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# EFFECT OF $D_2O$ ON INCREASING THE NEUTRON FLUX NEAR A FUELED CAPSULE IN THE PLUM BROOK MOCK-UP REACTOR

by Klaus H. Gumto
Lewis Research Center
Cleveland, Ohio

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# EFFECT OF $D_2O$ ON INCREASING THE NEUTRON FLUX NEAR A FUELED CAPSULE IN THE PLUM BROOK MOCK-UP REACTOR

by Klaus H. Gumto

National Aeronautics and Space Administration
Lewis Research Center
Cleveland, Ohio

## ABSTRACT

The effects of surrounding a fueled capsule mock-up with a tank filled with $D_2O$ were tested in the HT-2 test hole of the Plum Brook Mock-Up Reactor. Replacing the $H_2O$ with $D_2O$ increased the thermal neutron flux by a factor of 2.4 at the fuel region, and a factor of 11 on the side of the test hole facing away from the core. The effects of the capsule wall and the fuel, as well as the gamma heating were also investigated.

## SUMMARY

The feasibility of increasing the thermal neutron flux levels in an experiment capsule by replacing $H_2O$ with $D_2O$ was investigated in the HT-2 test hole of the Plum Brook Mock-Up Reactor (MUR). A series of four experiments were run using a capsule surrounded by a tank filled with $D_2O$. This tank replaced most of the water normally surrounding the experiment capsule.

The first experiment used a fuel assembly, a stainless steel capsule liner, and the $D_2O$ tank. The second experiment was identical to the first except that the $D_2O$ was replaced by light water. The third experiment was like the first, but without the stainless steel liner. This simulated an aluminum or zirconium walled capsule. The fourth experiment determined the effect of fuel loading on the flux distribution and differed from the first only in that the fuel assembly contained no fuel.

The results of the experiment show that surrounding the experiment capsule with $D_2O$ instead of light water increases the thermal flux incident on the fuel by a factor of 2.4 to $4.13 \times 10^{13}$ neutrons/cm$^2$ per second. In positions parallel to the HT-2 axis but away from the reactor core, the thermal neutron flux increased by a factor of 11 to $3.5 \times 10^{13}$ neutrons/cm$^2$ per second when $D_2O$ was used.

The fuel causes a flux depression of about 31 percent between the three fuel pins, and about 34 percent at the fuel pin surface. The ratio of the flux incident on the fuel to the flux at the center of the fuel was 2.9 to 1. The 0.250-inch (0.635-cm) thick stainless steel liner caused a flux drop of about 40 percent across the wall.

The gamma heating distribution is similar to the thermal neutron flux distribution except near the fuel. Here the secondary gamma radiation from the fuel increases the gamma heating instead of causing a depression as occurs in the neutron flux. Gamma heating in the HT-2 test hole ranged from about 3.5 watts/gram near the reactor core centerline to about 0.3 watts/gram at 28 inches (71 cm) east of the core centerline.

## INTRODUCTION

The Plum Brook Reactor is a 60 MW test reactor with a core composed of a 3 by 9 array of MTR type fuel elements cooled by light water. The reactor has many test holes for irradiation experiments. Some are located in the core and primary reflector while others are in the water on the sides of the core. The test holes in the water region have an uneven flux distribution across the hole due to neutron absorption in the $H_2O$. This report describes a set of experiments that were conducted to reduce the flux drop off across test hole HT-2 where the flux drop is more than a factor of ten. HT-2 is a 12-inch test hole running horizontally in the water region adjacent to the core. The method used was to replace as much of the $H_2O$ as possible in the test hole with $D_2O$. Some $H_2O$ will remain in the test hole because it is required to cool the test hole and experiment parts. This method has been used successfully at the University of Michigan on their swimming pool reactor (ref. 1).

The tests were conducted in the Plum Brook Mock-Up Reactor (MUR) which is a low power mock-up of the 60 MW Plum Brook Reactor. The effect of replacing $H_2O$ with $D_2O$ was measured with a simulated circulating gas capsule in the test hole (ref. 2). The effect of fuel and capsule wall material on the flux was also measured.

## DESCRIPTION OF THE EXPERIMENT

This section describes the Mock-Up Reactor (MUR), the apparatus used to conduct the experiments in the MUR, and lastly, describes the experiment configurations.

## The Mock-Up Reactor (MUR)

The MUR (fig. 1(a) and (b)) is a low power, swimming pool type of reactor located at the Plum Brook Reactor (PBR) facility (ref. 3). It is dimensionally identical to the PBR, which it is designed to simulate. The MUR contains 27 fuel elements when fully loaded, identical to those used in the PBR. Light water cools the fuel elements by natural convection and also serves as moderator and as secondary reflector. Beryllium acts as the primary reflector. Typically, the MUR is operated at 10 KW.

The core used in the MUR at the time of the experiments was designated MUR-G. Figure 1(b) shows a top view of the MUR with the amount of $U^{235}$ (in grams) per fuel element in core G indicated.

The HT-2 test hole is in the same location as in the PBR. Experiments are lowered from the surface of the pool to an insertion table. A hand crank located at the edge of the pool operates the insertion mechanism which locates the experiment within HT-2.

## Experiment Apparatus

The experiment apparatus (fig. 2) consisted of four major parts: the experiment canister, the $D_2O$ tank, the experiment capsule, and the fuel assembly. The fuel assembly was placed inside the experiment capsule which rested in a hole passing through the $D_2O$ tank. These three parts were inserted into the experiment canister.

Experiment canister. - All experimental assemblies were contained in an aluminum experiment canister. This canister fit inside the HT-2 test hole and had a handle which engaged the insertion mechanism. The canister was 39.5-inches (100.0-cm) long, with an outside diameter of 11.25 inches (28.6 cm) and a usable inside diameter of 10.22 inches (26.0 cm). One end had a welded hemispherical head, and the other end was threaded to mate with a flat plate which provided a seal with an O-ring. However, for this series of experiments, cooling water was allowed to enter the canister.

$D_2O$ tank. - Figure 3 shows a cross section of the 10-inch (25.4 cm) $D_2O$ tank. This tank was 28-inches (71-cm) long and had a 3.65-inch (8.29-cm) diameter axial hole to accomodate the experiment capsule. This hole was offset 2.25 inches from the tank axis toward the core. Eight tubes with an outside diameter of 0.187 inch (0.475 cm) also passed through the tank in the axial direction to allow insertion of the dosimeters into the tank interior. All parts of the tank were made of aluminum.

Experiment capsule. - The experiment capsule contained the capsule test section, which includes the stainless steel liner, the fuel pin holder, and the fuel pins. These are shown in a cross sectional view in figure 4. The experiment capsule was an aluminum cylinder 3.5 inches (8.9 cm) in diameter with a 0.375-inch (0.935-cm) thick wall and 36-inches (91.5-cm) long. One end was closed with a hemispherical weld cap, while a threaded plug provided access to the inside at the other end. An O-ring sealed the capsule when the plug was in place.

A stainless steel liner fit the inside of the experiment capsule to mock up the wall of the circulating gas capsule. This liner was made from a tube 32.5-inches (82.3-cm) long, with an outside diameter of 2.75 inches (7.0 cm) and a wall thickness of 0.25 inch (0.635 cm).

Fuel assembly. - The fuel assembly shown in figure 5 consisted of three parts: the fuel pin holder, the fuel pins, and the spacer pieces. The assembly fit inside the experiment capsule, either with the stainless steel liner in place or with it removed. Figure 4 shows a cross section of the fuel assembly inside the experiment capsule.

The fuel pins were made by rolling a 2-inch (5-cm) wide strip of uranium foil on a 3/;6-inch (0.475 cm) outside diameter aluminum tube. The uranium was 93.2 percent enriched $U^{235}$, and weighed about 9 grams per fuel pin. The rolled up foil and the aluminum tube were then slipped inside a 0.500-inch (1.27-cm) diameter by 0.035-inch (0.089-cm) wall thickness stainless steel tube. This tube was 11-inches (27.7-cm) long, and mocked up the gas coolant flow guide.

The fuel pin holder (fig. 5) was made from a molybdenum shell (fig. 4) and two end plates. The molybdenum shell was a tube with an inside diameter of 1.5 inches (3.8 cm) and a wall thickness of 0.094 inches (0.238 cm). One end was threaded on the outside, while the other end had a groove for a retaining ring on the inside diameter. The ends of the shell were closed by a circular stainless steel disc 1.5 inches (3.8 cm) in diameter and 0.250-inch (0.635-cm) thick. Three aluminum pins attached to each disc and spaced 120 degrees apart on a 0.438-inch (1.11-cm) radius circle positioned the fuel pins inside the fuel pin holder.

An aluminum rod positioned the fuel pin holder axially inside the experiment capsule. The 0.250-inch (0.635-cm) diameter rod passed through holes in the end plates of the fuel pin holder and extended to both ends of the experiment capsule. This permitted the fuel pin holder to slide along the rod to the desired axial position, where it was then locked in place.

Two stainless steel spacer rings centered the fuel pin holder within the experiment capsule. One set of rings was used with the stainless steel liner in place, while the other set was used when the liner was removed from the capsule. The rings were held by set screws.

### Test Configurations

Four different configurations were tested. The first (run 1) used the $D_2O$ tank, the mock-up fuel assembly and the stainless steel liner. The second configuration (run 2) was the same as run 1 except that the $D_2O$ was replaced by light water. This was done by removing the $D_2O$ tank and substituting an aluminum and plastic rack. This rack held the experiment capsule and the dosimeters in the same positions as the tank. During the experiment the rack was immersed in the core water.

The third configuration (run 3) again differed from run 1 only in one respect. For this run the stainless steel liner was removed in order to mock up an aluminum or zirconium capsule instead of the stainless steel capsule. The last configuration (run 4) was similar to run 1 except the fuel pins were removed and dummy pins containing no fuel were inserted. This test was run to determine if the fuel significantly perturbed the flux.

## EXPERIMENTAL METHODS

### Thermal Neutron Flux Measurements

The thermal neutron flux levels were measured with gold foils and wires and uranium-aluminum alloy wires. The gold wires were 0.5-inches (1.27-cm) long and 0.03 inch (0.076 cm) in diameter, while the foils were disc shaped with a diameter of 0.250 inch (0.635 cm) and a thickness of 0.005 inch (0.013 cm).

Several of the gold wires had cadmium sleeves. These were mounted at random throughout the experiment capsule and the $D_2O$ tank. Their purpose was to measure the fast flux, as the cadmium sleeves absorbed neutrons with energies below 0.5 MeV. From these measurements the ratio of the fast flux to the total neutron flux was obtained, and thus provided a correction factor for the bare gold wires which measured the total flux.

The uranium-aluminum alloy wires had the same dimensions as the gold wires. The uranium-aluminum wires were used in order to obtain the fission rate directly, which could then be converted to the thermal neutron flux, thus providing a check for the gold measurements. The uranium-aluminum alloy wires were used in the fuel region of the experiment capsule, where they were interspersed among the gold wires.

The foils were used on the outside surfaces of both the $D_2O$ tank and the experiment capsule. Their flat shape reduced the changes of being rubbed off during the assembly. The wire dosimeters were used in the remaining positions. Wire dosimeters which were to be inserted into the tubes passing through the $D_2O$ tank and into the fuel pin tubes were taped to 1/16-inch (0.159-cm) diameter aluminum welding rod. This allowed both accurate positioning axially and easy mounting and removal of the dosimeters.

The position of each dosimeter within the experiment capsule is given by the following code. The location of the dosimeter in a plane perpendicular to the test hole axis is indicated by a capital letter. These locations are shown in figures 3 and 4. The axial position is indicated by a station number as shown in figure 2. The station number represents the axial distance east of the reactor center line in inches. The location letter and the station number together then give a unique position for each dosimeter.

Tables I to IV give the location and station number of each dosimeter used, along with the measured thermal neutron flux value. It should be noted that the flux measurements at the fuel surface (locations U1, U2, and U3) and on the center rod (location A) were made with dosimeters facing the core.

Following the irradiation, the dosimeters were counted on a 512-channel pulse-height analyzer using a sodium iodide crystal. The analyzer gave the number of counts under the gold 198 ($Au^{198}$) photopak for the gold dosimeters. The uranium-aluminum alloy wires were counted following a 5 day decay period. The pulse-height analyzer counted the activity of the fission product lanthanum 140 ($La^{140}$) and barium 140 ($Ba^{140}$).

