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IRRADIATION TESTS OF URANIUM METAL TUBES FOR D_2O - COOLED REACTORS

COMPILED BY

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DU PONT

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IRRADIATION TESTS OF URANIUM METAL TUBES FOR D_2O-COOLED REACTORS

Compiled by

Stephen R. Nemeth

Approved by

W. B. DeLong, Director
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February 1966

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ABSTRACT

This report summarizes the dimensional changes in experimental uranium metal tubular fuel elements exposed in the Heavy Water Components Test Reactor (HWCTR). The irradiation tests simulated conditions in a pressurized heavy-water-moderated-and-cooled power reactor. The Zircaloy-clad coextruded fuel elements were approximately 2 inches OD and up to 10 feet long. Tests were conducted at metal temperatures up to 500°C, reactor pressures up to 1200 psi, and exposures up to 6800 MWD/Te U. The uranium metal fuel tubes that contained dilute (<1000 ppm) Fe, Si, and Al additions performed better (<3% volume change) than the U-1.5 wt % Mo, the U-2 wt % Zr alloys, or the unalloyed uranium tubes (6-10% volume change). Irradiation results for Zr-9.3% enriched uranium HWCTR driver fuel tubes and Th-1.4% enriched uranium fuel tubes are also presented. The performance of both of these fuels was satisfactory; volume changes of 5.0% at 1.83 atom % burnup for the Zr-9.3% U alloy and <1% at 3500 MWD/Te Th-U for the Th-1.4% U alloy were observed.

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INTRODUCTION

For several years the Savannah River Laboratory has been engaged in the development of Zircaloy-clad, uranium metal fuel tubes for use in heavy-water-moderated-and-cooled power reactors.

The fuel tubes for this development were fabricated by Nuclear Metals, West Concord, Mass. The processes for manufacture of the coextruded metal tubes discussed in this report are described in DP-943 and DP-976.

Early irradiations of metal fuel tubes were performed in the Savannah River Plant reactors, the Vallecitos Boiling Water Reactor, and the NRU reactor at Chalk River, prior to the startup of the Heavy Water Components Test Reactor (HWCTR) at Savannah River.

The majority of the irradiation tests were conducted in the HWCTR, which is a heavy-water-moderated-and-cooled reactor with a coolant temperature of 250°C at 1200 psi pressure. Thirty-seven of the experimental uranium-metal fuel tubes were irradiated in the HWCTR. In addition, two Zircaloy-clad, Th-1.4% ^{235}U tubes were irradiated to investigate the performance of breeder reactor fuels. Concurrently, two sets (24 tubes per set) of Zr-9.3% ^{235}U alloy tubular fuel elements were irradiated as driver fuel for the HWCTR.

This report presents the irradiation conditions and dimensional changes of the test and driver elements irradiated in the power reactor program between 1957 and 1964.

SUMMARY

Before October 1962, when operation of the Heavy Water Components Test Reactor (HWCTR) started, 21 full-length Zircaloy-clad metal fuel tubes were irradiated in special tests in the SRP production reactors because they were the only facilities available for irradiating full-length fuel elements. In these special tests, the Zircaloy-clad test fuels experienced exposures and specific powers of interest for D_2O-moderated-and-cooled power reactors. Maximum exposures for some of these test elements exceeded 5000 MWD/Te U. However, the coolant temperature and maximum surface and metal temperatures of these test assemblies were as much as 200°C lower than required for the power reactors. These irradiation tests indicated the superior performance of unalloyed uranium over the U-2 wt % Zr alloy from the standpoint of volume change (Table I).

Irradiation tests in the Vallecitos Boiling Water Reactor and the NRU reactor at Chalk River confirmed that unalloyed uranium was superior to the U-2% Zr alloy at core temperatures of approximately 400°C.

During operation of the HWCTR, a total of 22 long (120 inches) uranium metal fuel tubes and 15 short (about 12 inches) fuel tubes were irradiated. The test elements were irradiated in several exposure increments at relatively high fission rates and temperatures for exposure up to 6800 MWD/Te U at time-averaged central metal temperatures up to 500°C.

Irradiation of the uranium metal fuel tubes established the superior performance of the dilute alloys containing Fe, Si, and Al additions (up to 1000 ppm each) compared with unalloyed uranium, uranium-2 wt % zirconium and uranium-1.5 wt % molybdenum. Volume changes of less than 3% occurred in the dilute alloy tubes, compared with volume changes of 6 to 10% in the other alloys.

The beneficial effects of external restraint were demonstrated by contrasting the fuel performance in the HWCTR with previous performance of similar fuels under low-restraint conditions. External restraint provided by thick cladding and high reactor coolant pressure restricted the volume growth of the uranium fuel cores during irradiation.

In connection with studies of heavy-water-moderated-and-cooled Th-^{233}U alloy-fueled breeder reactors, two Zircaloy-clad Th-1.4 wt % ^{235}U tubular elements were irradiated in HWCTR. An exposure of 3500 MWD/Te Th-U was reached with satisfactory performance before operation of the HWCTR was terminated.

The HWCTR driver assemblies, of which two full complements of tubes (24 per set) were irradiated, performed exceptionally well. Volume increases of about 5% were experienced in tubes irradiated to 1.83 atom % burnup. The fuel tubes were composed of coextruded Zircaloy-clad, zirconium-9.3 wt % enriched uranium alloy cores. None of the components of any of the assemblies showed any damage due to irradiation or mechanical causes.

DISCUSSION

A. METAL FUEL TUBE IRRADIATIONS IN SRP REACTORS [1]

Prior to operation of the Heavy Water Components Test Reactor (HWCTR), a number of Zircaloy-clad metal fuel tubes (listed in Table I) were irradiated in special tests in the SRP reactors because they were the only facilities available for irradiating full-length fuel elements. In these special tests the exposures and specific powers of these tubes were in the range that was of interest for D_2O-cooled power reactors. However, the coolant temperature and maximum surface and metal temperatures of the test fuel tubes were as much as 200°C lower than the corresponding design temperatures for power reactor assemblies. Maximum exposures of greater than 5000 MWD/Te U were reached for some of the test elements.[2] These tests indicated the superior performance of unalloyed uranium over U-2 wt % Zr from the standpoint of volume change (Table I).

Details of the irradiation test conditions and results are presented in the classified reports listed under the reference section of this report.

B. URANIUM-2 WT % ZIRCONIUM FUEL TUBE (VBWR IRRADIATION) [3]

A coextruded 2.06-inch-diameter tubular fuel element of uranium-2 wt % zirconium alloy clad with Zircaloy-2 was irradiated in the Vallecitos Boiling Water Reactor (VBWR) to 1410 MWD/Te U burnup with a maximum core temperature of 433°C. The purpose of this irradiation was to determine the dimensional stability of the alloy under conditions approaching those contemplated for D_2O-cooled power reactors (see Table II).

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(1) References 1, 2, 3, 4, and 5.
Reference 6 - DP-232, 245, 285, 295, 315, 345, 375, 405, 415, 425, 435, 445, 455, 475, 545, 565, 575, 585, 595, 615, 625, 645, 655.

(2) Reference 6 - DP-245, p 38.

(3) Reference 6 - DP-345, 395, 435, 475, 935.
Reference 8.

Interim inspection of the fuel tube showed that the amount of swelling increased from 0.8 volume % at 770 MWD/Te to 3.6 volume % at 1410 MWD/Te (0.7% final cladding strain). This test indicated that the dimensional stability of the U-2 wt % Zr alloy was inferior to that of unalloyed uranium under similar conditions.

C. UNALLOYED URANIUM FUEL TUBE (NRU IRRADIATION)[4]

A full-length (132 inches), 2.06-inch-OD x 1.467-inch-ID, unalloyed uranium fuel tube with a coextruded Zircaloy-clad thickness of 0.030 inch was irradiated to an exposure of 1060 MWD/Te U at a maximum core temperature of 400°C in the E-20 loop of the NRU reactor at Chalk River. The purpose of the irradiation was to obtain data on the stability of such fuel at temperature and pressure conditions approximating those expected in a D_2O-cooled power reactor (see Table III).

Postirradiation inspection showed a maximum volume change of 1.9% and a cladding strain of 0.19%. There was no indication of damage to the fuel tube because of irradiation or mechanical causes. Test results were sufficiently promising to warrant continuation of the development of uranium metal fuels for use in D_2O-cooled power reactors.

D. URANIUM-1.5 WT % MOLYBDENUM ALLOY FUEL TUBE ENRICHED TO 3% ^{235}U (DESIGNATED 3EMT-2)[5]

To characterize the irradiation behavior of a solid-solution alloy fuel tube, the 3EMT-2 fuel assembly was designed and fabricated using a fuel tube with a U-1½ wt % Mo alloy core (U enriched to 3% ^{235}U). An accumulated exposure of 4970 MWD/Te U at a core temperature of 500°C was reached at the time operation of HWCTR was terminated.

The calculated operating characteristics for the 3EMT-2 assembly are listed in Table IV. A 540°C HWCTR driver core temperature limit was assumed using a Zircaloy outer housing. Early exposure (May 11, 1964 through July 13, 1964) utilized a stainless steel outer housing because of a 600°C driver temperature limit, with hydraulic figures shown in Table IV differing by about 5%. Exposure with the Zircaloy housing extended from July 26, 1964 through December 1, 1964.

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(4) Reference 6 - DP-285, 315, 345, 495, 505, 515.
Reference 7.

(5) Reference 6 - DP-915, 924, 945.

The 3EMT-2 assembly consisted of a Zircaloy-clad fuel tube 2.072-inch-OD x 1.695-inch-ID x 45.4 inches long. Cladding thickness was 0.025 inch of nickel-free Zircaloy-2 used with zirconium end plugs. A stainless steel or a Zircaloy-2 housing (depending on the driver temperature limits) with stainless steel top and bottom fittings and a 60-inch inner housing of stainless steel was used. See Figures 1 and 2 for details of the assembly. The inner housing defines an annular flow channel for the inner surface coolant. A 14-inch stainless steel tube below the fuel tube positioned the core at the axial level of maximum neutron flux.

Outer housing, OD, inches	- 2.564 (Zr)
	- 2.625 (SS)
Outer housing, ID, inches	- 2.494 (Zr)
	- 2.498 (SS)
Outer housing, rib circle, inches	- 2.100
Inner housing, rib circle, inches	- 1.670
Inner housing, OD, inches	- 1.020
Inner housing, ID, inch	- 0.660

Nominal diametral clearances between the fuel tube and the ribs were 40 mils on the OD and 30 mils on the ID. Ribs in the stainless steel outer housing were formed in an L shape from 60-mil-thick stainless steel sheet. The base of the L was welded to the housing through holes in the wall; the rib height was 0.20 inch. Zircaloy housing ribs were nominally 0.20 inch high and 60 mils thick; they were welded into grooves milled in the OD of the inner housing.

