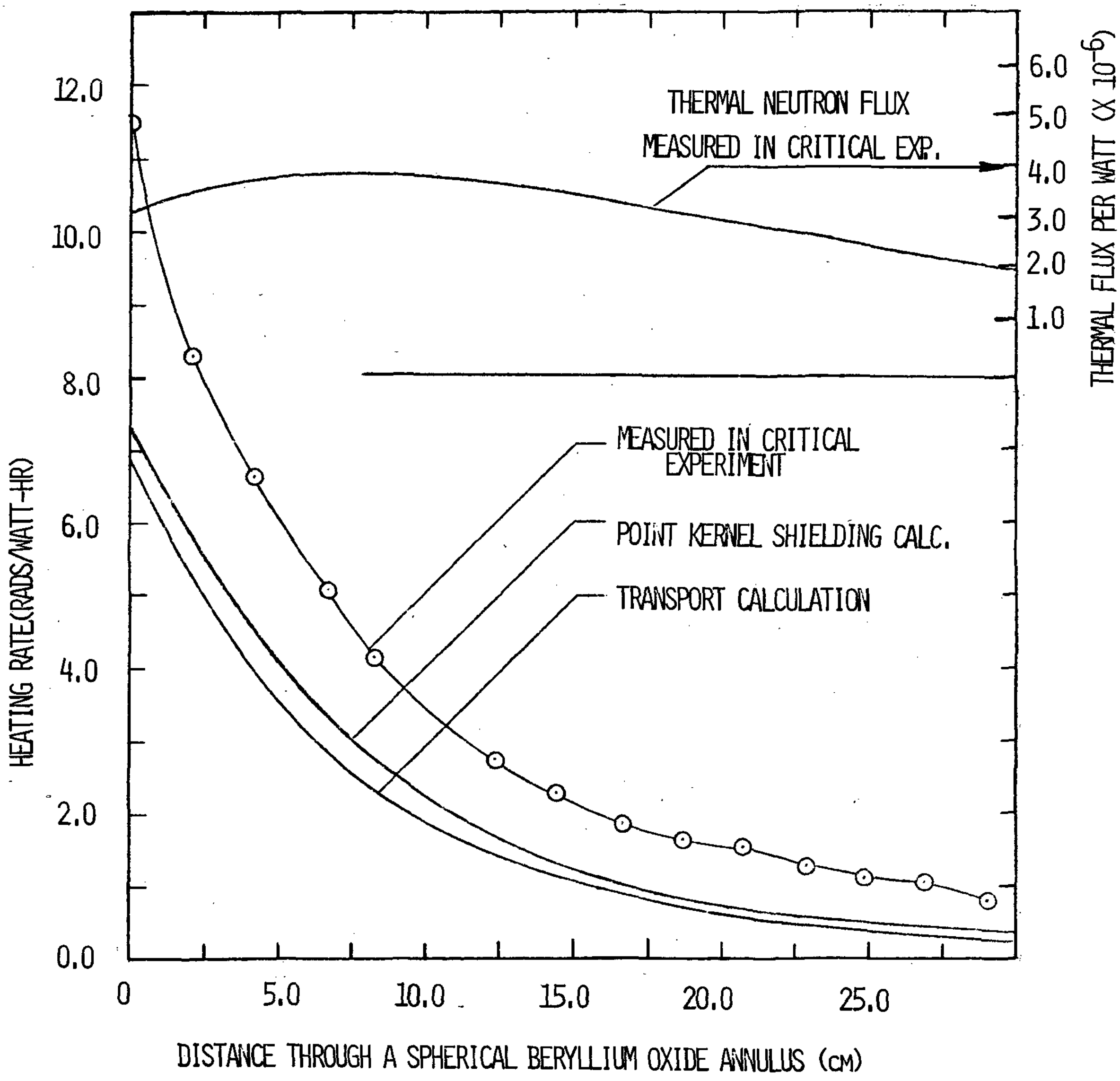

Figure 6.1 Comparison of Measured and Calculated Gamma Ray Heating Rate

7.0 DISCUSSION OF RESULTS AND PROBLEMS ENCOUNTERED

The large difference between the measured and calculated values for gamma ray heating rate in a gas core reactor with beryllium oxide heat shield is somewhat discouraging and at this point is unexplained. When the calculational results had been refined to the point where only minor improvements were being obtained*, considerable effort was then expended to determine if a shift in the reactor power calibration had occurred. The reactor power calibration depends on the count obtained on a catcher foil in a 2 π beta counter arrangement. The calibration constant had previously been derived from thermal flux fissions and from direct comparison with an absolute fission chamber. That constant had been used for years for such calibrations at the Low Power Test Facility.

In order to recheck this calibration factor, two small absolute fission chambers constructed with plates containing carefully determined amounts of U-235 were exposed, along with catcher foils. Gold foils (0.005 mil thick) both bare and cadmium covered were exposed in a subcritical assembly driven with a 2 milligram Cf-252 neutron source and also in a beryllium sigma pile with a 1 milligram Cf-252 neutron source. The results from the two absolute fission chamber counts gave excellent relative agreement between measurements. However, the results calculated from bare and cadmium covered gold foil counts were lower by some 18%. A calibration factor was adopted which was the mean between these two results. It is therefore believed that the reactor power could not have been in error by more than ±10%. This probably has a high (~95%) confidence limit, since the standard deviation on each of the two methods was approximately ±5%. The reason for the difference between the two results is not known. However, neither experiment was performed in the most favorable environment of highly thermal flux with minimum perturbations.