The count data, elapsed time from reactor scram to counting, dosimeter data (mass and material), dosimeter position relative to the sodium iodide crystal, irradiation time and reactor power level are used as input for a computer program. This program calculates the absolute disintegration rate of each dosimeter. It then corrects this rate to the scram time and calculates the flux per watt of reactor power. The program then prints out the neutron flux adjusted to a reactor power of 60 MW and the equivalent fission power generation at 60 MW. The program calculates the thermal flux by correcting the cadmium-covered data for thermal neutron leakage through the covers and subtracting the cadmium-covered data from the bare fold data. Another program performs a similar analysis using the uranium-aluminum dosimeters.

## Gamma Heating Measurements

The gamma heating was measured by LiF thermoluminescent dosimeters (TLD). These were short plastic rods 0.04 inch (0.1 cm) in diameter and 0.25 inch (0.635 cm) long. They could not be used on the $D_2O$ tank surface (locations O,P,Q, and R) and the experiment capsule surface (locations D,E,F, and G) because their thickness interfered with the sliding fit of the test pieces.

The dosimeters were mounted in the same positions as the neutron flux dosimeters, separated by about 0.125 inch (0.318 cm) from them. Table V lists the dosimeter locations used in the four runs together with the measured gamma heating values. The location and station number code is the same as the one used in the neutron flux measurements.

Following the irradiation, the TLD's were counted and the results were analyzed by a computer program. This gave the gamma heating in watts per gram at a reactor power of 60 MW.

## RESULTS AND DISCUSSION

The MUR-G was operated at 10 KW and a rod bank height of 16 inches (40.7 cm) for 20 minutes for all four runs. The results of the computer analysis of the dosimeter measurements are given in tables I to V. These tables list the thermal neutron flux levels and the gamma heating at a reactor power of 60 MW for each dosimeter position. In order to calculate the fission power generated, the following relation should be used. A neutron flux of $2.13 \times 10^{13}$ neutrons/$cm^2$-sec generates 1 KW of power in every gram of $U^{235}$.

The uncertainties in the measurements are: ±21 percent in the absolute flux values, ±15 percent in the relative flux values, that is, in the comparison of flux values between runs, and ±23 percent in the gamma heating measurements.

### Neutron Flux Distribution

Figures 6 to 9 show the measured flux levels across the horizontal midplane of HT-2 at 0, 8, 16 and 28 inches (0, 20.3, 40.7, and 71 cm) east of the reactor core centerline for all four runs. These figures used the measurements at locations O, S, D, B, A, C, G, J, M, N, and R. The 8-inch (20.3 cm) position passes through the center of the 2-inch (5 cm) long fuel pins. Figures 10 to 13 show the flux distributions in the axial direction. The measurements in figure 10 were made on the surface of the experiment capsule on the side facing the core (location D) while figure 11 shows the axial flux distribution along the experiment capsule axis (location A). Figure 12 shows the flux levels along the HT-2 axis (location J) while figure 13 shows the measured flux at location N, which is 4 inches (10 cm) north of the HT-2 axis. Figures 10 to 13 include data for all four runs, and figures 6, 7, 8, and 12 also include curves for the unperturbed flux levels in HT-2. These were measured previously by the MUR operating staff and are included here as a reference point for the experimental measurements.

Effect of the $D_2O$. - Several ratios were calculated by comparing runs 1 (with $D_2O$) and 2 (with $H_2O$). These ratios represent an average along dosimeter locations parallel to the HT-2 axis. First, replacing the light water with $D_2O$ increases the average flux incident on the fuel by a factor of 2.4. This was calculated by comparing the results at locations U1, U2, and U3, where the dosimeters were mounted on the fuel surface facing the core. On the inside of the fuel (locations V1, V2, and V3), the average flux is 2.3 times higher. Similarly, the flux incident on the aluminum rod (location A), which passes between the three fuel pins along the capsule axis, is also 2.3 times greater with $D_2O$. At the HT-2 axis (location J), the flux at 8 inches (20.3 cm) east of the reactor core centerline is 2.5 times greater.

The greatest effect of the $D_2O$, however, occurs at locations more remote from the core. On the horizontal midplane, 4 inches (10 cm) north of the HT-2 axis (location N), the flux is 11 times higher for the $D_2O$ run than for the light water run.

Figures 10 to 13 show the fluxes in the three runs using $D_2O$ peak about 4 inches (10 cm) east of the core centerline. This is due to neutron absorption in the water outside the $D_2O$ tank, causing a flux depression just inside the tank.

Effect of the stainless steel liner. - The effect of the stainless steel liner was calculated by comparing the results of runs 1 and 3 at locations A, B, C, U and V. This comparison shows that the stainless steel liner reduces the flux inside the liner by 40 percent in the region near the fuel. The drop increases at dosimeter stations further east of the reactor core centerline, reaching 70 percent at 28 inches (71-cm) east of the core centerline. This is due to the neutrons having to penetrate a greater thickness of stainless steel, because at greater distances from the core the neutrons are diffusing axially and on the average, strike the liner more obliquely.

On the outside of the experiment capsule and liner, the perturbation due to the stainless steel depends on the location. Since most neutrons come from the reactor core side, the side facing away from the core shows a reduction of the flux due to neutron absorption in the stainless steel liner located between the dosimeter and the core. At this location (G) the flux drops 45 percent, while at the side facing the core (location D), the drop is only 9 percent. On the top of the capsule (location E) the flux is 28 percent lower, while at the bottom it is 35 percent lower (location F). These averages are for the region from 0 to 16 inches (0 to 40.6 cm) east of the core centerline.

Comparing the measurements inside the stainless steel tube simulating the coolant gas flow guide (location V) with those on the center rod (location A) in run 4 allows the calculation of the flux drop across the tube. Using the average of the three tubes, the flux is 10 percent lower at the inside.

Effect of the molybdenum fuel holder shell. - The average flux drop across the 0.094-inch (0.238-cm) thick molybdenum shell is about 20 percent. This result is obtained by comparing the flux levels on the outside of the shell (location B) with those on the inside (location A) in run 4. The effect of the stainless steel tubes simulating the coolant gas flow guides was neglected for this comparison.

Effect of the fuel. - The presence of the fuel pins affects only the region in the vicinity of the fuel pins. Effects at the center of the pins, the fuel surface and between the three fuel pins are considered. All comparisons are made at the fuel midplane, that is, a plane perpendicular to the HT-2 axis and 8 inches (20.3 cm) east of the reactor core centerline.

A comparison of runs 1 and 4 (see tables I and IV) shows that the flux levels for run 4 are higher at dosimeter locations north of the HT-2 axis (location O) and lower at locations south of the axis (location R) than the flux levels of run 1. Similarly, run 4 has higher flux levels above the HT-2 axis (location P) and lower levels below it (location Q) than run 1. Figures 6 to 9 show the curves for runs 1 and 4 crossing each other near the $D_2O$ tank centerline. This indicates that the tank has been rotated. It has been estimated that a rotation of the top of the $D_2O$ tank about 5 degrees toward the core in run 4 would account for this effect. In order to make comparisons between runs 1 and 4, the rotation had to be taken into account.

The measured flux levels in both runs should be the same everywhere except near the fuel. Comparing the measured flux values along the capsule centerline (location A) between the two runs results in a correction factor of 1.19. This factor is applied to the measurements in run 4.

The flux drop at the experiment capsule centerline (dosimeter A-8) due to the presence of the fuel pins is 31 percent when using the corrected measurements for run 4. Similarly, the flux at the surface of the fuel (location V) is 34 percent lower. This value is useful when it is necessary to estimate the effects of a fueled experiment, and only data for an unfueled run are available. For the three runs using fuel (runs 1 to 3), the ratio of the flux incident on the fuel to the flux at the center of the fuel pin was 2.9 : 1.

In regions north of the experiment capsule centerline, the measured flux levels are higher for run 2 with $H_2O$ than the flux levels in the empty test hole flooded with $H_2O$ (unperturbed case). This is shown in figure 12 for location J, the HT-2 centerline. Here, the partial void in the experiment capsule permits more neutrons to travel further from the core before they are absorbed, resulting in a higher flux level near the capsule.

## Gamma Heating Distribution

Figure 14 shows the results of the gamma heating measurements along the experiment capsule axis (location A). The curves for the unperturbed cases were extrapolated from previous measurements by the MUR staff. The gamma heating values measured in the experiments are presented in table V. Due to the scatter of the measurements and the large uncertainty (±23 percent) only the fitted curves are shown. They do, however, show the general pattern of the gamma heating distribution, such as the increase in the vicinity of the fuel pins due to the secondary gamma radiation produced by the fissions in the mock-up fuel pins.

The measured gamma heating at the capsule axis ranged from about 3.5 watts/gram at the core centerline to about 0.3 watts/gram at 28 inches (71 cm) east of the core centerline. The high gamma heating rates measured near the fuel midplane between the three fuel pins are applicable only in the regions very near the fuel pins. Here the gamma heating reached as high as 8.47 watts/gram between the three fuel pins, and 15.1 watts/gram inside the fuel pins at a reactor power of 60 MW. However, at points more than 2 inches (5 cm) away from the fuel, the contribution from the secondary gamma radiation becomes negligible.

## SUMMARY OF RESULTS

Four experiments were run in the Plum Brook Mock-Up Reactor in the HT-2 test hole. These experiments were run to determine the effects of $D_2O$ on the thermal neutron flux in a fueled experiment capsule surrounded by $D_2O$ contained in an aluminum tank. The following results were obtained:

1. Replacing the $H_2O$ with $D_2O$ increases the thermal neutron flux incident on the fuel 2.4 times to $4.13 \times 10^{13}$ neutrons/$cm^2$-sec at 60 MW.

2. The $D_2O$ increases the flux at the center of the fuel by a factor of 2.3 to $1.22 \times 10^{13}$ neutrons/$cm^2$-sec at 60 MW.

3. The neutron flux at the centerline of the test hole was by a factor of 2.5 higher at $5.27 \times 10^{13}$ neutrons/$cm^2$-sec at 60 MW when $D_2O$ was used.

4. At the side facing away from the core, the $D_2O$ increased the flux about a factor of 11 to $3.5 \times 10^{13}$ neutrons/cm$^2$-sec.

5. An 0.250-inch (0.635-cm) thick stainless steel capsule liner lowered the flux on the inside of the capsule by 40 percent, while the molybdenum shell of the fuel holder caused a flux drop of about 22 percent.

6. Twenty-seven grams of $U^{235}$ contained in three 2-inch (5-cm) long fuel pins resulted in the following flux depressions near the fuel. At the capsule axis between the fuel pins, the flux dropped 31 percent. On the surface of the fuel pins, the flux was lowered by 34 percent. The ratio of the flux incident on the surface of the fuel to the flux at the center of the fuel was 2.9 : 1.

7. The gamma heating at the capsule axis ranged from 3.5 watts per gram at 60 MW near the core centerline to about 0.3 watt per gram at 60 MW at 28 inches (71 cm) east of the core centerline. The gamma heating near the fuel was higher due to fissioning in the fuel pins.

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TABLE I. - THERMAL NEUTRON FLUX LEVELS MEASURED IN HT-2.