The 3EMT-2 assembly fuel tube exhibited a maximum volume increase of 5.8% after 4970 MWD/Te U exposure at a maximum central metal temperature of 485°C. This is in contrast to the 10% volume increase for unalloyed uranium fuel, but greater than the 2-3% for the short metal fuel tubes (SMT) under analogous conditions. The ID contracted 0.017 inch while the OD increased 0.007 inch. See Figure 3 for detailed postirradiation measurements. None of the components of this assembly showed signs of damage due to irradiation (other than swelling) or from mechanical causes.

E. UNALLOYED NATURAL URANIUM TUBES WITH RESTRAINING CLADDING (DESIGNATED RMT)[6]

The RMT assembly shown in Figure 4 consisted of a Zircaloy-clad fuel tube 2.070-inch-OD x 1.568-inch-ID x 118 inches long. The core was of unalloyed uranium with a nickel-free Zircaloy-2 cladding thickness of 0.060 inch on both inner and outer surfaces. A Zircaloy outer housing (2.494-inch ID) had four full-length, internal ribs welded to it by electron beam welding. Nominal rib circle was 2.110 inches diameter to provide a 0.040-inch diametral clearance for the fuel tube. The inner housing was stainless steel with four full-length ribs (0.062 inch wide) welded to it for a rib circle of 1.520 inches in diameter (0.050-inch diametral clearance for the fuel tube).

This fuel assembly was designed to determine the effect of cladding restraint on the swelling of an unalloyed uranium core. Restraint, in addition to that caused by coolant pressure, was offered by the strength of 0.060-inch-thick Zircaloy cladding. Metal test element previously irradiated had cladding 0.020 inch to 0.025 inch thick. Operating characteristics for the RMT-1-2 assembly in HWCTR are presented in Table V. The RMT assembly reached 3320 MWD/Te U irradiation exposure at termination of the HWCTR operation.

The OD and ID changes (+0.007 inch and -0.002 inch, respectively) in the RMT fuel tube were negligibly small, with a calculated maximum volume change of +2.6% (10% for 0.020-inch clad tubes at about the same exposure). See Figure 5 for detailed postirradiation measurements. None of the components of this assembly showed signs of damage from irradiation or mechanical causes.

F. THIN-WALLED OUTER TUBE OF ENRICHED URANIUM METAL (DESIGNATED ETWO)[7]

The enriched thin-walled outer (ETWO) assemblies consisted of a Zircaloy-clad fuel tube of 2.06-inch-OD x 1.70-inch-ID x 118 inches long. The core was unalloyed uranium enriched to 2.1 wt % ^{235}U with a nickel-free Zircaloy cladding of 0.025-inch thickness. A four-ribbed, Zircaloy outer housing and a four-ribbed, stainless steel inner housing defined the flow annuli for this type of assembly (Figure 6). The maximum exposures for the two assemblies irradiated

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(6) Reference 6 - DP-885, 895 915, 925, 945.
(7) Reference 6 - DP-855, 865, 875, 885, 895, 935, 945.

in HWCTR were 6830 MWD/Te U for 2ETWO-2 and 6470 MWD/Te U for 2ETWO-3 at time-averaged maximum central metal temperatures of 512 and 537°C, respectively. Calculated operating characteristics for the ETWO assemblies are shown in Table VI.

These tests were terminated when an interim inspection revealed that severe growth had occurred. Large variations in the outside diameter from the unirradiated condition were measured; these revealed that (1) significant inward growth had occurred, and that (2) the inner housing ribs had restricted this growth and caused a lobated pattern in the fuel pieces. The inner housings could not be withdrawn from the fuel pieces after application of up to 4500 pounds longitudinal force in one case. Although neither fuel piece had failed, further irradiation was cancelled because of severe growth and lack of knowledge of the internal dimensions of the fuel.

At exposures below 5000 MWD/Te, the outside diameters of the fuel tubes underwent little change near the ends of the tubes; above this exposure the outer surfaces became irregular. Because the inner housings were stuck, it is apparent that the inside diameters decreased more than 30 mils. Also, because there is a direct relation between dimensional change and location of the ribs on the inner housings (as shown in Figures 8, 9, 10, and 11), there is indication that the fuel tubes might have shown more instability had they not been supported by the housings.

Longitudinal profiles of the outside diameters of the two fuel tubes are shown in Figure 7; the values for exposure are shown at the top of each curve sheet. At exposures up to about 5000 MWD/Te, little change is noted in diameters; above this exposure the outer surfaces of both tubes became very irregular.

G. SEGMENTED TUBES OF ALLOYED URANIUM METAL (DESIGNATED SMT)[8]

Short-length, Zircaloy-clad, tubular fuel elements with brazed end seals were designed and fabricated to provide a means for irradiating a wide range of uranium alloy compositions and/or heat treatments in a minimum number of reactor positions. The segmented metal tube (SMT) assemblies were designed to accommodate up to ten fuel pieces for a maximum fuel column length of 113 inches (Figure 12).

(8) Reference 6 - DP-855, 865, 875, 885, 895, 915, 925, 945.

Fuel elements, 1.69-inch-OD x 1.23-inch-ID x 11.25 inches long, were spaced inside the Zircaloy housing tube by ribbed washers located at each fuel element junction.

Two SMT assemblies were irradiated in HWCTR: (1) SMT-1-2, which contained ten fuel elements composed of the U-330 ppm Fe-900 ppm Al alloy, was irradiated to 4325 MWD/Te and (2) SMT-1-3, which contained five fuel elements with U-350 ppm Fe-300 ppm Si-800 ppm Al alloy, was irradiated to 5600 MWD/Te. The goal exposure of 10,000 MWD/Te was not reached because of termination of HWCTR operation.

Calculated operating characteristics for the SMT assemblies in HWCTR are presented in Table VII. The fuel elements swelled very little during irradiation - a maximum of 2.6% at 4300 MWD/Te for the U-Fe-Al tubes and 3.3% at 5600 MWD/Te for the U-Fe-Si-Al tubes. In both cases, changes in OD and ID were less than 0.010 inch. Volume change of the U-Fe-Al tubes was only 2.5% on exposure to 3300 MWD/Te, shown in Figure 13, but thereafter the tube swelled very little, shown in Figure 14.

Figure 15 shows the results of the first two interim inspections of the U-Fe-Si-Al tubes of assembly SMT-1-3, with Figure 16 showing the final results. These tubes resisted swelling more effectively than the U-Fe-Al tubes with only 2% volume change at 4500 MWD/Te.

Irradiation of the SMT assemblies established the superior performance of dilute uranium alloys compared with unalloyed uranium, which exhibited up to 10% volume increase for similar exposures under the external restraint imposed by 1200 psi reactor coolant pressure. Both SMT assemblies were suitable for further irradiation.

H. THORIUM-1.4 WT % URANIUM METAL TUBES (DESIGNATED TMT)[9]

In connection with studies of D_2O-moderated-and-cooled thorium breeder reactors, two pile-worthy Zircaloy-clad tubular elements of a Th-1.4 wt % ^{235}U alloy were produced. The fuel elements were designed to operate at an exposure-averaged maximum core temperature of about 500°C to a maximum exposure of 20,000 MWD/Te. Both of the elements were irradiated in HWCTR to an exposure of 3500 MWD/Te with satisfactory performance. Irradiation of the two TMT assemblies was interrupted by HWCTR termination.

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(9) Reference 6 - DP-925, 935, 945.
References 9 and 10.

The fuel tubes were coextruded having cores of thorium metal alloy with 93% enriched uranium to provide a core enrichment of 1.4% ^{235}U, and Zircaloy-2 cladding 0.030-inch thick. Fuel tubes were 2.540-inch-OD x 1.830-inch-ID x 118 inches long, with a core length of 108 inches. The TMT fuel assemblies (Figure 17) consisted of ribbed Zircaloy-2 inner and outer housing tubes with an outer housing ID of 2.900 inches and an inner housing OD of 1.420 inches.

Calculated operating characteristics for the TMT assemblies in HWCTR are presented in Table VIII. The maximum specific power of 62 MWD/Te Th-U alloy was based on a maximum thermal flux of 1.5×10^{14} n/(cm^2)(sec), which was the estimated flux available in the outer test ring of the HWCTR with the moderator at 200°C.

The thorium fuel power was expected to vary more with exposure and irradiation history than any element previously irradiated in the HWCTR. Two effects cause this; one is a continuing long-term effect of burnup, and the second is a short-term effect due to isotope decay during interim shutdown periods.

At low exposure, the power of the element is governed by the ^{235}U enrichment. With long exposure, the build-in of ^{233}U becomes more important. After about 10,000 MWD/Te Th-U, an equilibrium ^{233}U level is reached so that power is then largely independent of the initial ^{235}U content. At this time, maximum specific power at constant flux would have decreased to about 50 MWD/Te Th-U and maximum core temperature would be down from 500 to about 440°C. To compensate partially for this loss, the assembly was to be moved to an inner test ring position to take advantage of an approximately 10% greater flux for a similar increase in power.

Complicating this long-term effect, however, is the effect of increased ^{233}U build-in during periods of shutdown after significant exposures. The ^{233}U concentration can be significantly higher at the end of a shutdown period than at the beginning. Thus, power in the element can be higher in startup following an outage than prior to the outage. Following the re-startup, ^{233}U concentration and power will decay somewhat but a new higher power would result. Therefore, depending on its irradiation history, it was expected that during the latter part of its exposure, the power of a TMT element might have been greater than its earlier power and might limit the reactor power by minimum BOSF. To minimize this effect, the TMT fuel tubes were to be transferred to stainless steel housings in order to operate them at this attenuated thermal flux and reduced element power for the latter portion of their irradiation exposure.

Irradiation of the two TMT assemblies extended from August 25, 1964 through December 1, 1964, at which time the HWCTR operation was terminated. Following irradiation of the two test assemblies in HWCTR, they were removed and postirradiation measurements were made on one fuel tube from Assembly TMT-1-2. Comparison of the pre- and post-irradiation data (Figure 18) indicates that the volume change in the region of maximum exposure and core temperature was about 0.8% after an accumulated exposure of 3500 MWD/Te. The volume change resulted from an increase of about 0.005 inch in the OD and an increase of about 0.001 inch in the ID of the tube. No signs of damage due to irradiation or from mechanical causes were found on any of the components of these assemblies. It is assumed that the TMT assemblies would have reached the goal exposure of 20,000 MWD/Te if HWCTR operation had not been terminated.

I. THIN-WALLED NESTED TUBES (DESIGNATED TWNT)[10]

Of the fifteen fabricated pairs of inner and outer fuel tubes having unalloyed uranium cores, seven pairs and one outer tube were irradiated in HWCTR. The primary objectives of irradiation testing of these tubes were (1) to determine whether unalloyed, natural uranium metal fuel elements would operate to the exposures required for economic operation of heavy-water-moderated power reactors, and (2) to relate core volume growth and cladding strain to exposure for elements that have a maximum core temperature of about 400°C. These irradiation tests were terminated following failure of two of the assemblies as a result of wear by the twisted ribbon spacers through the cladding of the fuel tubes. Exposures of up to only 1085 MWD/Te U for the nested assemblies and 1195 MWD/Te U for the outer tube assembly (TWO-1-2) were attained at the termination of the irradiation test.