The problem of determining an appropriate correction factor for the TLD thermal neutron response has been mentioned in previous sections of this report. A direct measurement of this effect was attempted by exposing sets of TLD's in a beryllium sigma pile with 1 mg Cf-252 source. Cf-252 has an acceptable ratio of neutron to gamma output for such an experiment. However, the source was encapsulated in stainless steel and the gamma ray output from capsule activation was so great that the neutron contribution could not be separated from this extraneous gamma exposure. Had a Cf-252 neutron source, encapsulated in a material, such as magnesium, with a low (η,γ) reaction cross section, been available the experiment could have been successful. As discussed in Section 4.0, a calculated TLD response was used to make corrections to the data using the manufacturers quoted Li^6 content and thermal neutron (η,γ) reaction cross section data. Since these calculations assume that all of the α and/or β energy produced in a reaction

* Note: Rather extensive hand calculations were performed using seven gamma groups (ANL-5800) and (η,γ) cross section data from BNL-325. These calculations were in close agreement with results from the QAD-P5A code.

with the Li or F is absorbed in the TLD to produce TL effect, the data may have been overcorrected.

Reason to not suspect the thermal neutron response correction of 1.56×10^{-10} Rads/n/cm^2 is based on experimental evidence. The IDO Health and Safety Laboratory had conducted a thermal neutron response test on one batch of similar dosimeters from the same manufacturer. Using a Am-Be source, they determined that the thermal neutron response was an equivalent 1 Rad of gamma for every 40 Rem of thermal neutrons. This reduces to 0.26×10^{-10} Rad/n/cm^2, one sixth the value used in this report. If this lower value were used for the correction, the discrepancy between the measured and calculated doses at the outer portion of the heat shield would approach an order of magnitude.

Further indication of the relative reliability of the thermal neutron response correction is given by the comparison of the TLD and Bragg-Gray ionization chamber results. If it is assumed that the Co-60 calibration gives the same response as that from the spectrum in the gas core reactor, then the following relative comparison applies:

Directly measured TLD response	1.5 (Rads, relative)
Directly measured Bragg-Gray dose	1.03
Corrected TLD response using 1.56×10^{-10} Rad/n/cm^2	1.0
Corrected TLD response using 0.26×10^{-11} Rad/n/cm^2	1.42
Calculated using QAD	0.66
Calculated using transport	0.57

It is apparent that the correction factor used in this report gives fair agreement with the Bragg-Gray results. If the correction were 6% less, the agreement would be perfect. The 6% could be accounted for if this fraction of the released α energy actually escaped from the crystal. Note, however, these considerations are not within the accuracy of the data for comparing TLD's and Bragg-Gray results. The TLD standard deviation was 7%, that for the Bragg-Gray chamber result was 4%.

Gamma energy deposition has been measured in critical experiments using CaF_2:Mn thermoluminescent phosphor(15). This phosphorus has an appreciably lower thermal neutron response than does LiF but it has other objectionable characteristics such as non-linear fade with time after exposure thus complicating calibration. Small micro rods of CaF_2:Mn were exposed along with some of the LiF dosimeters in the BeO block. Extreme scatter in the data was experienced and the results are therefore not reported.*

* Improved batches of these dosimeters obtained from different manufacture were later used with success in another experiment, as reported in Ref. 15. However, at that time the gas core experiment had been dismantled.

If indeed the calculations are giving the incorrect answer, the cause has not been unequivocally identified. The forward portion of the heat shield is being calculated approximately 35% low. Since it involves mostly the contribution from the core gammas, it is difficult to identify the cause because calculations of this type are relatively routine. The latter is not true, however, for the deeper regions of the shield where the main contribution arises from capture gammas from D_2O and BeO. These notoriously very low absorbing materials are not customarily considered as sources of secondary gammas. The fundamental nuclear data for these elements has above average uncertainty. For instance, the thermal neutron absorption cross sections recommended in the latest issue of BNL-325 list uncertainties of 20%, 10%, and 14% for D, Be and O, respectively. Thus, it is quite likely that the large discrepancy between calculations and experiments for deep heat shield locations is largely the result of uncertainties in fundamental nuclear data.

The experimental results are considered accurate to within ±10%, at the 95% confidence level. Though the experiment was not of as ideal a geometry as would have been desired, the deviations from a truly "clean" spherical geometry had little effect on the results. For instance, though the core sensor well was unplugged from the BeO slab on down, the mean free path in the adjacent gaseous fuel and simulated propellant was approximately 100 cm, making the streaming in the sensor well appear to be a normal condition.

8.0 CONCLUDING REMARKS

The gas core rocket engine has unique gamma heating problems for its reflector. Reflector heat loads in the range of 300 to 400 MW of power will have to be radiated to space. For this reason, extra precision must be placed on the calculated heating rate values in the reflector. The results from this study show that more effort must be expended to improve the correlation between experimental and calculated results and thus improve confidence in calculational techniques. The study indicates that this effort should begin by verification of neutron capture cross section and gamma ray spectra emitted from the low cross-section elements deuterium, oxygen and beryllium.

The implication to gas core rocket design is obvious. If these experimental results are indeed correct, then calculated gamma heating rates will be low, by 35% to 60%. Thus peak heating rates in the heat shield of a 6000 MW reactor might be calculated to be 100 watts/gm, whereas in actual operation the heating might be observed to be 150 watts/gm. The difference represents heat loads almost never encountered in non-fueled materials in the high power commercial and test reactors of today.

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