RUN 1 ( FUEL, $D_2O$, STAINLESS STEEL LINER )

| Dosimeter location | | Thermal neutron flux, neutrons/$cm^2$-sec at 60 MW, x $10^{13}$ | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Dosimeter station number | | | | | | | | | | | | | |
| | | 0 | 2 | 4 | 6 | 7.5 | 8 | 8.5 | 10 | 12 | 14 | 16 | 20 | 24 | 28 |
| Center rod | A | 7.45 | - | 6.98 | - | - | 4.06 | - | - | 4.87 | - | 3.08 | 1.60 | 0.58 | 0.23 |
| Molybdenum shell | B | - | 8.04 | 7.91 | 7.90 | - | 6.86 | - | 6.48 | 5.47 | 4.60 | - | - | - | - |
| | C | - | 5.37 | 5.67 | 5.49 | - | 4.14 | - | 4.47 | 4.13 | 3.10 | - | - | - | - |
| Capsule surface | D | 12.31 | - | 12.78 | - | - | 12.30 | - | - | 9.61 | - | 5.66 | 2.56 | .80 | .25 |
| | E | 7.70 | - | 8.21 | - | - | 7.36 | - | - | 5.85 | - | 3.61 | 1.83 | .66 | .25 |
| | F | 7.40 | - | 7.90 | - | - | 7.16 | - | - | 5.86 | - | 3.65 | 1.79 | .70 | .25 |
| | G | 5.43 | - | 5.53 | - | - | 4.92 | - | - | 4.19 | - | 2.80 | 1.56 | .66 | .24 |
| $D_2O$ tank interior | H | - | - | - | - | - | 6.33 | - | - | - | - | - | - | .62 | - |
| | I | - | - | - | - | - | 6.17 | - | - | - | - | - | - | .74 | - |
| | J | - | - | - | - | - | 5.27 | - | - | - | - | - | - | .71 | - |
| | K | - | - | - | - | - | 5.62 | - | - | - | - | - | - | .70 | - |
| | L | - | - | - | - | - | 5.37 | - | - | - | - | - | - | .62 | - |
| | M | - | - | - | - | - | 4.60 | - | - | - | - | - | - | .68 | - |
| | N | - | - | - | - | - | 3.20 | - | - | - | - | - | - | .58 | - |
| $D_2O$ tank surface | O | 25.42 | - | - | - | - | 21.48 | - | - | - | - | 8.73 | 4.57 | - | .917 |
| | P | 6.21 | - | - | - | - | 5.36 | - | - | - | - | 3.10 | - | .65 | .22 |
| | Q | 6.24 | - | 8.47 | - | - | 4.76 | - | - | - | - | 2.32 | - | .53 | - |
| | R | 2.87 | - | 3.48 | - | - | 2.78 | - | - | 2.34 | - | 1.54 | - | - | .183 |
| Fuel pins | U1 | - | - | - | - | 4.95 | 4.13 | 5.65 | - | - | - | - | - | - | - |
| | V1 | - | - | 6.04 | - | 1.85 | 1.22 | - | - | 4.32 | - | - | - | - | - |
| | U2 | - | - | - | - | - | 3.02 | - | - | - | - | - | - | - | - |
| | V2 | - | - | 5.72 | - | - | 1.08 | - | - | 4.07 | - | - | - | - | - |
| | U3 | - | - | - | - | - | 2.98 | - | - | - | - | - | - | - | - |
| | V3 | - | - | 5.28 | - | - | 1.09 | - | - | 3.80 | - | - | - | - | - |

TABLE II. - THERMAL NEUTRON FLUX LEVELS MEASURED IN HT-2.

RUN 2 ( FUEL, $H_2O$, STAINLESS STEEL LINER )

| Dosimeter location | | Thermal neutron flux, neutrons/$cm^2$- sec at 60 MW, x $10^{13}$ | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Dosimeter station number | | | | | | | | | | | | | |
| | | 0 | 2 | 4 | 6 | 7 | 8 | 9 | 10 | 12 | 14 | 16 | 20 | 24 | 28 |
| Center rod | A | 5.28 | - | 3.51 | 2.93 | 2.36 | 1.72 | 1.84 | 2.47 | 2.21 | - | 1.60 | 0.70 | 0.22 | 0.07 |
| Molybdenum shell | B | - | 5.02 | 4.81 | 4.52 | - | 4.02 | - | 3.76 | 3.22 | 2.49 | - | - | - | - |
| | C | - | 3.09 | 3.11 | 2.93 | - | 2.09 | - | 2.29 | 2.01 | 1.49 | - | - | - | - |
| Capsule surface | D | 9.38 | - | 8.71 | - | - | 8.01 | - | - | 6.18 | -- | 3.36 | 1.26 | .31 | .08 |
| | E | 5.38 | - | 4.33 | - | - | 3.77 | - | - | 2.94 | - | 1.69 | 0.73 | .22 | .06 |
| | F | 5.05 | - | 4.11 | - | - | 3.60 | - | - | 2.77 | - | 1.62 | .71 | .22 | .06 |
| | G | 4.21 | - | 2.92 | - | - | 2.34 | - | - | 1.92 | - | 1.21 | .59 | .19 | .06 |
| $D_2O$ tank interior | H | 2.48 | - | 2.62 | - | - | 2.53 | - | - | 1.81 | - | 0.99 | .41 | .13 | .04 |
| | I | 2.65 | - | 2.50 | - | - | 2.26 | - | - | 1.65 | - | .97 | .41 | .14 | .04 |
| | J | 3.38 | - | 2.66 | - | - | 2.21 | - | - | 1.74 | - | 1.03 | .47 | .16 | .05 |
| | K | 2.47 | - | 2.23 | - | - | 2.04 | - | - | 1.51 | - | .86 | .38 | .13 | .04 |
| | L | 1.78 | - | 1.87 | - | - | 1.84 | - | - | 1.32 | - | .76 | .35 | .13 | .04 |
| | M | 1.06 | - | 1.07 | - | - | 0.99 | - | - | 0.72 | - | .38 | .17 | .06 | .02 |
| | N | 0.36 | - | 0.32 | - | - | .30 | - | - | .22 | - | .13 | .08 | .02 | .01 |
| $D_2O$ tank surface | O | - | - | - | - | - | - | - | - | - | - | - | - | - | - |
| | P | - | - | 2.85 | - | - | 2.81 | - | - | 1.92 | - | 1.04 | .43 | .14 | .04 |
| | Q | - | - | 1.99 | - | - | 1.94 | - | - | 1.41 | - | 0.81 | .37 | .13 | .04 |
| | R | - | - | 0.25 | - | - | 0.23 | - | - | 0.18 | - | .10 | .05 | .02- | .01 |
| | U1 | - | - | - | - | 2.67 | 1.53 | 2.01 | - | - | - | - | - | - | - |
| | V1 | - | - | 3.66 | - | 0.65 | 0.50 | 1.29 | - | 2.32 | - | - | - | - | - |
| | U2 | - | - | - | - | - | 1.61 | - | - | - | - | - | - | - | - |
| | V2 | - | - | 3.44 | - | -- | 0.50 | - | - | 2.10 | - | - | - | - | - |
| | U3 | - | - | - | - | - | 1.21 | - | - | - | - | - | - | - | - |
| | V3 | - | - | 3.06 | - | - | 0.42 | - | - | 1.88 | - | - | - | - | - |

TABLE III. - THERMAL NEUTRON FLUX LEVELS MEASURED IN HT-2 .

RUN 3 ( FUEL, $D_2O$, NO STAINLESS STEEL LINER )

| Dosimeter location | | Thermal neutron flux, neutrons/$cm^2$-sec at 60 MW, x $10^{13}$ | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Dosimeter station number | | | | | | | | | | | | | | | |
| | | 0 | 2 | 4 | 5 | 6 | 7.25 | 8 | 8.75 | 10 | 11 | 12 | 14 | 16 | 20 | 24 | 28 |
| Center rod | A | 11.8 | 9.27 | 9.36 | - | 9.26 | - | 6.70 | - | 8.31 | - | 7.11 | - | 4.97 | 3.00 | 1.46 | 0.75 |
| Molybdenum shell | B | - | 12.80 | 12.90 | - | 12.20 | - | 10.60 | - | 10.30 | - | 9.33 | 7.76 | - | - | - | - |
| | C | - | 9.28 | 9.79 | - | 9.57 | - | 7.49 | - | 8.01 | - | 7.48 | 5.69 | - | - | - | - |
| Capsule surface | D | 12.50 | - | 14.50 | - | - | - | 13.50 | - | - | - | 10.20 | - | 6.82 | 3.76 | 1.63 | .73 |
| | E | 11.00 | - | 10.90 | - | - | - | 9.42 | - | - | - | 8.07 | - | 5.41 | 3.38 | 1.68 | .76 |
| | F | 11.30 | - | 11.80 | - | - | - | 10.80 | - | - | - | 8.90 | - | 6.03 | 3.37 | 1.65 | .76 |
| | G | 10.70 | - | 10.10 | - | - | - | 8.98 | - | - | - | 7.52 | - | 5.34 | 3.35 | 1.65 | .79 |
| $D_2O$ tank interior | H | 5.85 | - | - | - | - | - | 6.77 | - | - | - | 5.59 | - | 3.90 | - | 1.09 | .49 |
| | I | 7.74 | - | - | - | - | - | 7.86 | - | - | - | 6.42 | - | 4.58 | - | 1.35 | .63 |
| | J | 9.34 | - | 9.28 | - | - | - | 8.10 | - | - | - | 6.66 | - | 4.69 | 2.72 | 1.37 | .63 |
| | K | 8.60 | - | - | - | - | - | 8.40 | - | - | - | 6.75 | - | 4.59 | - | 1.34 | .61 |
| | L | 7.42 | - | - | - | - | - | 7.74 | - | - | - | 6.18 | - | 4.08 | - | 1.08 | .50 |
| | M | 5.82 | - | - | - | - | - | 6.51 | - | - | - | 5.35 | - | 3.61 | - | 1.18 | .53 |
| | N | 3.57 | - | 4.66 | - | - | - | 4.52 | - | - | - | 3.81 | - | 2.73 | 1.68 | 0.85 | .40 |
| $D_2O$ tank surface | O | 25.40 | - | 23.70 | - | - | - | 20.80 | - | - | - | 15.60 | - | 9.14 | 4.06 | 1.46 | .53 |
| | Q | 22.20 | - | 22.00 | - | - | - | 19.90 | - | - | - | 15.20 | - | 8.77 | 4.07 | 1.45 | .59 |
| Fuel pins | U1 | - | - | - | - | - | - | 5.91 | - | - | - | - | - | - | - | - | - |
| | V1 | - | - | - | 10.50 | - | 4.99 | 1.75 | 1.79 | - | 8.32 | - | - | - | - | - | - |
| | $U2^a$ | - | - | - | - | - | 6.98 | 5.54 | 5.51 | - | - | - | - | - | - | - | - |
| | $U2^b$ | - | - | - | - | - | 5.85 | - | - | - | - | - | - | - | - | - | - |
| | $U2^c$ | - | - | - | - | - | 5.12 | - | - | - | - | - | - | - | - | - | - |
| | $U2^d$ | - | - | - | - | - | 6.73 | - | - | - | - | - | - | - | - | - | - |
| | V2 | - | - | - | 9.95 | - | 3.00 | 2.03 | 1.86 | - | 7.66 | - | - | - | - | - | - |
| | U3 | - | - | - | - | - | - | 4.98 | - | - | - | - | - | - | - | - | - |
| | V3 | - | - | - | 11.00 | - | 4.72 | 1.67 | 1.65 | - | 7.88 | - | - | - | - | - | - |

Dosimeter facing: a - south, b - top, c - north, d - bottom .

TABLE IV. - THERMAL NEUTRON FLUX LEVELS MEASURED IN HT-2.

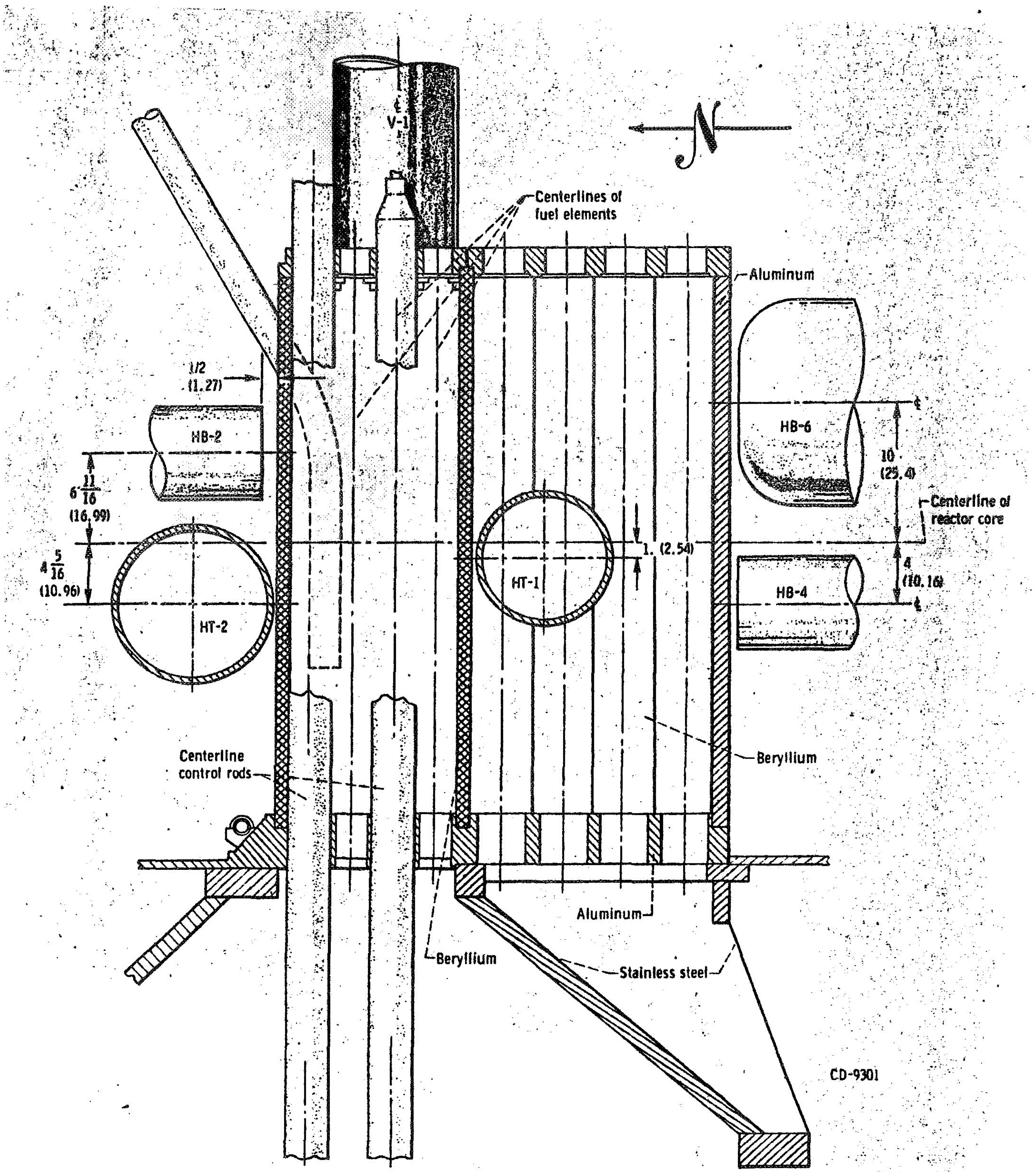
RUN 4 ( NO FUEL, $D_2O$, STAINLESS STEEL LINER )