The fuel tubes were coextruded, having natural uranium cores 0.130 inch thick and low-nickel Zircaloy-2 cladding 0.025 inch thick. Over-all length was 118 inches, with the following nominal diameters:

	Outer Tube, inches	Inner Tube, inches
Clad, OD	2.060	1.020
Core, OD	2.010	0.970
Core, ID	1.750	0.710
Clad, ID	1.700	0.660

(10) Reference 6 - DP-795, 805, 815, 835, 845.

Figure 19 shows the TWNT assembly, and Figure 20 shows the individual fuel tubes. Spacing between fuel tubes was maintained by a twisted Zircaloy ribbon wrapped helically around the inner tube with a 2-foot pitch. A similar ribbon was wrapped around the outer tube with an opposite helix to provide spacing between the fuel and the outer Zircaloy housing tube. Each ribbon was attached to the end seal of the fuel tube. In the case of the TWO-1-2 assembly, an inner housing tube with a helically wound twisted Zircaloy ribbon was used in place of the inner fuel tube.

The calculated operating characteristics of the TWNT assemblies in the HWCTR are presented in Table IX; data for the TWO-1-2 assembly are presented in Table X. It was planned to operate the TWNT assemblies up to exposures of 6600 MWD/Te U and the TWO-1-2 assembly to 13,000 MWD/Te U.

Irradiation of six TWNT assemblies and the TWO-1-2 assembly was started in HWCTR on October 5, 1962. A failure occurred in one assembly (TWNT-7) on November 27, 1962. A second failure (TWNT-14) occurred on December 26, 1963, at which time the remaining assemblies were discharged from the reactor. Although preirradiation, out-of-pile flow tests indicated no flow damage, subsequent flow tests resulted in two failures because of wear by the twisted ribbon spacers through the cladding on the inside of the outer fuel tubes (Figure 21).

Postirradiation inspection of the six sound assemblies gave further evidence that the most probable cause of the two failures in the HWCTR was fretting corrosion, which was caused by rubbing of the twisted-ribbon spacers on the cladding surfaces of the fuel tubes. Results of the inspection are tabulated on page 12.

Assembly Number	HWCTR Conditions[a] AC Flow, days	DC Flow, days	Vibration Damage[b] Housing Tube	Outer Fuel Tube Twisted Ribbon	Outer Surface	Inner Surface	Inner Fuel Tube Twisted Ribbon	Outer Surface
TWNT-5	58	39	Minor	Minor	Minor	Moderate	Moderate	Moderate
TWNT-7	44	24	-----Failed Assembly---------Not Inspected--------					
TWNT-9	58	39	Minor	Minor	Minor	Moderate	Moderate	Severe
TWNT-11	51	29	--	Nil	Nil	Minor	Minor	Minor
TWNT-12	58	39	Minor	Minor	Minor	Severe	Severe	Severe
TWNT-13	14	15	Minor	Nil	Minor	Minor	Nil	Minor
TWNT-14	58	39	-----Failed Assembly---------Not Inspected--------					
TWO-1-2	58	39	Minor	Nil	Minor	Minor	Nil[c]	Minor[c]

As a consequence of the two failures, all subsequent metal tube irradiations in the Du Pont power program were made with spacing provided by ribbed housings or ribbed washers.

Detailed postirradiation measurements were not made on the components of the six sound assemblies of this group of tubes because of their relatively low exposures, under 1200 MWD/Te U.

(a) AC Flow - 100 gpm per assembly @ 10 ft/sec velocity.
DC Flow - 30 gpm per assembly @ 3 ft/sec velocity.

(b) Vibration Damage:
- Severe - Grooves in fuel tubes 0.003 inch to 0.008 inch deep; ribbon wear 1/32 to 3/32 inch.
- Moderate - Grooves in fuel tubes up to 0.003 inch deep; ribbon wear up to 1/32 inch.
- Minor - Localized removal of oxide film on fuel tubes; no measureable ribbon wear.

(c) Inner Housing Tube - The TWO-1-2 assembly did not contain an inner fuel tube.

J. M-1 DRIVER TUBES NO. 22 AND NO. 48[11]

The initial set of HWCTR driver assemblies, designated M(etal)-1, had fuel tubes 2.300-inch-OD x 1.960-inch-ID x 118 inches long, made of Zr-9.3% U clad with Zircaloy (Figure 22). The uranium was enriched to 93% ^{235}U. The flow annuli for these assemblies were defined by an outer housing of Zircaloy and an inner housing of Zircaloy which supported stainless steel-boron leaves that served as a burnable poison. The assemblies operated from reactor startup on October 5, 1962 until November 24, 1963, at which time their reactivity became too low to maintain desired temperatures in the test assemblies, and they were replaced by a second set of metal drivers.

Tubes No. 22 and No. 48 were selected for inspection because their exposures and operating temperatures were the highest of the 24 fuel tubes used. Calculated operating characteristics of the driver assemblies are presented in Table XI. Inspections of both of these tubes at interim exposures prior to their final discharge indicated a uniform rate of volume change with increased burnup.

The irradiation of these tubes to maximum fission burnups of 1.83 atom % for No. 22 and 1.74 atom % for No. 48 produced maximum volume changes over short sections near the point of maximum burnup of 4.9 and 5.1%, respectively. The maximum calculated cladding strains were 0.22 and 0.18%. No damage due to irradiation or vibration was found.

Plots of the OD, ID, volume change, and exposure are shown in Figures 23 and 24. The outside diameter of tube No. 22 decreased 0.007 inch over a section 18 to 32 inches from the top, and increased about 0.005 inch over a section 60 to 78 inches from the top. The inside diameter decreased an average of about 0.008 inch and the length increased 0.38 inch. The maximum volume increase of 3.4% occurred about 75 inches from the top. Fuel tube No. 22 was inspected and measured on three occasions when maximum burnup and time-averaged maximum central metal temperatures were: 1.15 atom % - 500°C; 1.54 atom % - 504°C; and 1.83 atom % - 498°C.

Final inspection of tube No. 48 was made at a burnup of 1.74 atom % and a time-averaged maximum central metal temperature of 485.4°C; an interim inspection was made at 1.46 atom % and 485.7°C. Changes in the outside diameter of the fuel tube varied from a 0.009 inch decrease over a section 30 to 40 inches from the top to a 0.002 inch increase over a section 66 to 76 inches from the top.

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(11) Reference 6 - DP-905, 915, 935.
Reference 13.

The inside diameter decreased an average of about 0.011 inch, and the length increased 0.44 inch. The maximum volume increase of 3.2% occurred about 75 inches from the top.

The outside diameter curves shown in Figures 22 and 23 are the average of continuous longitudinal scans taken every 22½ degrees around each of the tubes. The diameters on each tube coincided within about 2 mils over the entire tube length, indicating that there was no appreciable ovality. The maximum volume increases were much less than were predicted for fuel of this composition irradiated under the stated conditions of burnup and temperatures; however, the volume changes shown were calculated on the basis of inside and outside diameter changes only. If the increases in length of the tubes were taken into account in the calculations, volume changes could be about 0.5% higher than shown.

None of the components of either assembly showed any signs of damage due to irradiation or from mechanical causes.

K. M-2 DRIVER TUBES NO. 1 AND NO. 18[12]

The second set of metal driver assemblies, designated M(etal)-2, were identical to the M-1 drivers. Fuel tubes were 2.300-inch-OD x 1.960-inch-ID x 118 inches long, made of Zr-9.3% U clad with Zircaloy. The uranium was enriched to 93% ^{235}U. Flow annuli for these assemblies were defined by an outer housing of Zircaloy and an inner housing of Zircaloy which supported stainless steel-boron leaves that served as a burnable poison. The inner housings containing the target components were replaced with bare Zircaloy inner housings late in the irradiation cycle (October 1964) to extend the reactivity lifetime of the M-2 driver lattice. Irradiation of these assemblies extended from December 29, 1963 through December 1, 1964, at which time nuclear operation of the HWCTR was terminated.

Tubes No. 1 and No. 18 were selected for periodic inspection because their operating temperatures were the highest of the 24 fuel tubes in the M-2 set. Both of these tubes were inspected at interim exposures (Figures 25 and 26), which indicated a relatively uniform rate of volume change with increased burnup.

(12) Reference 6 - DP-905, 925.

Irradiation of these tubes to maximum fission burnups of 1.50 atom % for No. 1 and 1.54 atom % for No. 18 produced maximum volume changes over short sections near the points of maximum burnup of only 4.3 and 4.1%, respectively (Figures 27 and 28 . The maximum cladding strains were 0.17 and 0.27%. No damage due to irradiation or vibration was found.

REFERENCES

A. CLASSIFIED REPORTS

1. W. H. Gleaves. Irradiation of Natural Uranium Tubes Clad with Zirconium. USAEC Report DP-244, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1957)(Secret).

2. R. B. Olcott, et al. Progress Report on Irradiation of Metallic Natural Uranium Fuel for Power Reactors. USAEC Report DP-340, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1960)(Secret).

3. R. B. Olcott, et al. Irradiation of Zr-Clad Uranium Metal Tubes for Power Reactors. USAEC Report DP-404, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1959) (Secret).

4. W. H. Gleaves, et al. Irradiation of Uranium-2% Zirconium Slugs at Elevated Temperatures. USAEC Report DP-433, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1959) (Secret).

5. W. R. McDonell and G. R. Caskey. Irradiation Tests of Power Reactor Fuel Elements - Progress Report. USAEC Report DP-476, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1960)(Secret).

Additionally, the following Progress Reports are pertinent.

DP-496	DP-616	DP-736
DP-516	DP-636	DP-756
DP-536	DP-656	DP-776
DP-556	DP-676	DP-796
DP-576	DP-696	DP-816
DP-596	DP-716	

B. UNCLASSIFIED REPORTS

6. D. F. Babcock. Power Reactor Studies - Quarterly Progress Report. USAEC Report DP-232, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1957).

Additionally, the following Progress Reports are pertinent.

DP-245	DP-425	DP-535	DP-645	DP-755	DP-865
DP-265	DP-435	DP-545	DP-655	DP-765	DP-875
DP-285	DP-445	DP-555	DP-665	DP-775	DP-885
DP-295	DP-455	DP-565	DP-675	DP-785	DP-895
DP-315	DP-465	DP-575	DP-685	DP-795	DP-905
DP-345	DP-475	DP-585	DP-695	DP-805	DP-915
DP-375	DP-485	DP-595	DP-705	DP-815	DP-925
DP-385	DP-495	DP-605	DP-715	DP-825	DP-935
DP-395	DP-505	DP-615	DP-725	DP-835	DP-945
DP-405	DP-515	DP-625	DP-735	DP-845	DP-965
DP-415	DP-525	DP-635	DP-745	DP-855	

7. L. M. Arnett, et al. Final Hazards Evaluation of the Heavy Water Components Test Reactor (HWCTR). USAEC Report DP-600, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1962).

8. R. C. Pugh. Irradiation of a Uranium Metal Fuel Tube at Chalk River. USAEC Report DP-667, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1961).

9. H. C. Quigley. Irradiation of a U-2 wt % Zr Fuel Tube in the VBWR. USAEC Report DP-709, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1962).

10. S. R. Nemeth. Thorium-1.4 wt % ^{235}U Metal Fuel Tubes - Fabrication and Irradiation in HWCTR. USAEC Report DP-943, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1965).

11. S. R. Nemeth. Metal Fuel Tube Manufacture. USAEC Report DP-976, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1965).

12. C. L. Angerman and W. R. McDonell. Irradiation Behavior of Uranium Metal Tubes in a Pressurized Heavy Water Reactor. USAEC Report DP- , E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (to be issued).

13. C. L. Angerman and G. R. Caskey. Irradiation Behavior of Zr Driver Tubes for HWCTR. USAEC Report DP-994, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1965).

TABLE I

SUMMARY OF POWER PROGRAM METAL FUEL TUBE IRRADIATION RESULTS

NM No.	Irrad. Assy. No.	Core Composition	Core Condition	Cladding Material	Dimensions, inches OD	ID	Core Length	Core Thk.	Clad. Thk.
9131[a]	TFEN-11	U	Beta treated air cooled	Zr	2.950	2.440	150	0.195	0.030
9132[a]	TFEN-12	U	Beta treated air cooled	Zr	2.950	2.440	150	0.195	0.030
6930[a]	TFEN-14	U	Beta treated air cooled	Zr	2.950	2.440	150	0.195	0.030
6988[a]	TFEN-64	U	Beta treated air cooled	Zr	2.950	2.440	150	0.195	0.030
11	SPR-6	U-2 wt % Zr	Gamma treated[b]	Zr-2	2.065	1.470	117	0.268	0.015
15	SPR-2	U-2 wt % Zr	Diffusion treated[c]	Zr-2	2.064	1.468	120	0.268	0.015
19	SPR-3	U-2 wt % Zr	Diffusion treated[c]	Zr-2	2.063	1.468	119	0.268	0.015
22	SPR-8	U-2 wt % Zr	Diffusion treated[c]	Zr-2	2.067	1.468	116	0.270	0.015
28	SPR-7	U-2 wt % Zr	Spheroidization treated[d]	Zr-2	2.064	1.467	116	0.268	0.015
29	SPR-4	U-2 wt % Zr	Diffusion treated[c]	Zr-2	2.064	1.468	116	0.268	0.015
30	SPR-5	U-2 wt % Zr	Diffusion treated[c]	Zr-2	2.069	1.468	116	0.271	0.015
34	SPR-9	U-2 wt % Zr	As extruded	Zr-2	2.060	1.467	34	0.267	0.015
41[e]	-	U-2 wt % Zr	Diffusion treated[c]	Zr-2	2.066	1.467	34	0.270	0.015
53	SPR-11	U	Beta treated air cooled	Zr-2	2.057	1.469	108	0.234	0.030
59	SPRD-12	Zr-9.3 wt % U (93% ^{235}U)	As extruded	Zr-2	2.300	1.961	112	0.137	0.015
60	SPRD-13	Zr-9.3 wt % U (93% ^{235}U)	As extruded	Zr-2	2.298	1.961	112	0.136	0.015
72[f]	-	U	Beta treated air cooled	Zr-2	2.060	1.467	107	0.237	0.030
75	TWT-2	U	Beta treated air cooled	Zr-4[g]	2.065	1.694	104	0.135	0.025
82[h]	3 EMT-2	U-1.5 wt % Mo (3% ^{235}U)	Beta treated air cooled	Zr-4[g]	2.072	1.695	42	0.138	0.025
103	TWIT-3	U	Beta treated air cooled	Zr-4[g]	1.023	0.654	106	0.135	0.025
104	TWIT-3	U	Beta treated air cooled	Zr-4[g]	1.023	0.654	107	0.135	0.025
108	TWT-3	U	Beta treated air cooled	Zr-4[g]	2.059	1.695	110	0.132	0.025
117	TWIT-2	U	Beta treated air cooled	Zr-4[g]	1.024	0.653	109	0.135	0.025
119	TWIT-2	U	Beta treated air cooled	Zr-4[g]	1.022	0.656	110	0.133	0.025
133[h]	RMT-1-2	U	Beta treated air cooled	Ni free Zr-2	2.070	1.568	108	0.191	0.060
148[h]	2 ETWO-2	U (2.1% ^{235}U)	Beta treated oil quenched	Ni free Zr-2	2.057	1.699	110	0.129	0.025

(a) Experimental metal tubes fabricated prior to Du Pont power program.
(b) Gamma treated - heated 800°C for 1/2 hour, "air cooled."
(c) Diffusion treated - heated 880°C for 7 hours, "air cooled."
(d) Spheroidization treated - heated 680°C for 72 hours, following diffusion treatment.
(e) Irradiated at VBWR.
(f) Irradiated at NRU E-20 loop.
(g) Zr-4 currently designated as low nickel Zr-2.

TABLE I (Cont'd)

SUMMARY OF POWER PROGRAM METAL FUEL TUBE IRRADIATION RESULTS

Irradiation Conditions				Postirradiation Examination					
Exposure, MWD/tonne U Max	Temperature, °C Sheath Max	Interface Max	Center Max	Dimension Change, inch Max ΔOD	Max ΔID	ΔL	Core ΔV/V Max Percent	Max OD Clad Strain (Calculated)	Comments
*				+0.010	-	0	-	+0.4	
*				+0.009	-	0	-	+0.3	
*				+0.005	-	0	-	+0.2	
*				+0.008	0.000	-	+2.1	+0.3	
*				+0.021	-0.005	-	+5.3	+1.0	
*				+0.006	-0.004	-	+2.0	+0.3	
*				+0.024	-0.014	+0.4	+7.4	+1.2	
*				+0.030	-0.010	-	+8.1	+1.5	Failed
*				+0.010	-	-	-	+0.5	Stuck inner housing
*				+0.020	-0.012	-	+6.2	+1.6	Failed
*				+0.028	-0.010	+0.4	+7.6	+1.4	
*				-	-	-	-	-	Failed
1410	300	340	433	+0.014	-0.004	+0.035	+3.6	+0.7	
*				+0.008	-	-	-	+0.4	Stuck inner housing
*				-0.004	-0.004	+0.44	+0.2	-	
*				+0.010	-0.013	+0.31	+7.3	-	
1060	250	335	400	+0.004	-0.006	-	+1.9	+0.19	
*				+0.011	+0.005	+0.13	+1.0	+0.2	
4970	274	421	485	+0.009	-0.017	-	+5.8	-	
*				+0.007	-	-0.13	+1.3	-	
*				+0.007	-	-0.19	+2.0	-	
*				+0.015	-0.009	-	+6.0	+0.5	
*				+0.006	-	-	-	+0.3	
*				+0.006	-	-	-	+0.3	
3320	297	460	485	+0.007	-0.002	-	+2.6	-	
6830	306	367	512	+0.045	>0.030	-	-	-	Stuck inner housing

(h) Irradiated at HWCTR.
(i) Reference 6 - DP-245, p 38.

* The metal tubes(*) irradiated in the SRP reactor tests reached exposures and specific powers for D_2O-cooled power reactors; however, coolant temperatures and maximum surface and metal temperatures were as much as 200°C lower than required for power reactors. Maximum exposures of greater than 5000 MWD/Te were reached for some of the test elements.(i)

TABLE I (Cont'd)

SUMMARY OF POWER PROGRAM METAL FUEL TUBE IRRADIATION RESULTS

NM No.	Irrad. Assy. No.	Core Composition	Core Condition	Cladding Material	Dimensions, inches OD	ID	Core Length	Core Thk.	Clad. Thk.
149 (h)	2 ETWO-3	U (2.1% ^{235}U)	Beta treated oil quenched	Ni free Zr-2	2.060	1.699	109	0.131	0.025
153 (h)	SMT-1-2 (j)	U-350 Fe-900 Al	Beta treated oil quenched	Zr-2	1.690	1.234	10.7	0.184	0.022
154 (h)	SMT-1-2 (j)	U-350 Fe-900 Al (gamma)	Beta treated oil quenched	Zr-2	1.690	1.234	10.7	0.184	0.022
158 (h)	SMT-1-3 (j)	U-350 Fe-900 Al-300 Si	Beta treated oil quenched	Zr-2	1.690	1.234	10.7	0.184	0.022
2-1 (h)	TMT-1-2	Th-1.5 wt % U (93% ^{235}U)	As extruded	Zr-2	2.539	1.840	108	0.289	0.030
2-3 (h)	TMT-1-3	Th-1.5 wt % U (93% ^{235}U)	As extruded	Zr-2	2.536	1.831	108	0.292	0.030
OT-1 (h)	TWNT-14	U	Beta treated air cooled	Zr-2	2.060	1.694	110	0.133	0.025
OT-3 (h)	TWNT-7	U	Beta treated air cooled	Zr-2	2.060	1.694	110	0.133	0.025
OT-5 (h)	TWNT-13	U	Beta treated air cooled	Zr-2	2.060	1.694	110	0.133	0.025
OT-6 (h)	TWNT-5	U	Beta treated air cooled	Zr-2	2.060	1.694	110	0.133	0.025
OT-7 (h)	TWNT-9	U	Beta treated air cooled	Zr-2	2.060	1.694	110	0.133	0.025
OT-9 (h)	TWO-1-2	U	Beta treated air cooled	Zr-2	2.060	1.694	110	0.133	0.025
OT-11 (h)	TWNT-11	U	Beta treated air cooled	Zr-2	2.060	1.694	110	0.133	0.025
OT-12 (h)	TWNT-12	U	Beta treated air cooled	Zr-2	2.060	1.694	110	0.133	0.025
IT-3 (h)	TWNT-14	U	Beta treated air cooled	Zr-2	1.022	0.656	109	0.133	0.025
IT-5 (h)	TWNT-7	U	Beta treated air cooled	Zr-2	1.022	0.656	109	0.133	0.025
IT-7 (h)	TWNT-12	U	Beta treated air cooled	Zr-2	1.022	0.656	109	0.133	0.025
IT-9 (h)	TWNT-9	U	Beta treated air cooled	Zr-2	1.022	0.656	109	0.133	0.025
IT-11 (h)	TWNT-5	U	Beta treated air cooled	Zr-2	1.022	0.656	109	0.133	0.025
IT-12 (h)	TWNT-11	U	Beta treated air cooled	Zr-2	1.022	0.656	109	0.133	0.025
IT-13 (h)	TWNT-13	U	Beta treated air cooled	Zr-2	1.022	0.656	109	0.133	0.025
No. 22 (h)	M-1 (k) Driver-22	Zr-9.3 wt % U (93% ^{235}U)	As extruded	Zr-2	2.300	1.961	112	0.137	0.015
No. 48 (h)	M-1 (k) Driver-48	Zr-9.3 wt % U (93% ^{235}U)	As extruded	Zr-4 (g)	2.300	1.961	112	0.137	0.015
No. 1 (h)	M-2 (k) Driver-1	Zr-9.3 wt % U (93% ^{235}U)	As extruded	Zr-2	2.300	1.961	112	0.137	0.015
No. 18 (h)	M-2 (k) Driver-18	Zr-9.3 wt % U (93% ^{235}U)	As extruded	Zr-2	2.300	1.961	112	0.137	0.015

(g) Zr-4 currently designated as low nickel Zr-2.
(h) Irradiated at HWCTR.
(j) The two SMT assemblies containing five short elements (11 1/4") made from each of the long extruded tubes 153, 154, and 158; SMT-1-2, 5 each from tubes 153 and 154; SMT-1-3, 5 from tube 158 and 5 dummies.