| Dosimeter location | | Thermal neutron flux, neutrons/$cm^2$- sec at 60 MW, x $10^{13}$ | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Dosimeter station number | | | | | | | | | | | | | |
| | | 0 | 2 | 4 | 5 | 6 | 8 | 10 | 11 | 12 | 14 | 16 | 20 | 24 | 28 |
| Center rod | A | 8.10 | 5.76 | 5.41 | - | 5.53 | 4.95 | 4.45 | - | 4.12 | - | 2.61 | 1.35 | 0.51 | 0.20 |
| Molybdenum shell | B | - | 7.50 | - | - | - | 6.56 | - | - | - | 4.12 | - | - | - | - |
| | C | - | 5.66 | - | - | - | 4.91 | - | - | - | 3.00 | - | - | - | - |
| Capsule surface | D | 10.95 | - | 11.50 | - | - | 11.10 | - | - | 7.96 | - | 5.07 | 2.28 | .75 | .25 |
| | E | 9.65 | - | 9.14 | - | - | 8.90 | - | - | 6.95 | - | 4.25 | 2.05 | .78 | .27 |
| | F | 8.12 | - | 6.03 | - | - | 5.75 | - | - | 4.44 | - | 2.91 | 1.57 | .62 | .24 |
| | G | 8.25 | - | 5.57 | - | - | 5.56 | - | - | 4.25 | - | 2.74 | 1.55 | .65 | .25 |
| $D_2O$ tank interior | H | 9.15 | - | - | - | - | - | - | - | 7.39 | - | - | 2.13 | - | .33 |
| | I | 8.07 | - | - | - | - | - | - | - | 6.28 | - | - | 2.04 | - | .33 |
| | J | - | - | 6.10 | - | - | - | - | - | - | - | 3.03 | - | - | .28 |
| | K | - | - | 5.27 | - | - | - | - | - | 3.08 | - | 2.50 | - | .61 | - |
| | L | 3.86 | - | - | - | - | - | - | - | 3.18 | - | - | 1.14 | - | .24 |
| | M | 4.54 | - | 5.17 | - | - | - | - | - | - | - | 2.58 | - | .67 | - |
| | N | - | - | - | - | - | 3.63 | - | - | 2.98 | - | - | 1.14 | - | .25 |
| | S | 18.60 | - | 18.00 | - | - | 16.60 | - | - | 12.20 | - | - | - | - | .24 |
| $D_2O$ tank surface | O | 20.60 | - | 19.60 | - | - | 18.10 | - | - | 13.40 | - | 7.53 | 3.14 | .90 | .27 |
| | P | 9.35 | - | 10.50 | - | - | 9.80 | - | - | 7.52 | - | 4.44 | 2.07 | .77 | .31 |
| | Q | 2.74 | - | 3.70 | - | - | 3.46 | - | - | 2.67 | - | 1.71 | 0.97 | .45 | .19 |
| | R | - | - | - | - | - | 3.02 | - | - | 2.50 | - | 1.66 | 1.01 | .45 | .21 |
| | V1 | - | - | - | 5.57 | - | 4.19 | - | 3.47 | - | - | - | - | - | - |
| | V2 | - | - | - | 4.90 | - | 4.52 | - | 3.67 | - | - | - | - | - | - |
| | V3 | - | - | - | 5.31 | - | 4.48 | - | 3.78 | - | - | - | - | - | - |

TABLE V. - GAMMA HEATING MEASUREMENTS IN HT-2

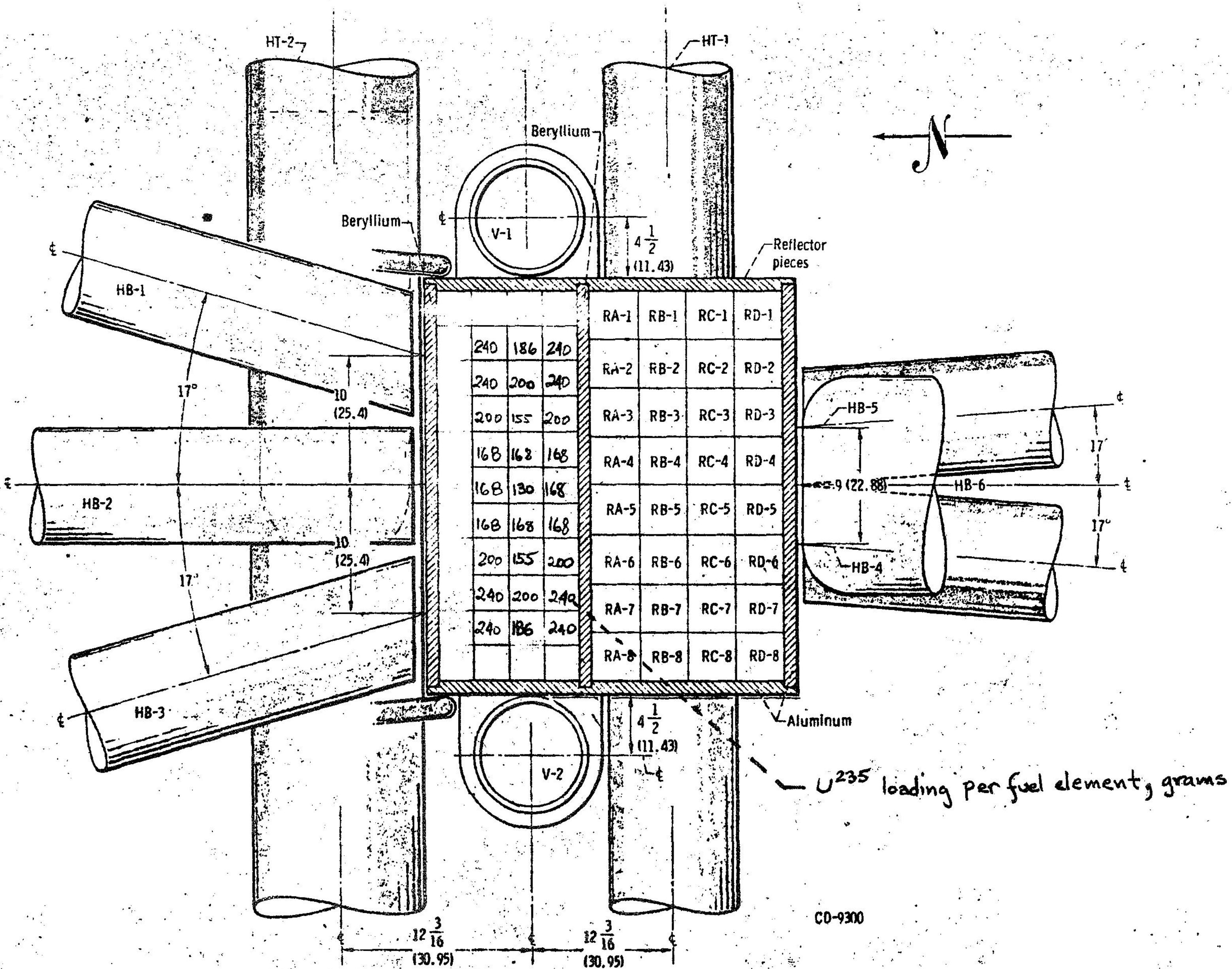
Runs 1 - 4.

| Run 1 | | Run 2 | | Run 3 | | Run 4 | |
|---|---|---|---|---|---|---|---|
| Dosimeter location and station number | Gamma heating, watts / gram $H_2O$ at 60 MW | Dosimeter location and station number | Gamma heating, watts / gram $H_2O$ at 60 MW | Dosimeter location and station number | Gamma heating, watts / gram $H_2O$ at 60 MW | Dosimeter location and station number | Gamma heating, watts / gram $H_2O$ at 60 MW |
| A4 | 2.87 | A4 | 2.20 | A0 | 2.48 | A0 | 3.41 |
| A8 | 4.56 | A12 | 1.46 | A2 | 3.14 | A2 | 3.90 |
| A12 | 2.13 | | | A4 | 3.34 | A4 | 2.79 |
| | | B8 | 3.01 | A6 | 2.93 | A6 | 2.81 |
| B4 | 3.58 | B12 | 1.97 | A8 | 8.47 | A8 | 3.09 |
| B8 | 4.05 | | | A10 | 2.54 | A10 | 2.85 |
| B12 | 2.64 | | | A12 | 1.94 | A12 | 2.28 |
| | | | | A20 | 0.74 | A16 | 1.49 |
| I4 | 1.31 | | | A24 | .42 | A20 | 0.89 |
| I20 | 0.46 | | | A28 | .26 | A24 | .41 |
| J4 | 1.17 | | | | | A28 | .20 |
| J20 | 0.46 | | | H4 | 1.28 | | |
| K4 | 1.29 | | | H20 | 1.52 | B1 | 4.14 |
| K20 | 0.45 | | | I4 | 1.28 | B7 | 4.04 |
| L4 | 1.23 | | | I20 | 0.50 | B13 | 1.97 |
| L20 | 0.48 | | | K4 | 1.57 | | |
| M4 | .87 | | | K20 | 0.51 | C1 | 2.16 |
| M20 | .40 | | | L4 | 1.42 | C7 | 2.28 |
| N4 | .72 | | | L20 | 0.49 | C13 | 1.38 |
| N20 | .33 | | | M4 | 1.01 | | |
| | | | | M20 | 0.43 | | |
| O4 | 4.55 | | | | | | |
| O12 | 3.12 | | | V1-5 3/8 | 2.51 | | |
| O20 | 0.85 | | | V1-7 3/4 | 15.10 | | |
| P4 | 1.43 | | | V1-8 3/8 | 13.10 | | |
| P12 | 0.94 | | | V1-11 3/8 | 2.51 | | |
| P20 | .47 | | | | | | |
| Q12 | .76 | | | | | | |
| Q20 | .39 | | | | | | |
| R20 | .17 | | | | | | |




(a) Side view at core north-south vertical midplane.

Figure 1.- Mockup reactor. (All dimensions are in inches (cm).)




(b) Top view.

Figure 1. - Concluded.