TABLE I (Cont'd)

SUMMARY OF POWER PROGRAM METAL FUEL TUBE IRRADIATION RESULTS

Irradiation Conditions				Postirradiation Examination					
Exposure, MWD/tonne U Max	Temperature, °C Sheath Max	Interface Max	Center Max	Dimension Change, inch Max ΔOD	Max ΔID	ΔL	Core ΔV/V Max Percent	Max OD Clad Strain (Calculated)	Comments
6470	306	367	537	+0.040	>0.030	-	-	-	Stuck inner housing
4325	306	385	391	+0.004	-0.010	-	+2.6	-	
4325	306	385	391	+0.004	-0.010	-	+2.6	-	
5600	306	385	424	+0.006	-0.010	-	+3.3	-	
3560	266	406	505	+0.005	+0.001	-	+0.8	-	
3470	266	406	495		No postirradiation measurement				
1010	302	353	375		No postirradiation measurement				Failed
700	301	353	375		No postirradiation measurement				Failed
385	306	363	389	-0.001	-	-0.13	-	-	
960	295	339	365	+0.004	-	-0.22	-	-	
1005	306	361	385	+0.004	-	-0.11	-	-	
1195	306	364	394	-0.004	-	-0.19	-	-	
890	299	346	375	+0.004	-	-0.05	-	-	
1085	306	359	386	+0.004	-	-0.13	-	-	
1010	308	-	369		No postirradiation measurement				Failed
700	308	-	375		No postirradiation measurement				Failed
1085	308	-	386	+0.004	-	-0.08	-	-	
1005	308	-	368	+0.004	-	-0.11	-	-	
960	308	-	365	+0.004	-	-0.06	-	-	
890	308	-	375	+0.004	-	-0.09	-	-	
385	308	-	384	-0.001	-	-0.11	-	-	
1.83 atom %	-	-	540	-0.007	-0.015	+0.38	+4.9	+0.22	
1.74 atom %	-	-	540	-0.009	-0.013	+0.44	+5.1	+0.18	
1.50 atom %	-	-	570	+0.005	-0.014	-	+4.3	+0.17	
1.54 atom %	-	-	585	+0.007	-0.009	-	+4.1	+0.27	

(k) The four driver tubes listed are representative of the 48 drivers irradiated in the HWCTR and were selected because their exposures and operating temperatures were considered the highest of the two sets.

TABLE II

Calculated Operating Characteristics of the U-2 wt % Zr Test Assembly Irradiated in VBWR

	Maximum	Average
Output of element, kw	250	200
Average specific power, MW/Te U	20	16.1
Peak-to-average flux	1.16	1.16
Maximum heat flux, pcu/(ft^2)(hr)		
Inner surface	292,000	233,000
Outer surface	254,000	202,000
Mean coolant temperature, $^\circ$C	285	285
Maximum surface temperature, $^\circ$C	300	299
Maximum cladding-core interface temperature, $^\circ$C	345	335
Maximum core temperature, $^\circ$C	433	405
Reactor pressure, psia	1000	1000
Exit steam quality, %		
Inner surface	4	3
Outer surface	2	1.5
Test channel inlet subcooling, Btu/lb	10	10

<u>TABLE III</u>

Calculated Operating Characteristics of the
<u>Unalloyed Uranium Fuel Tube Irradiated in NRU E-20 Loop</u>

Reactor power, MW	200
Total heat output of loop, kw	935
Heat output of fuel, kw	885
Maximum specific power, MW/Te U	25.2
Maximum heat flux, pcu/(ft^2)(hr)	275,000
Maximum-to-average flux ratio	1.265
Coolant flow, gpm	202
Loop pressure, psig	1090
Coolant inlet temperature, $^\circ$C	177
Coolant outlet temperature, $^\circ$C	196
Maximum surface temperature, $^\circ$C	250
Maximum core-cladding interface temperature, $^\circ$C	335
Maximum core temperature, $^\circ$C	400

TABLE IV

Calculated Operating Characteristics of 3EMT-2 Assembly in HWCTR

Zircaloy Outer Housing; 540°C Driver Limit

Reactor pressure, psig	1200
Moderator temperature, °C	200
Reactor inlet temperature, °C	187
Reactor power, MW	36
Flow to 3EMT assembly, gpm	200
Assembly power (flow x ΔT), MW	0.77
Maximum/average power, axial	1.2
Nominal maximum specific power[(a)], MW/Te U	116
Nominal maximum core temperature, °C	500
Equivalent core length, ft	3.12
Actual exposure, MWD/Te U	4970

	Outer Annulus	Inner Annulus
Nominal max heat flux, pcu/(hr)(ft^2)	570,000	575,000
Coolant flow, gpm	94.6	90.0
Coolant velocity, ft/sec	20.4	21.0
Coolant ΔT, °C	17.2	14.8
Saturation temperature at outlet, °C	296	296
Nominal outlet subcooling, °C	93	95
Min outlet subcooling[(b)] (with HSF), °C	85	90
Nominal max cladding surface temp, °C	273	274
Nominal max fuel-cladding interface temp, °C	419	421
Flow area, $in.^2$	1.504	1.376
Surface heat transfer area, ft^2/ft	0.540	0.445

(a) Includes moderator heating.

(b) Includes allowances for deviations from nominal geometry.

TABLE V

Calculated Operating Characteristics of RMT-1-2 Assembly in HWCTR

Reactor pressure, psig	1200
Moderator temperature, °C	200
Coolant inlet temperature, °C	183
Reactor power, MW	47
Flow to RMT-1 assembly[(a)], gpm	57
Assembly power, MW	0.64
Maximum/average power, axial	1.7
Nominal maximum specific power[(b)], MW/Te U	52
Nominal maximum core temperature, °C	495
Actual exposure, MWD/Te U	3320

	Outer Surface	Inner Surface
Nominal max heat flux, pcu/(hr)(ft^2)	251,000	277,000
Coolant flow[(a)], gpm	29.1	22.9
Coolant speed, ft/sec	6.4	6.8
Coolant ΔT, °C	46	46[(c)]
Saturation temperature at outlet, °C	296	296
Nominal outlet subcooling, °C	67	67
Min outlet subcooling[(d)] (with HSF), °C	57	58
Nominal max cladding surface temp, °C	297	296
Nominal max fuel-cladding interface temp, °C	451	460

(a) Purge flow of 5 gpm through inner housing, bypasses temperature monitor.

(b) Total fission power, includes moderator heating.

(c) An estimated 4.5% of power to inner channel is transferred to the 5-gpm axial stream. Together with a small amount of gamma heating in the axial stream, the resultant ΔT is approximately 11°C.

(d) Includes allowances for deviations from nominal geometry.

TABLE VI

Calculated Operating Characteristics of ETWO Assembly in HWCTR

Reactor pressure, psig	1200
Moderator temperature, °C	250
Coolant inlet temperature, °C	235
Head available for fuel assemblies, ft	71
Flow to 2ETWO assembly, gpm	300
Assembly power, MW	1.1
Maximum/average power, axial	1.7
Maximum specific power, MW/Te U	78
Maximum core temperature, °C	468
Actual exposure, MWD/Te U	6830

	Outer Surface or Channel	Inner Surface or Channel
Max nominal heat flux, pcu/(hr)(ft^2)	419,000	441,000
Flow, gpm	204	89
Velocity, ft/sec	21.4	20.3
Channel temperature rise, °C	11.0	22.0
Saturation temperature at outlet, °C	296	296
Nominal subcooling at outlet, °C	50	39
Min subcooling at outlet[a] (with HSF), °C	45	31
Max cladding surface temp, °C	302	306[b]
Max fuel-cladding interface temp, °C	363	367

(a) Includes allowances for deviations from nominal geometry.
(b) Local boiling.

TABLE VII

Calculated Operating Characteristics of SMT Assemblies in HWCTR

Reactor pressure, psig	1200
Moderator temperature, °C	250
Coolant inlet temperature, °C	235
Head available for fuel assemblies, ft	71
Flow to SMT assembly, gpm	150
Assembly power, MW	0.73
Maximum/average power, axial	1.7
Maximum specific power, MW/Te U	44
Maximum core temperature, °C	460
Actual exposure, MWD/Te U	5600

	Outer Surface	Inner Surface
Max heat flux, pcu/(hr)(ft^2)	343,000	368,000
Flow, gpm	93	57
Velocity, ft/sec	11.5	14.8
Channel temperature rise, °C	16.7	21.2
Saturation temperature at outlet, °C	296	296
Nominal subcooling at outlet, °C	44.3	40.2
Min subcooling at outlet[a] (with HSF), °C	43	38
Max cladding surface temperature[b], °C	306	306
Max fuel-cladding interface temperature, °C	385	385

(a) Includes allowances for deviations from nominal geometry.
(b) Local boiling.

TABLE VIII

Calculated Operating Characteristics of TMT Assemblies in HWCTR with the M-2 Drivers

Reactor pressure, psig	1200
Coolant inlet temperature, °C	185
Assembly coolant flow[a], gpm	150
Assembly coolant power, kw	1200
Maximum/average power, axial	1.8
Maximum nominal specific power[b], MW/Te Th-U	62
Maximum nominal core temperature, °C	500
Actual exposure, MWD/Te U	3500

	Outer Channel	Inner Channel
Max nominal heat flux, pcu/(hr)(ft^2)	400,000	455,000
Coolant flow, gpm	82.8	61.2
Coolant velocity, ft/sec	18.1	18.8
Coolant temperature rise, °C	31.1	33.2
Max (with HCF) effluent temperature, °C	234	234
Saturation temperature at effluent, °C	296	296
Min (with HCF) effluent temperature[c], °C	62	62
Max nominal cladding surface temp, °C	263	266
Max nominal fuel-cladding interface temp, °C	393	406

(a) Includes 6 gpm axial purge through inner housing.
(b) Includes moderator heating.
(c) Includes allowances for deviations from nominal geometry.