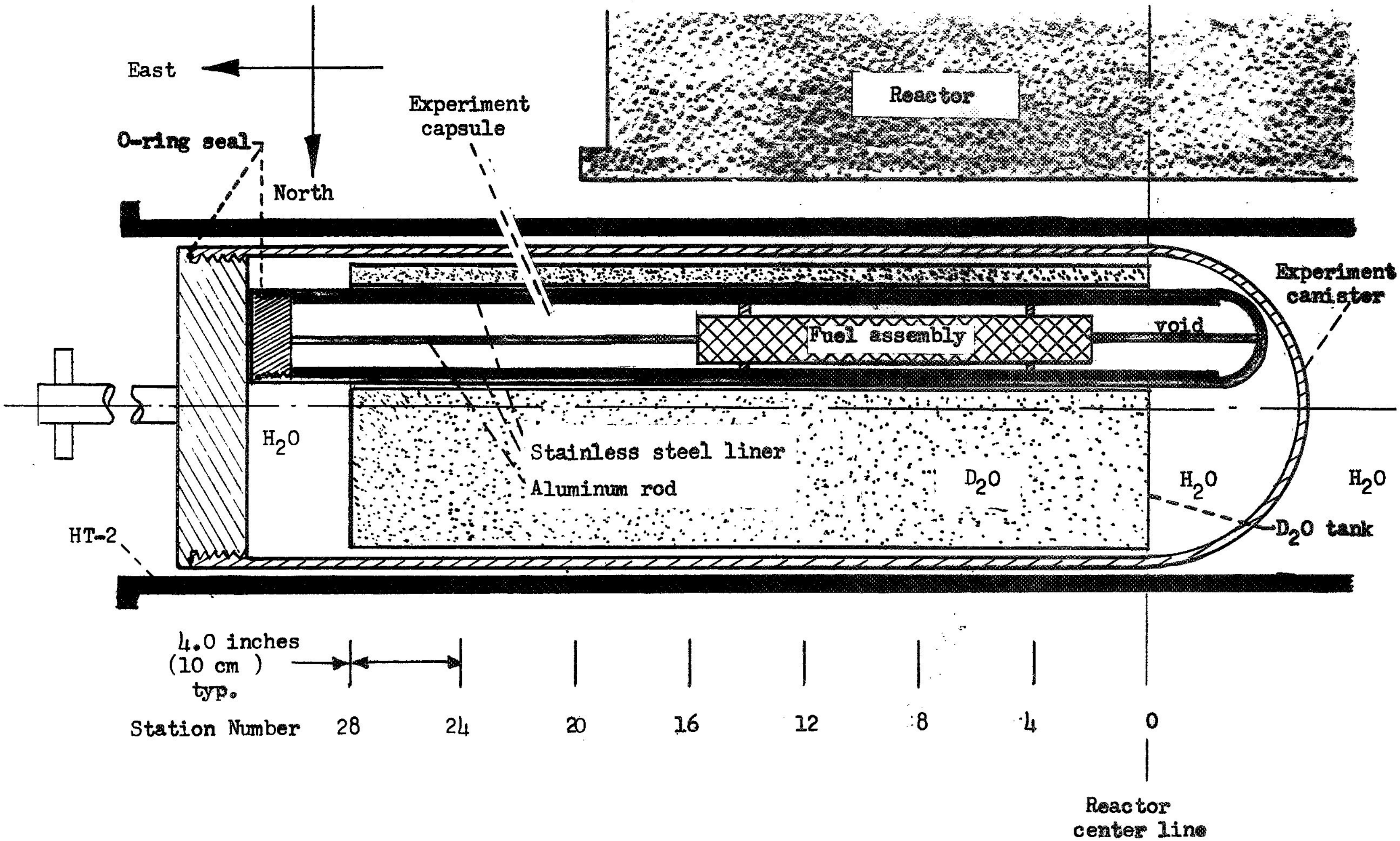



Figure 2. - Top view of experiment apparatus with fully inserted capsule.

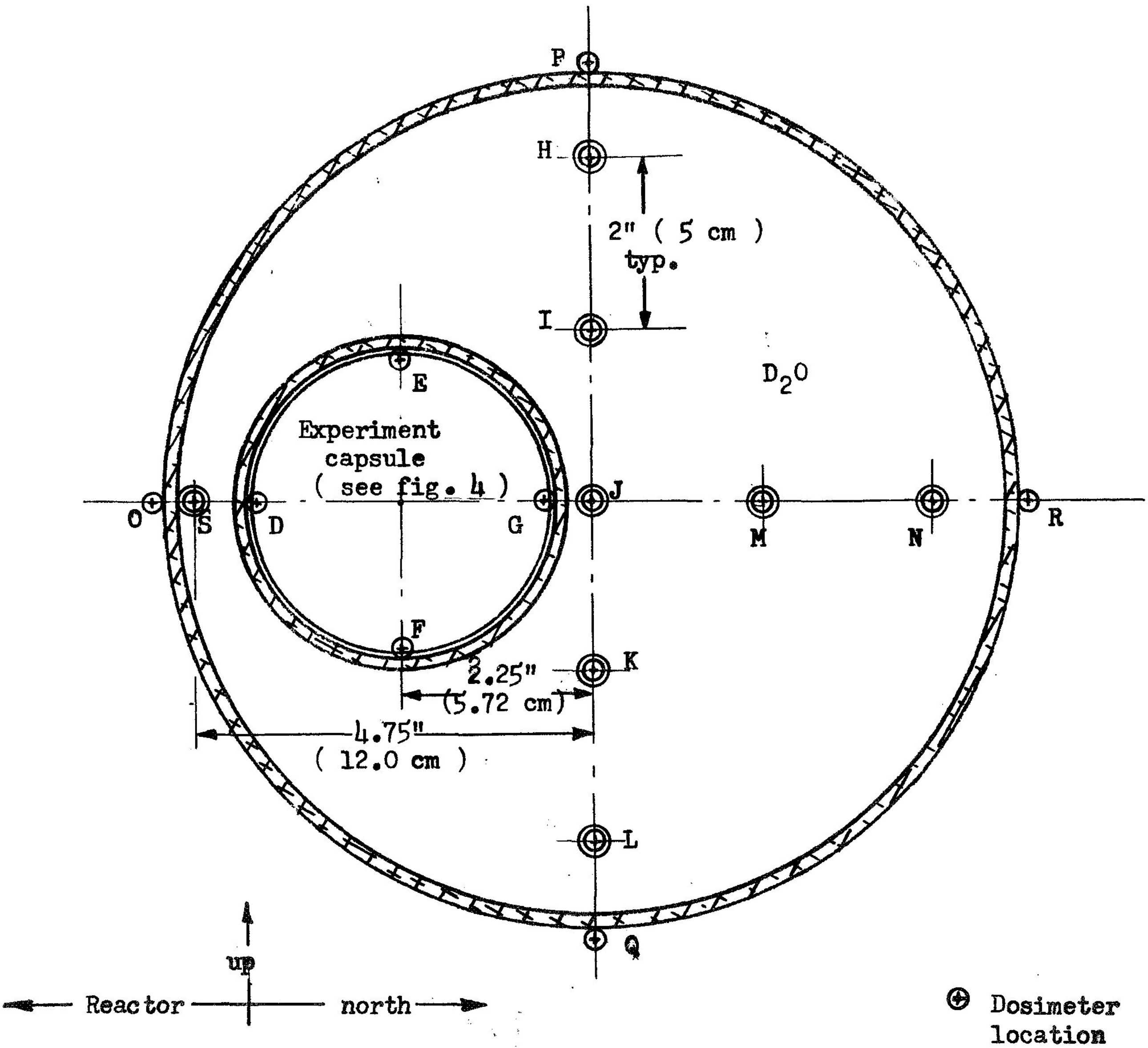



Figure 3. - Cross section of the 10 inch ( 25.4 centimeter ) diameter $D_2O$ tank showing the dosimeter locations.

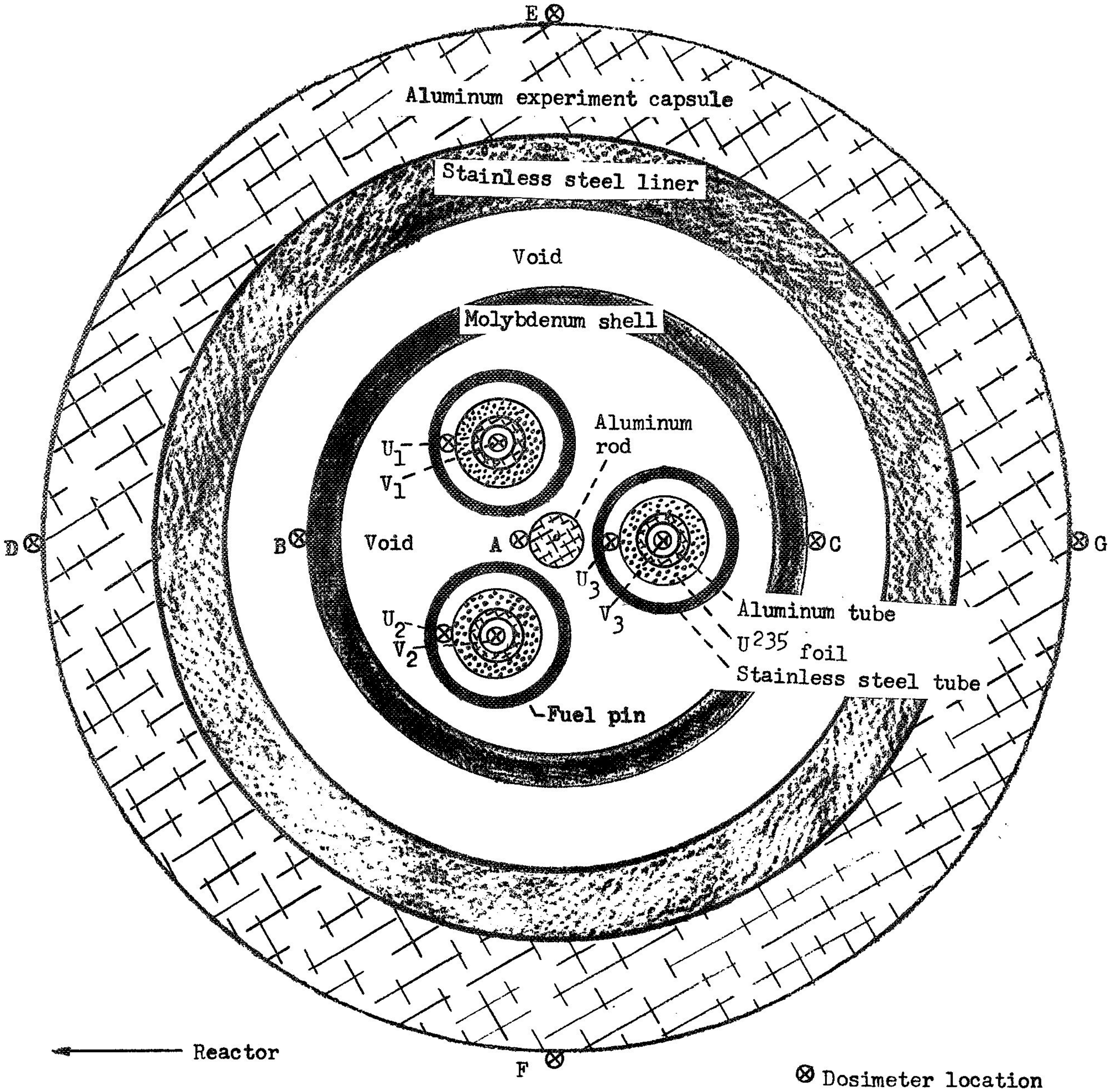



Figure 4. - Cross section of the experiment capsule showing the dosimeter locations.

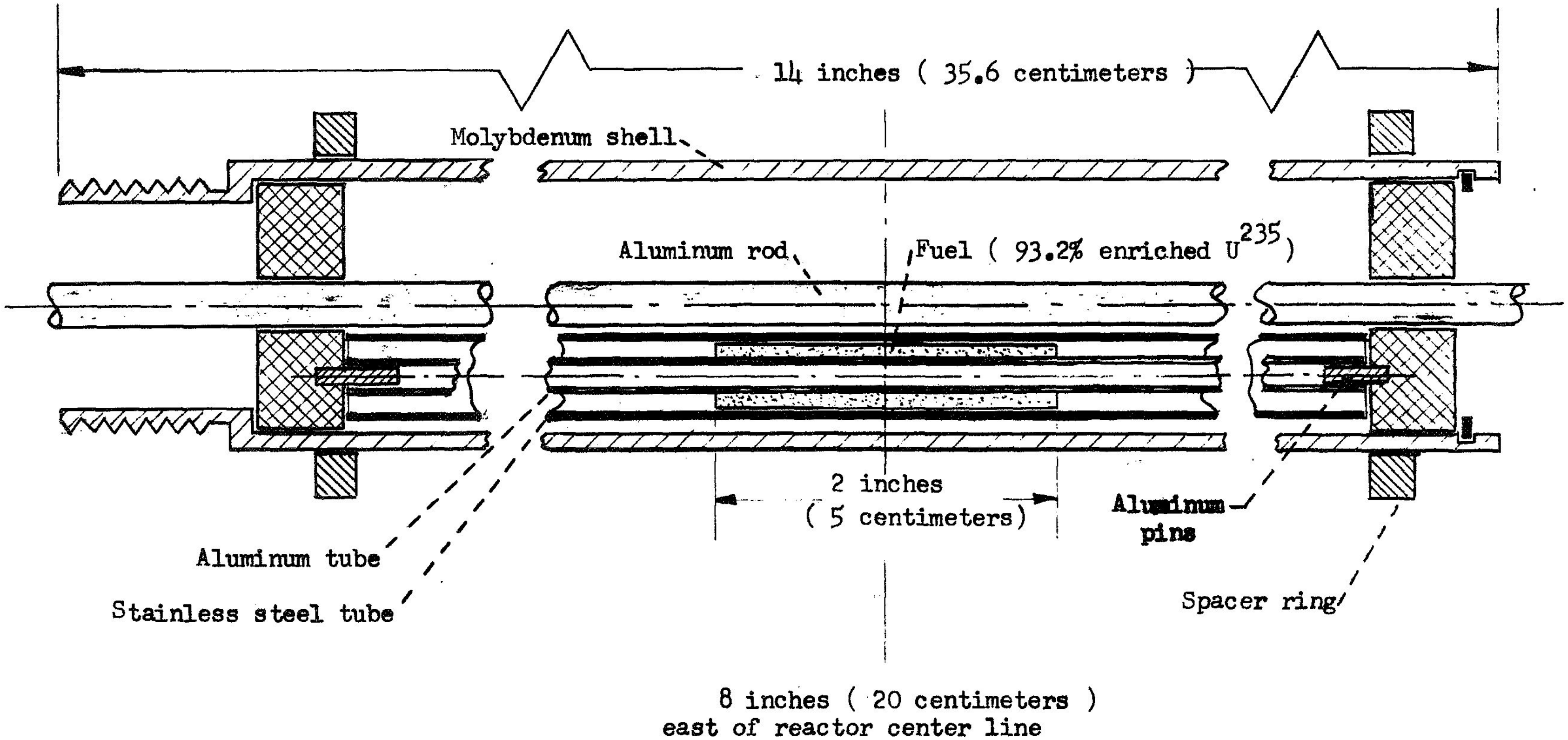



Figure 5. - Details of the fuel assembly.