TABLE IX

Calculated Operating Characteristics of TWNT Assemblies in HWCTR

Reactor pressure, psig	1300
Moderator temperature, °C	250
Coolant inlet temperature, °C	230
Flow to TWNT assembly, gpm	100
Assembly power, MW	0.85
Maximum/average power, axial	1.7
Actual exposure, MWD/Te U	1085

	Outer Tube	Inner Tube
Maximum specific power, MW/Te U	43.0	39.7
Maximum core temperature, °C	403	396

	Outer Tube		Inner Tube	
	Outer Surface	Inner Surface	Outer Surface	Inner Surface
Max surface temp, °C	305	311[a]	305	308
Max heat flux, pcu/(hr)(ft²)	229,000	246,000	207,000	238,000
Flow, gpm	45.5	43.1	43.1	11.4
Velocity, ft/sec	9.45	9.64	9.64	10.7
Channel temperature rise, °C	26.5	37.3	37.3	35.2
Sat. temp at outlet, °C	300	300	300	300
Nom subcooling at outlet, °C	43	33	33	35
Min subcooling at outlet[b], (with HSF), °C	40	28	28	30

(a) Local boiling.
(b) Includes allowances for deviations from nominal geometry.

TABLE X

Calculated Operating Characteristics of TWO-1-2 Assembly in HWCTR

Reactor pressure, psig	1300
Moderator temperature, °C	250
Coolant inlet temperature, °C	230
Flow to TWO assembly, gpm	110
Assembly power, MW	0.69
Maximum/average power, axial	1.7
Maximum specific power, Te U	48.5
Maximum core temperature, °C	416
Actual exposure, MWD/Te U	1195

	Outer Surface	Inner Surface
Maximum heat flux, pcu/(hr)(ft^2)	258,000	277,000
Flow, gpm	50.5	49.5
Velocity, ft/sec	10.5	10.7
Channel temperature rise, °C	27.4	24.8
Saturation temperature at outlet, °C	300	300
Nominal subcooling at outlet, °C	42	45
Min subcooling at outlet[a] (with HSF), °C	40	43
Max cladding surface temperature, °C	306	311[b]
Max fuel-cladding interface temp, °C	372	382

(a) Includes allowances for deviations from nominal geometry.
(b) Local boiling.

TABLE XI

Calculated Operating Characteristics of HWCTR M-1 and M-2 Driver Tubes

Reactor pressure, psig	1200
Moderator temperature, °C	200
Coolant inlet temperature, °C	185
Reactor power, MW	42
Flow to driver assembly, gpm	300
Assembly power, MW	1.38
Coolant temperature rise, °C	17.4
Nominal max core temperature, °C	550
Nominal max heat flux, pcu/(hr)(ft^2)	500,000
Velocity, ft/sec	
Inner annulus	19
Outer annulus	23
Nominal max cladding surface temp, °C	320
Maximum core temperature, °C	550
Maximum actual exposure, atom % burnup	1.83

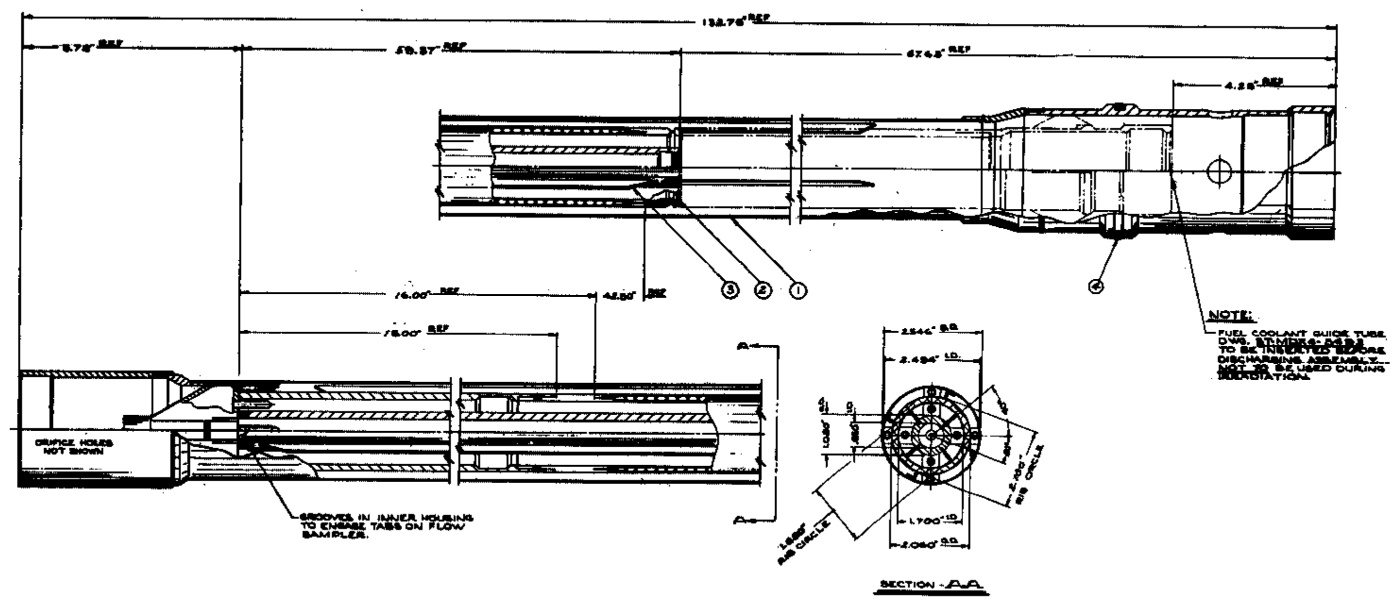


FIG. 1 3 EMT FUEL ASSEMBLY (STAINLESS STEEL HOUSING)

(1) 3 EMT Outer Housing Assembly
(2) 3 EMT Fuel Tube and Spacer
(3) 3 EMT Inner Housing Tube
(4) Piston Ring (HWCTR)

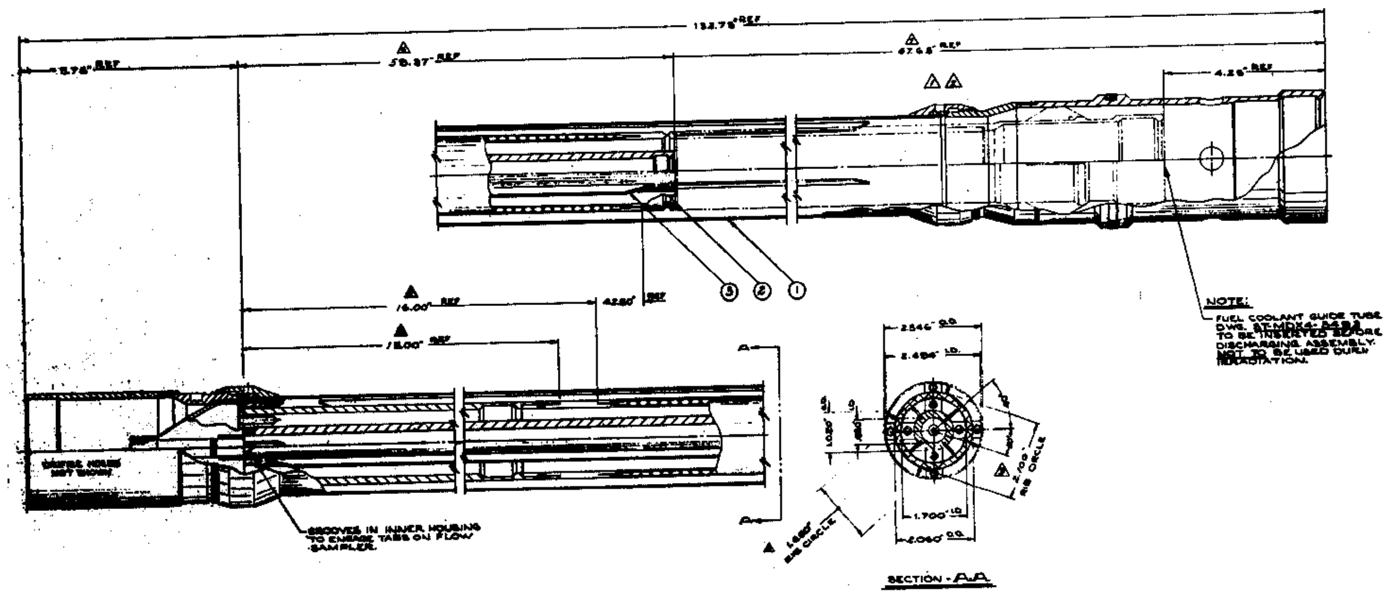


FIG. 2 3 EMT FUEL ASSEMBLY (ZIRCALOY HOUSING)

(1) 3 EMT Outer Housing Assembly
(2) 3 EMT Fuel Tube and Spacer
(3) 3 EMT Inner Housing Tube