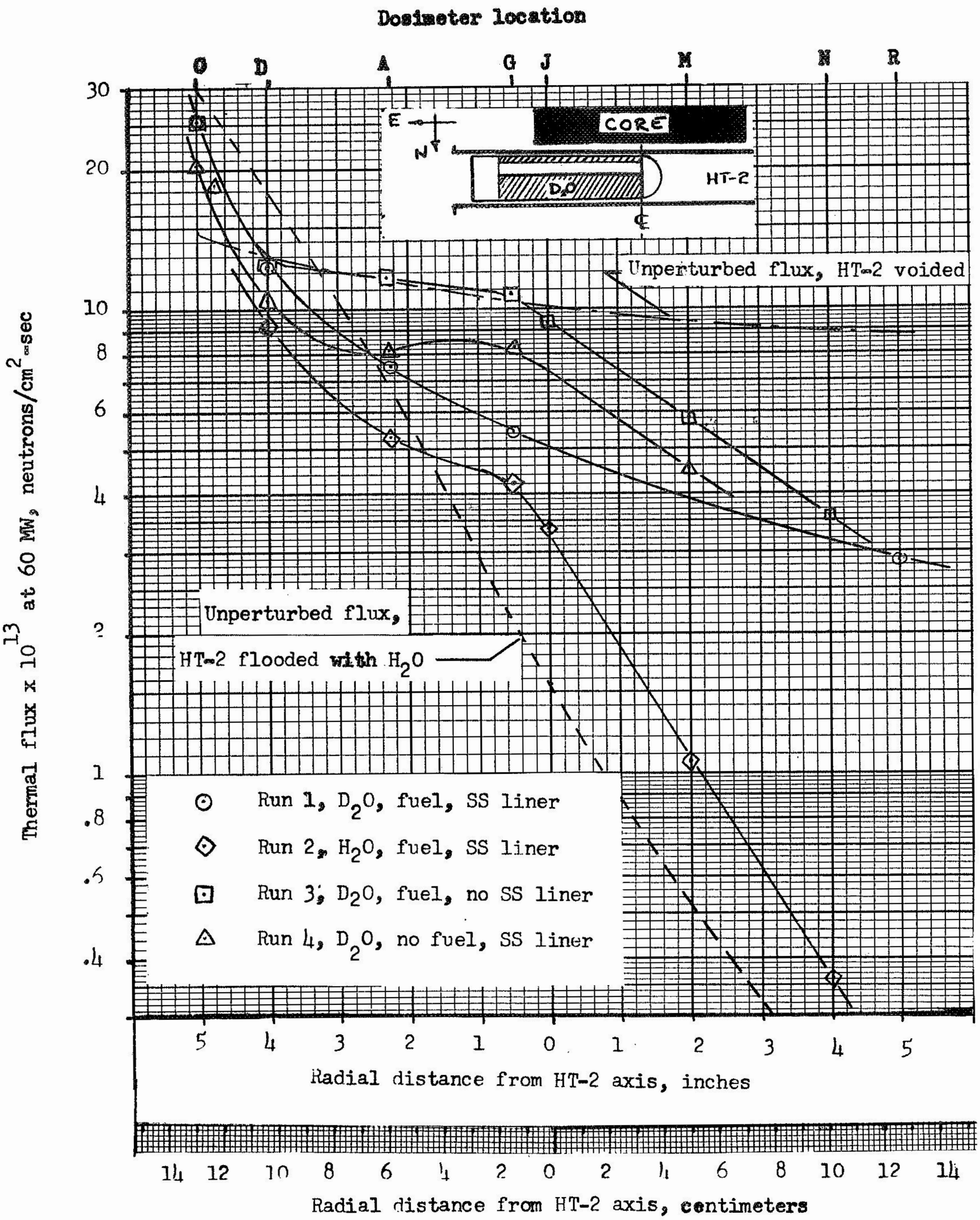



Figure 6.- Thermal neutron flux distribution on the horizontal midplane of HT-2 as a function of distance from the HT-2 axis, at the reactor core centerline.

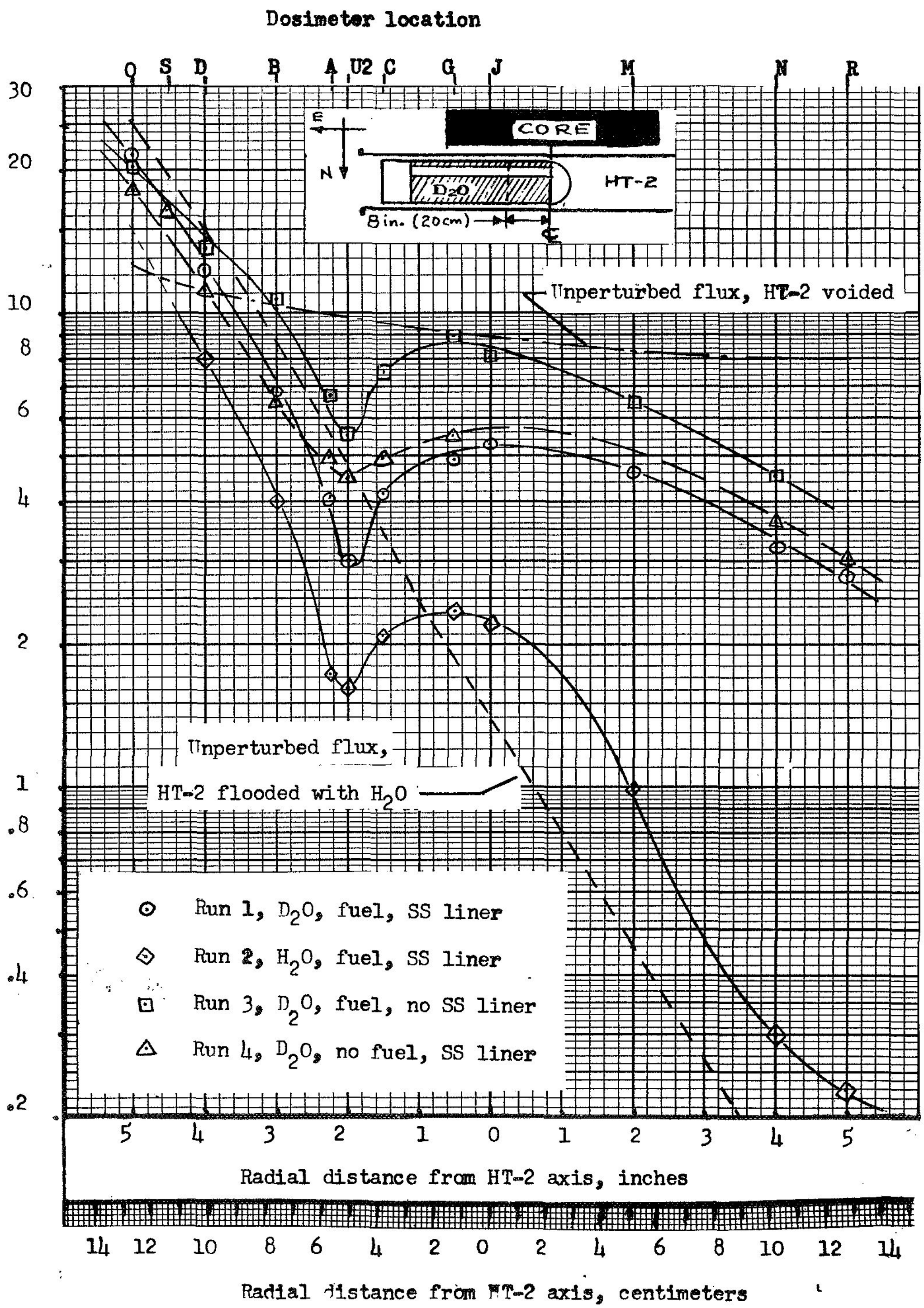



Figure 7.- Thermal neutron flux distribution on the horizontal midplane of HT-2 as a function of distance from the HT-2 axis, 8 inches (20 centimeters) east of the core centerline.

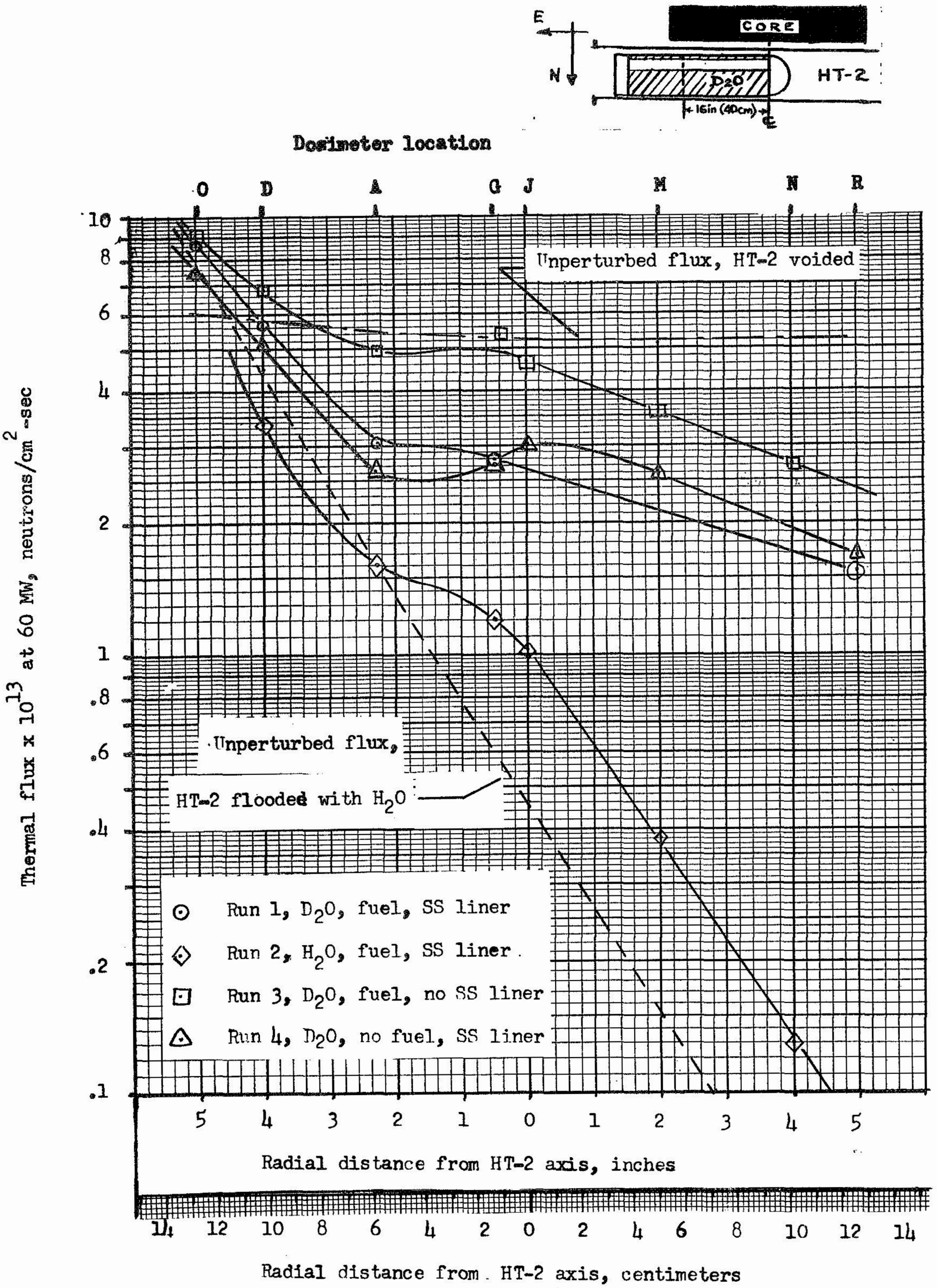



Figure 8.- Thermal neutron flux distribution on the horizontal midplane of HT-2 as a function of distance from the HT-2 axis, 16 inches (40 centimeters ) east of the core centerline.