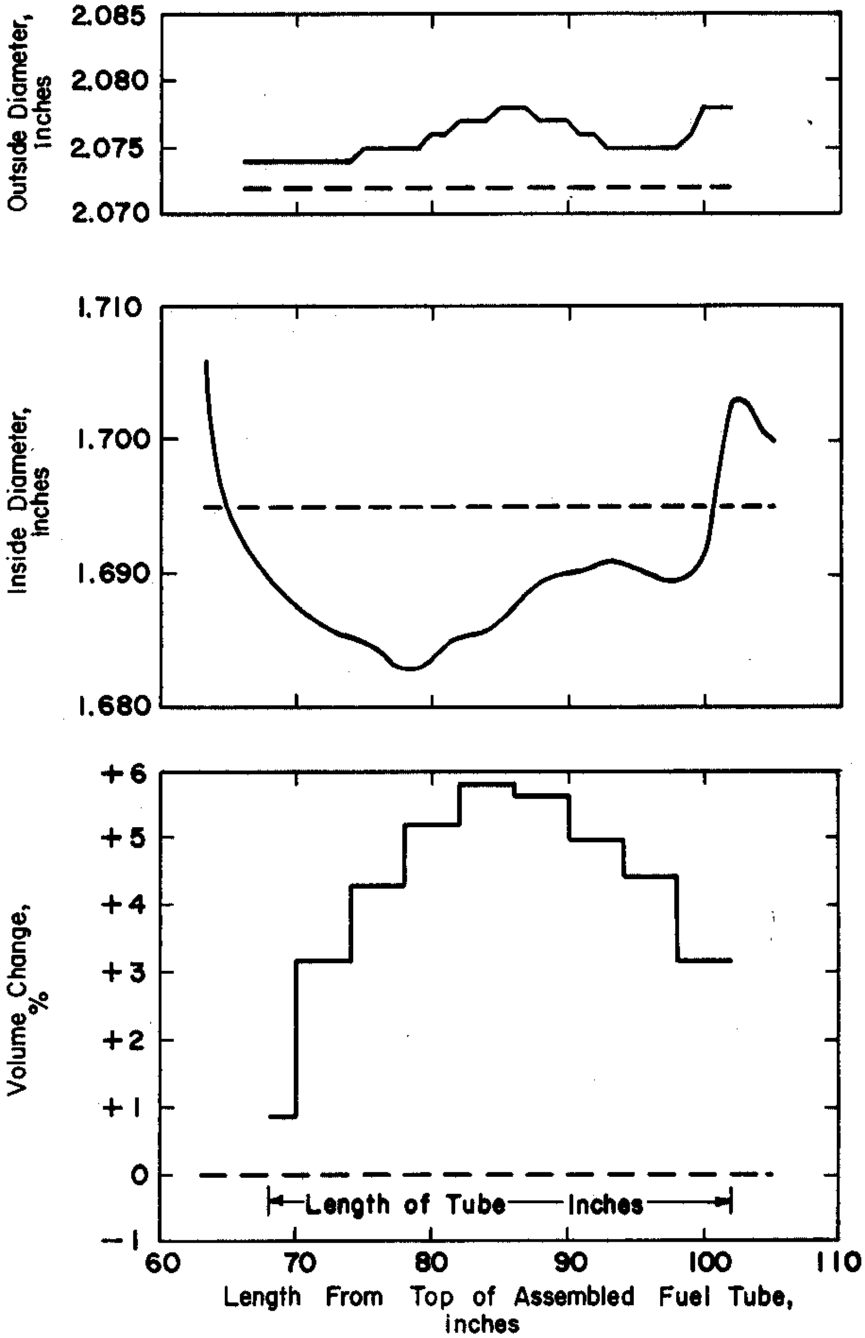


FIG. 3 HWCTR TEST ASSEMBLY 3EMT-2
Postirradiation Measurements

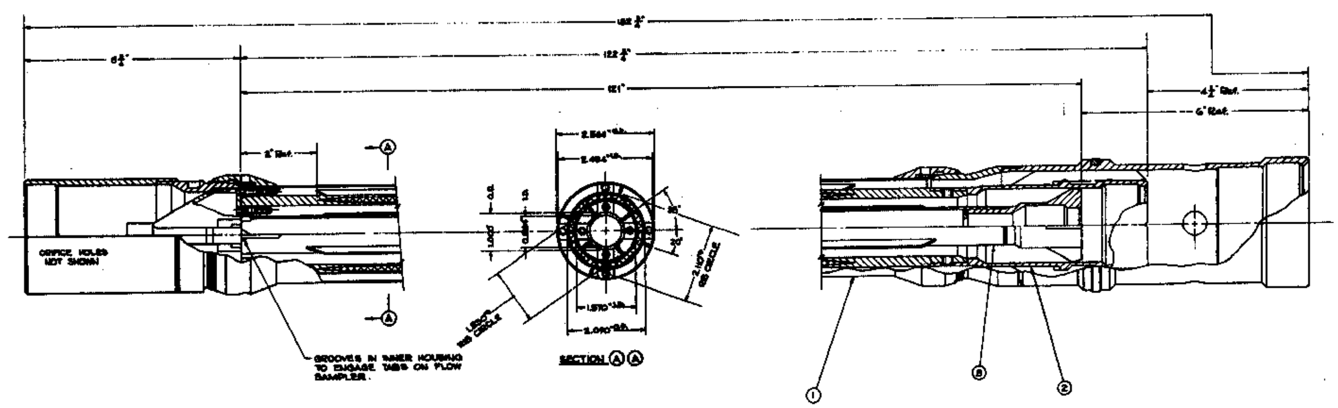


FIG. 4 RMT FUEL ASSEMBLY

(1) RMT Outer Housing Assembly
(2) RMT Fuel Tube Subassembly
(3) RMT Inner Housing Weldment

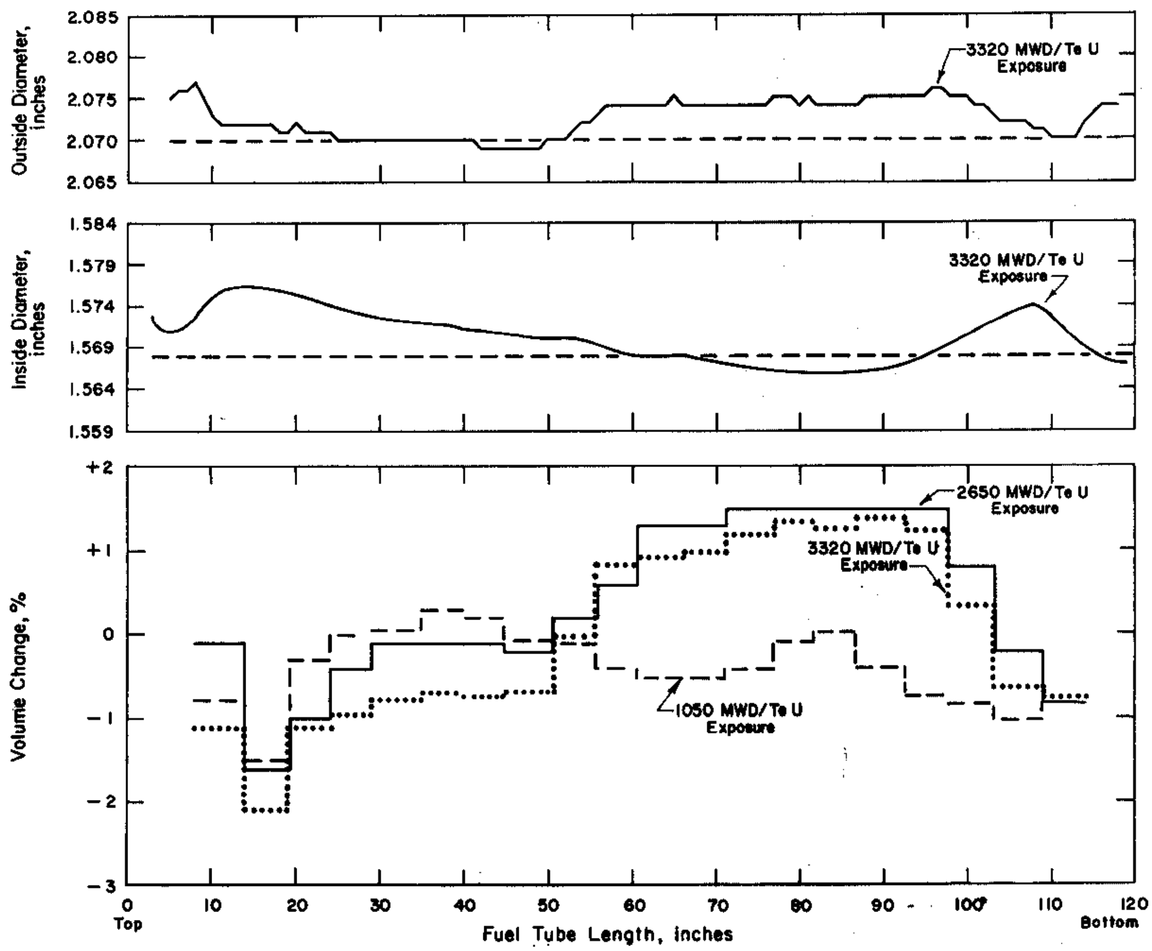


FIG. 5 HWCTR TEST ASSEMBLY RMT-1-2
POSTIRRADIATION MEASUREMENTS

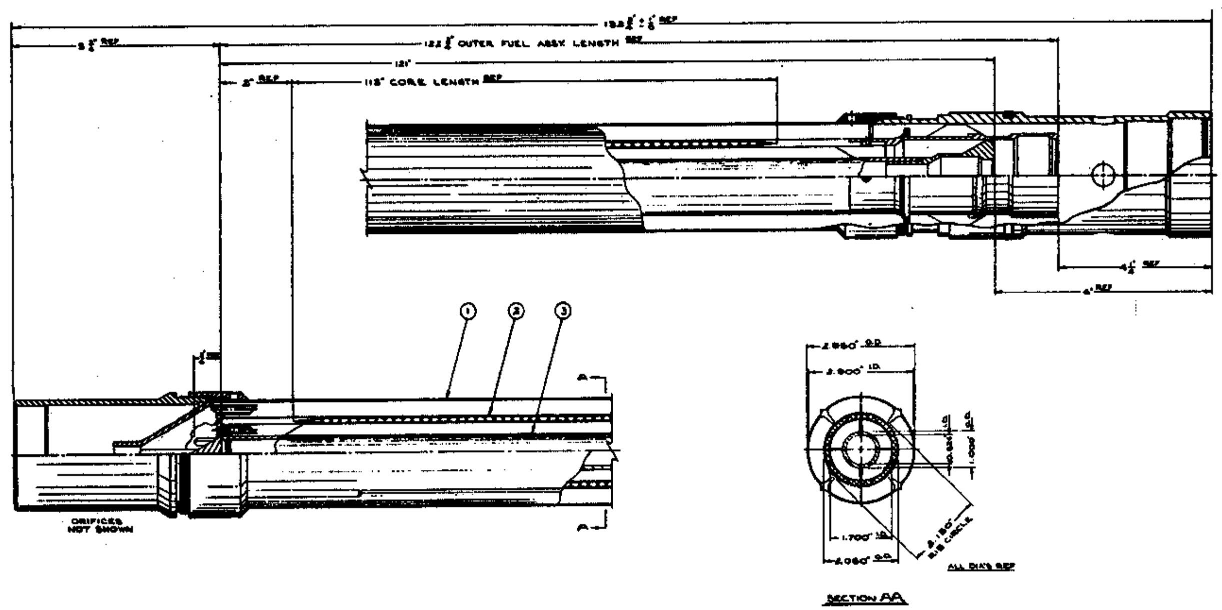


FIG. 6 ETWO FUEL ASSEMBLY

(1) Fuel Element Housing Assembly
(2) ETWO Outer Fuel Tube Assembly
(3) Inner Housing Weldment