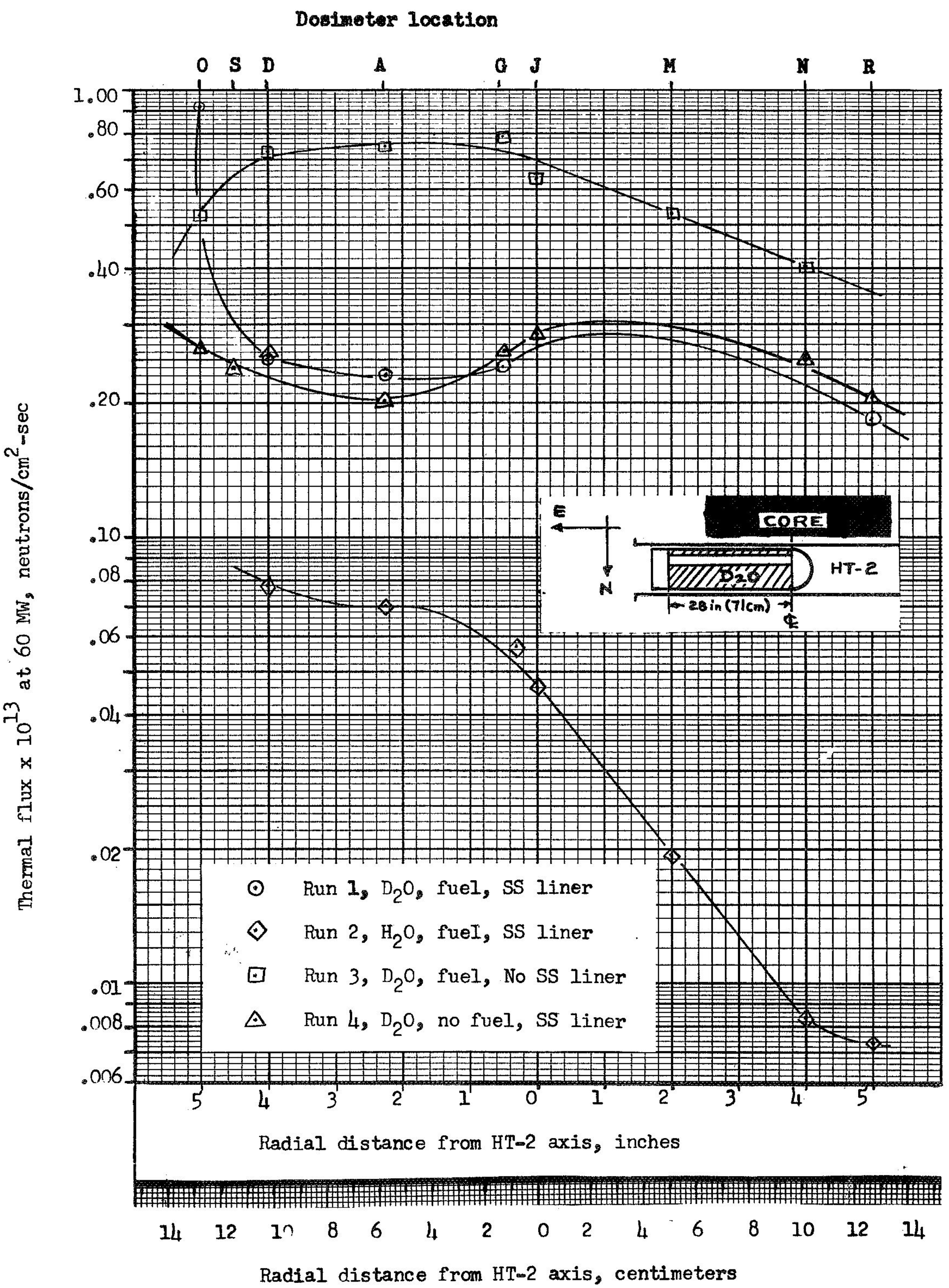



Figure 9.- Thermal neutron flux distribution on the horizontal midplane of HT-2 as a function of distance from the HT-2 axis, 28 inches ( 71 centimeters) east of the core centerline.

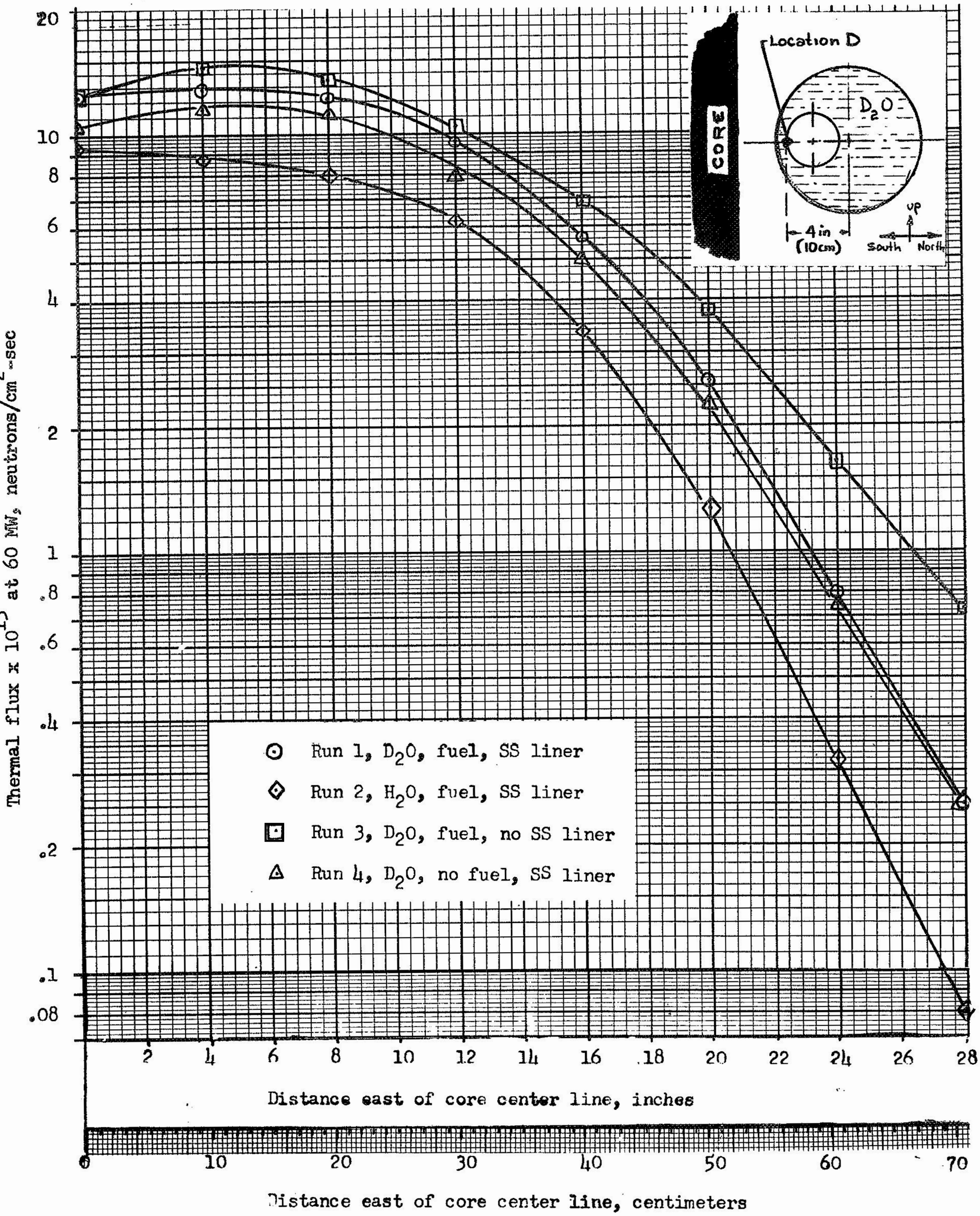



Figure 10.- Thermal neutron flux distribution as a function of distance east of the core centerline, as measured on the experiment capsule, 4 inches ( 10 centimeters ) south of the HT-2 axis. ( Location D )

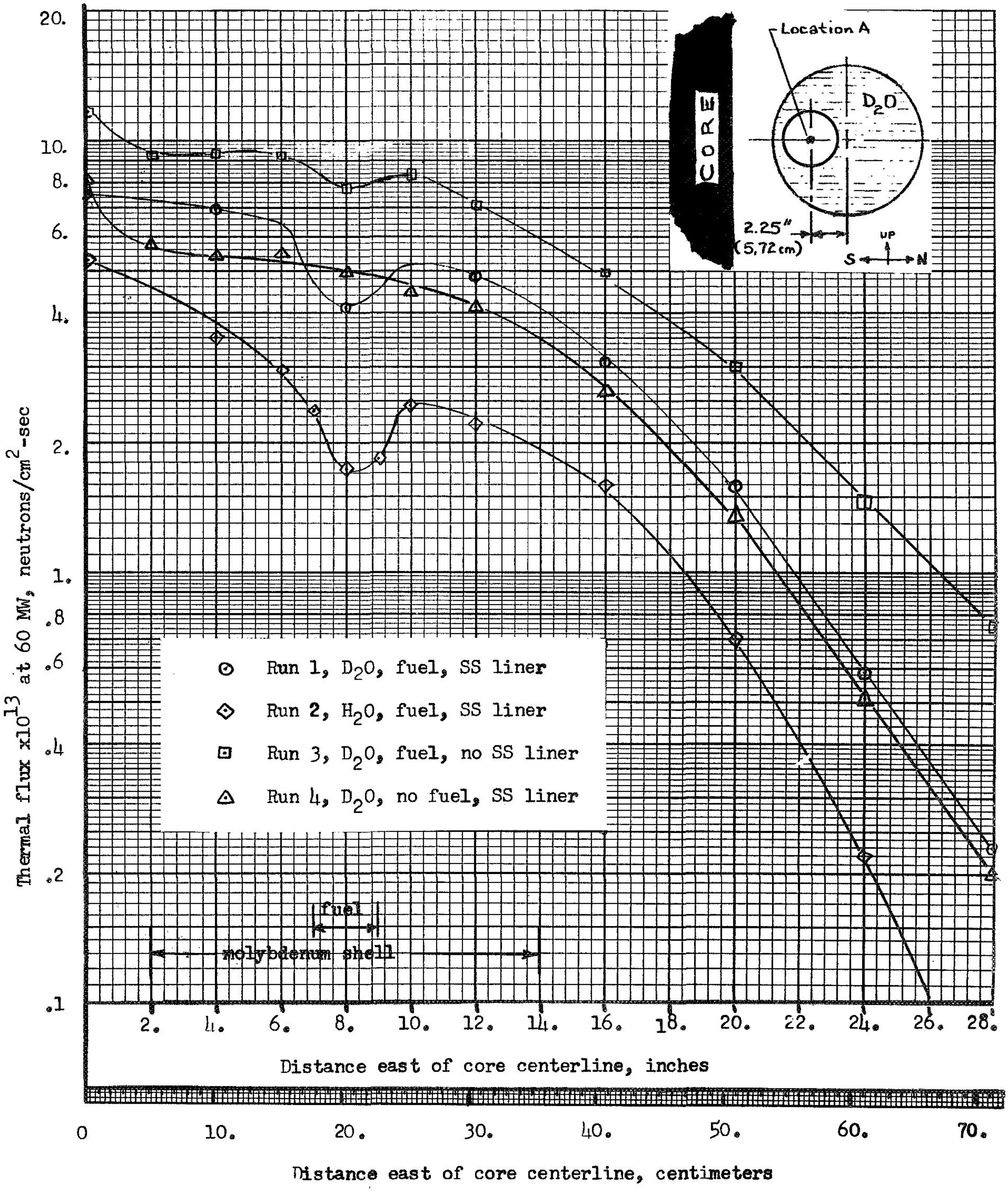



Figure 11.- Thermal neutron flux distribution as a function of distance east of the core centerline, as measured on the experiment capsule axis, 2.25 inches (5.72 centimeters ) south of the HT-2 axis. ( **Location A** )

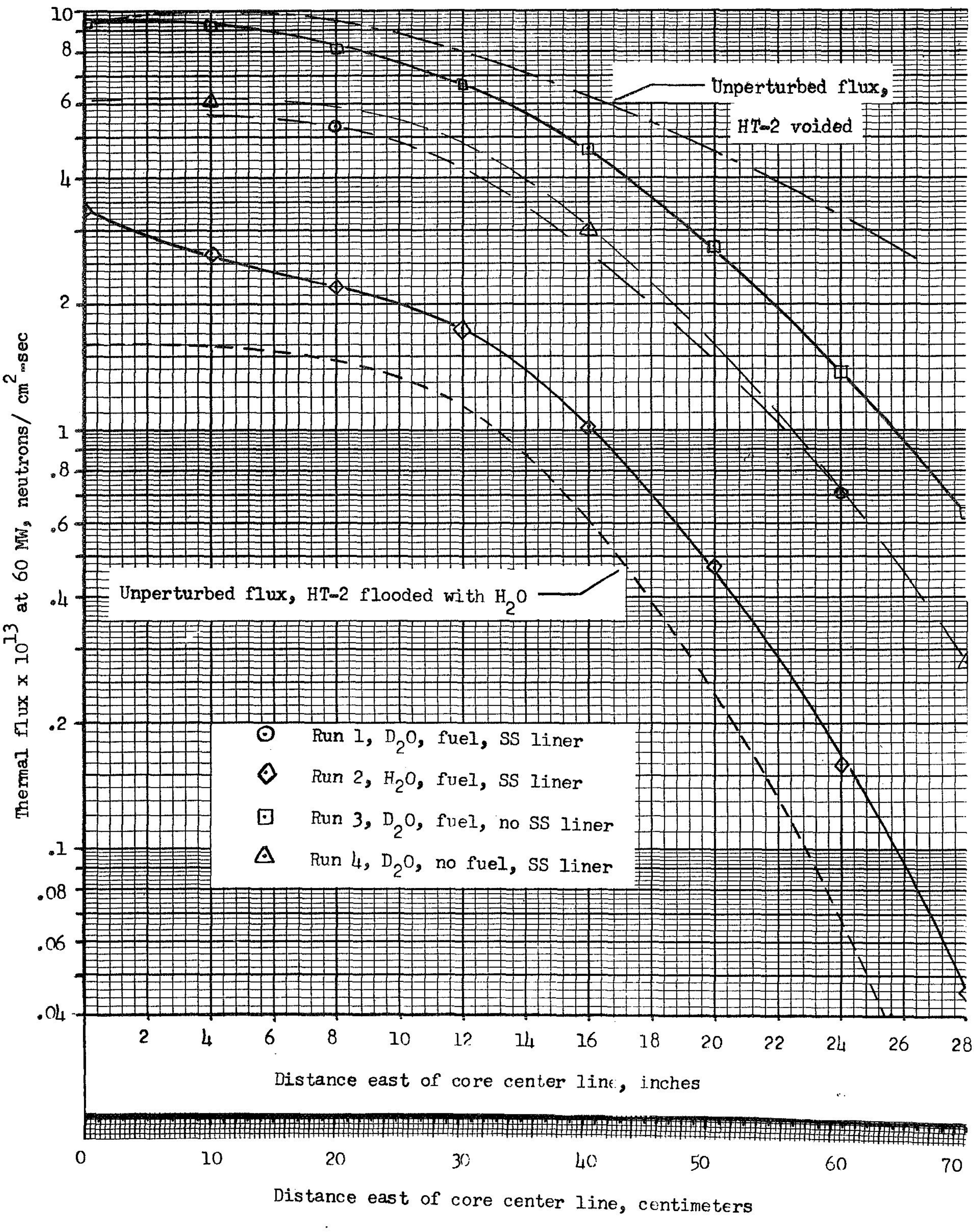



Figure 12.- Thermal neutron flux distribution as a function of the distance east of the core centerline, as measured on the HT-2 axis. ( **Location J** )

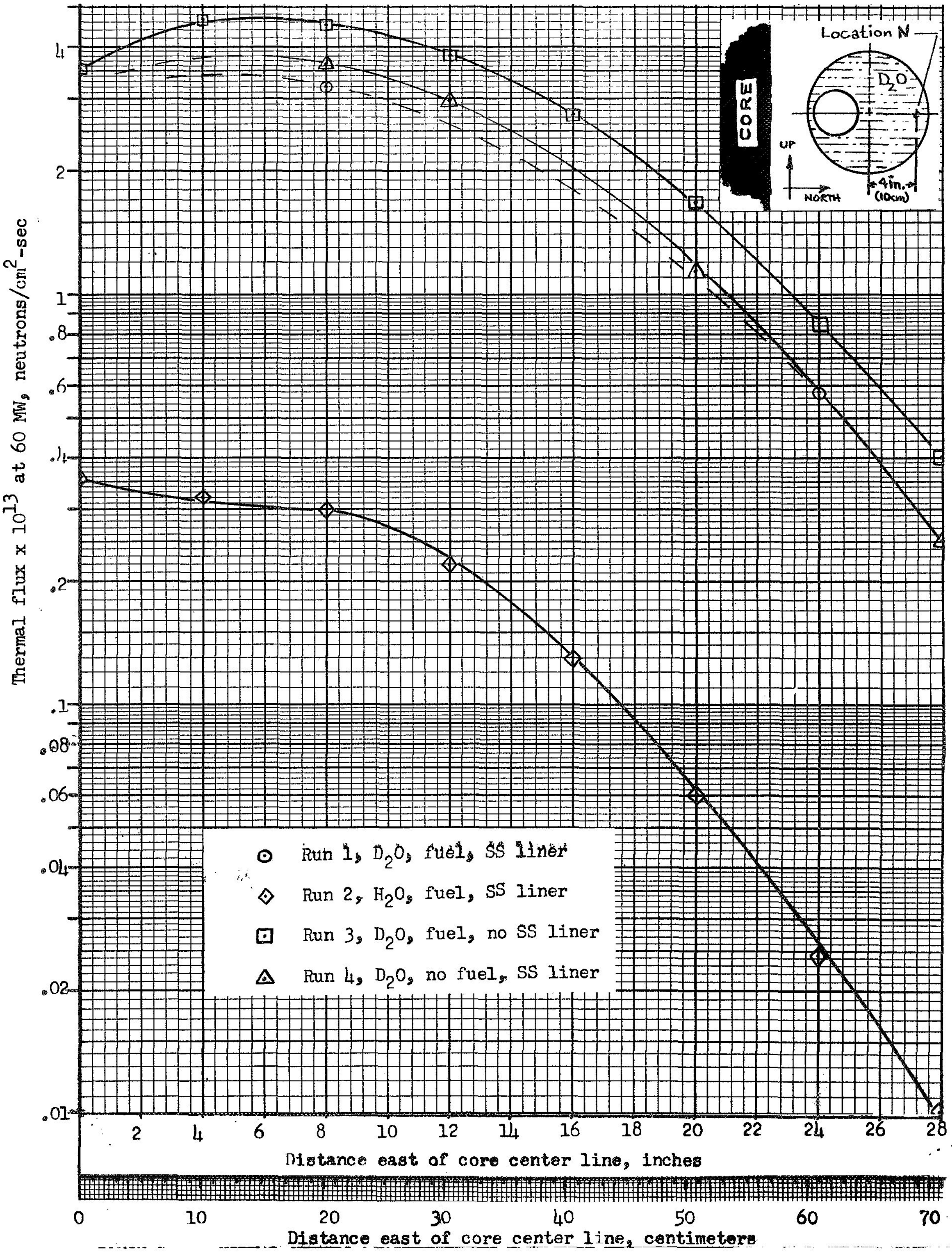



Figure 13.- Thermal neutron flux distribution as a function of distance east of the core center line, as measured 4 inches ( 10 centimeters) north of the HT-2 axis. ( **Location N** )

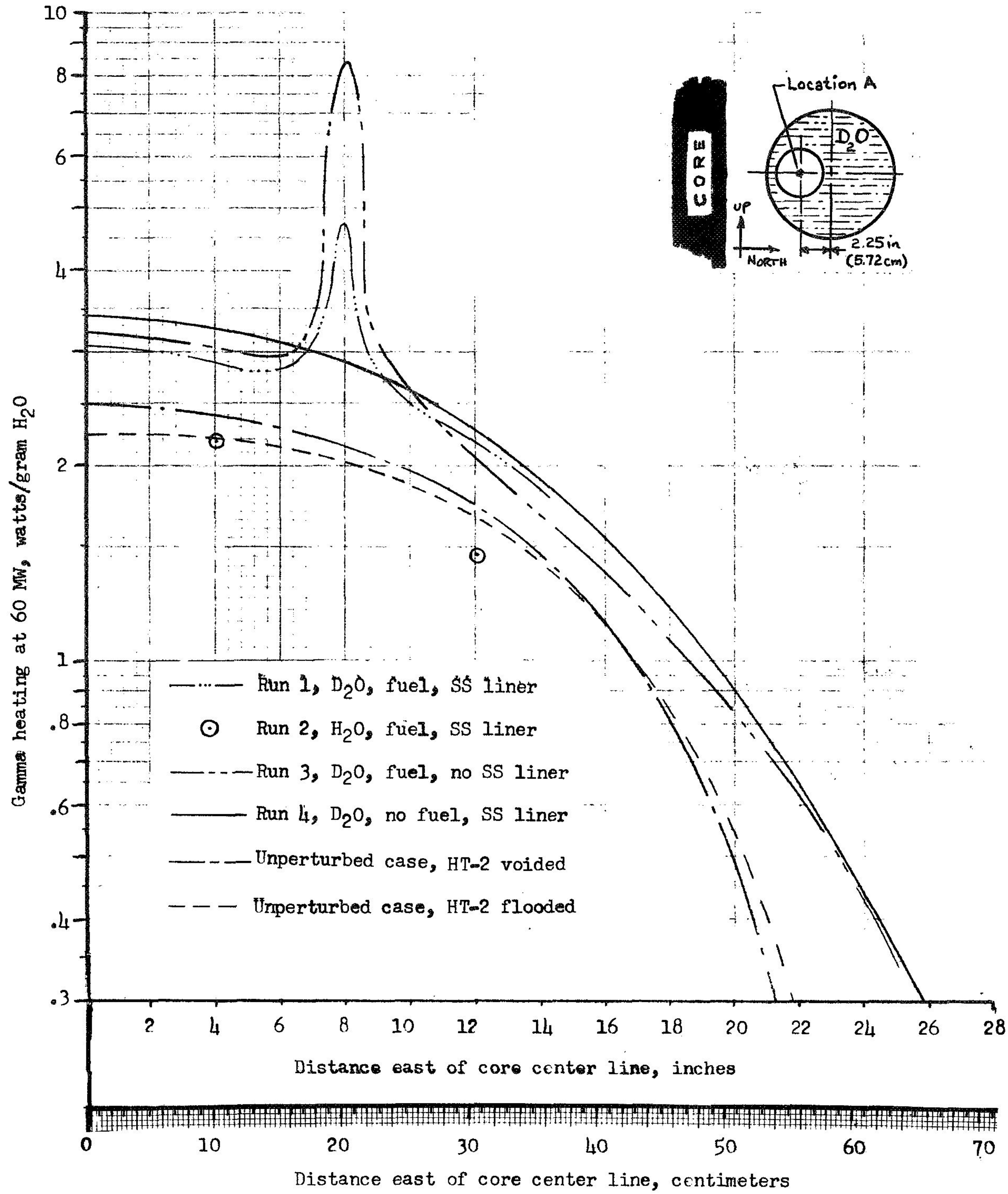



Figure 14.- Gamma heating distribution as a function of distance east of the core centerline, as measured on the experiment capsule axis, 2.25 inches ( 5.72 centimeters ) south of the HT-2 axis. ( **Location A** )