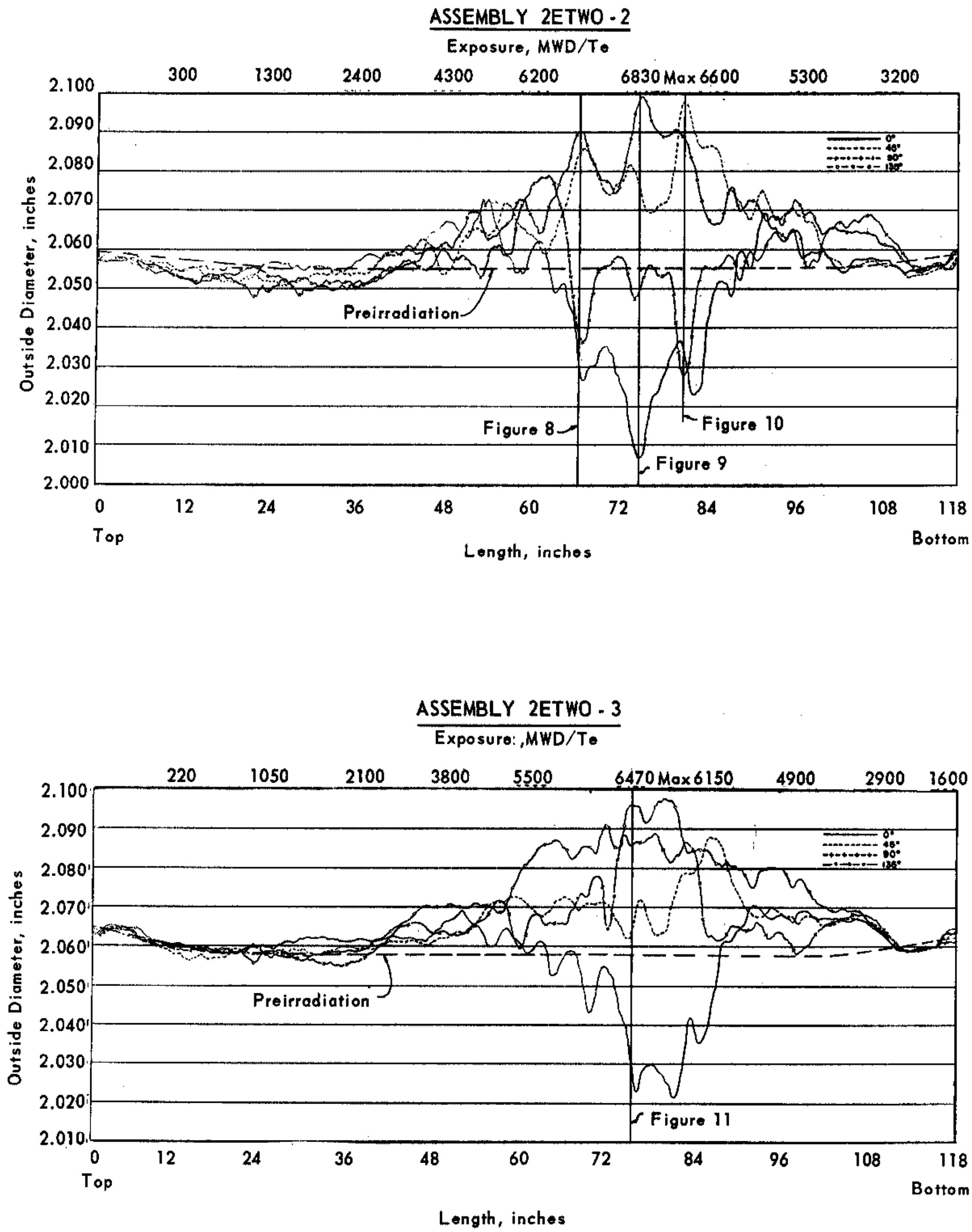


FIG. 7 OUTSIDE DIAMETER ETWO ASSEMBLIES

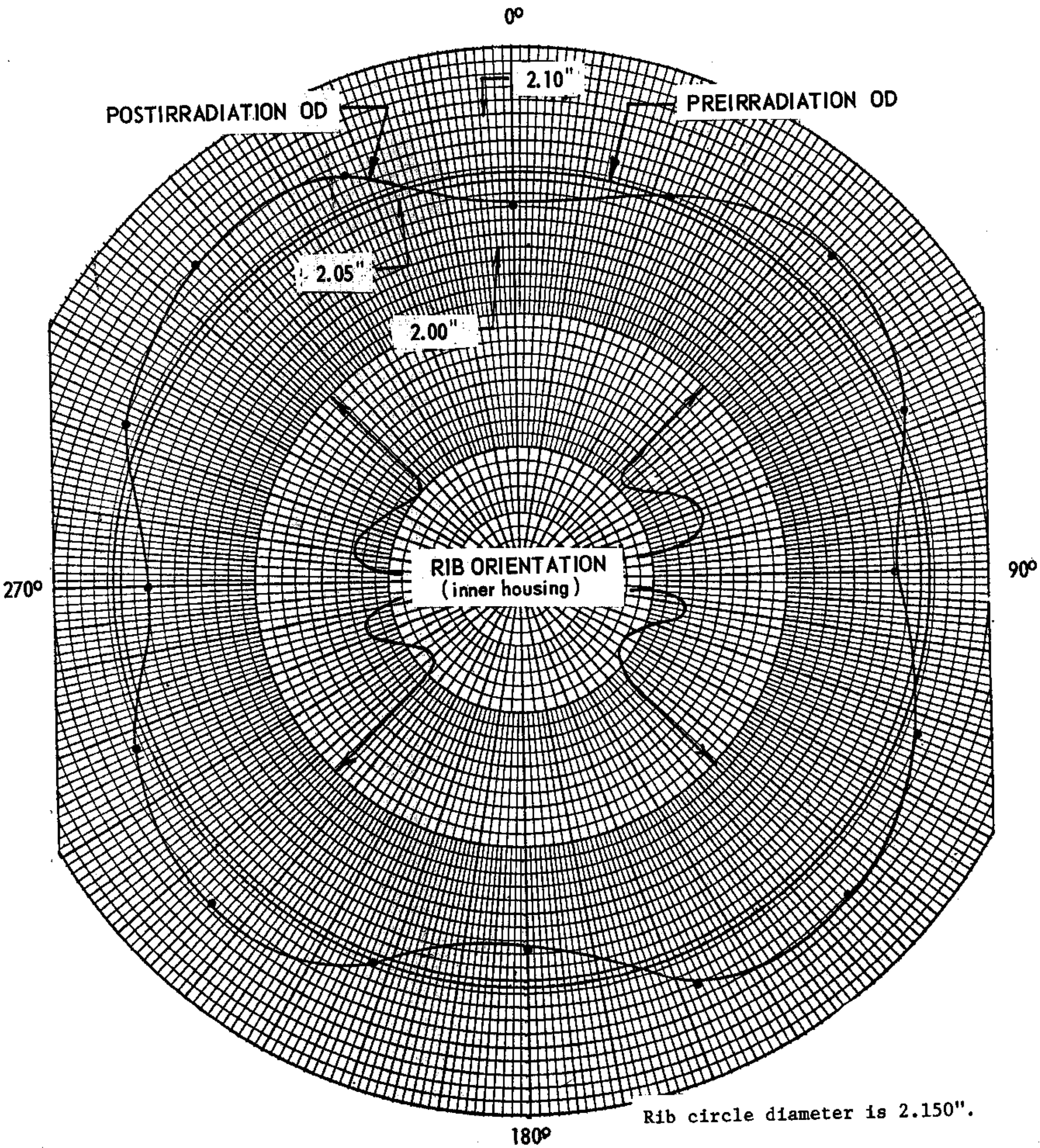


FIG. 8 OUTSIDE DIAMETER OF 2ETWO-2 FUEL TUBE, SECTION 1
Measured at elevation corresponding to exposure of 6400 MWD/Te.
(See Fig. 7)

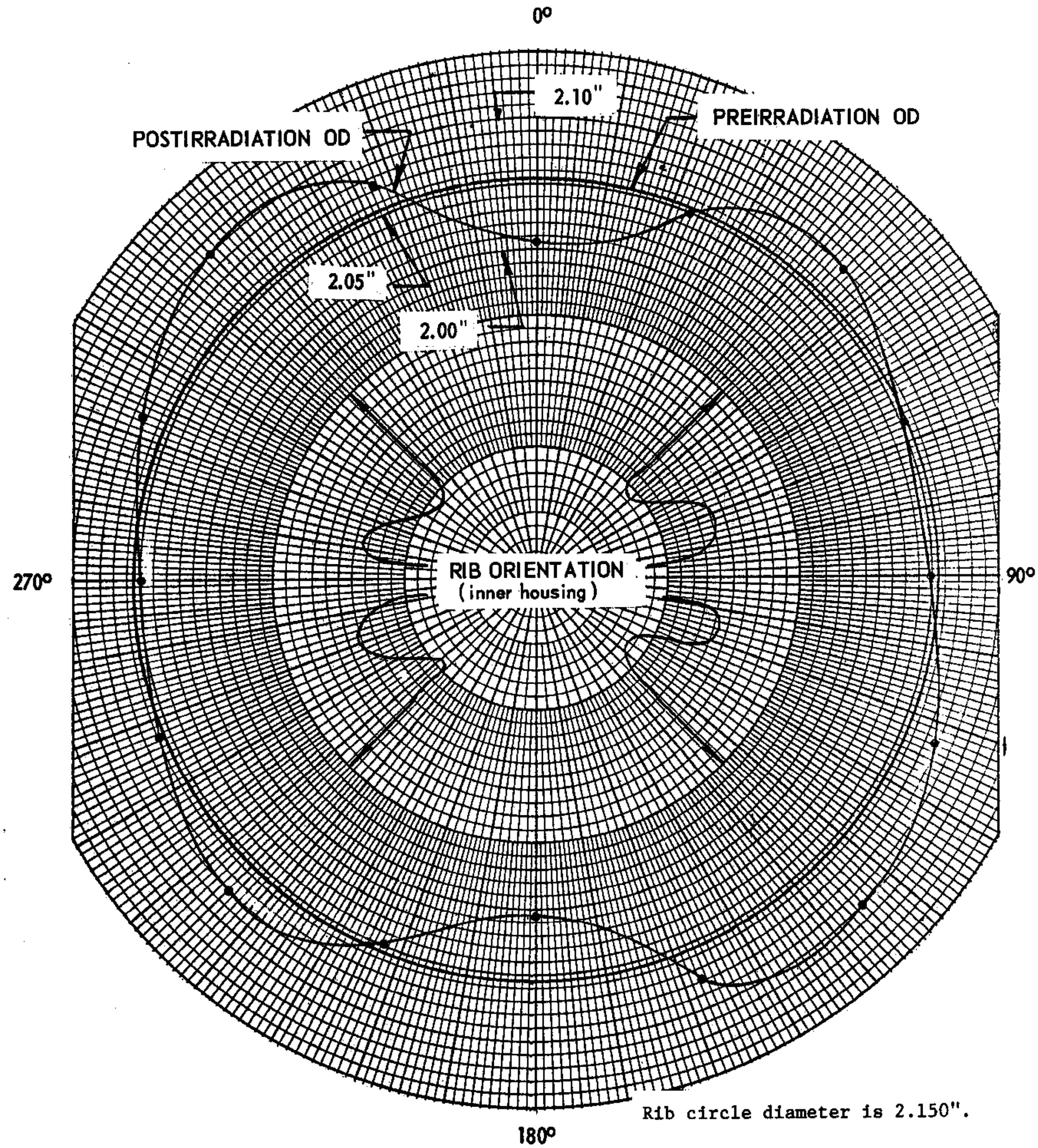


FIG. 9 OUTSIDE DIAMETER OF 2ETWO-2 FUEL TUBE, SECTION 2
Measured at elevation corresponding to exposure of 6830 MWD/Te.
(See Fig. 7)

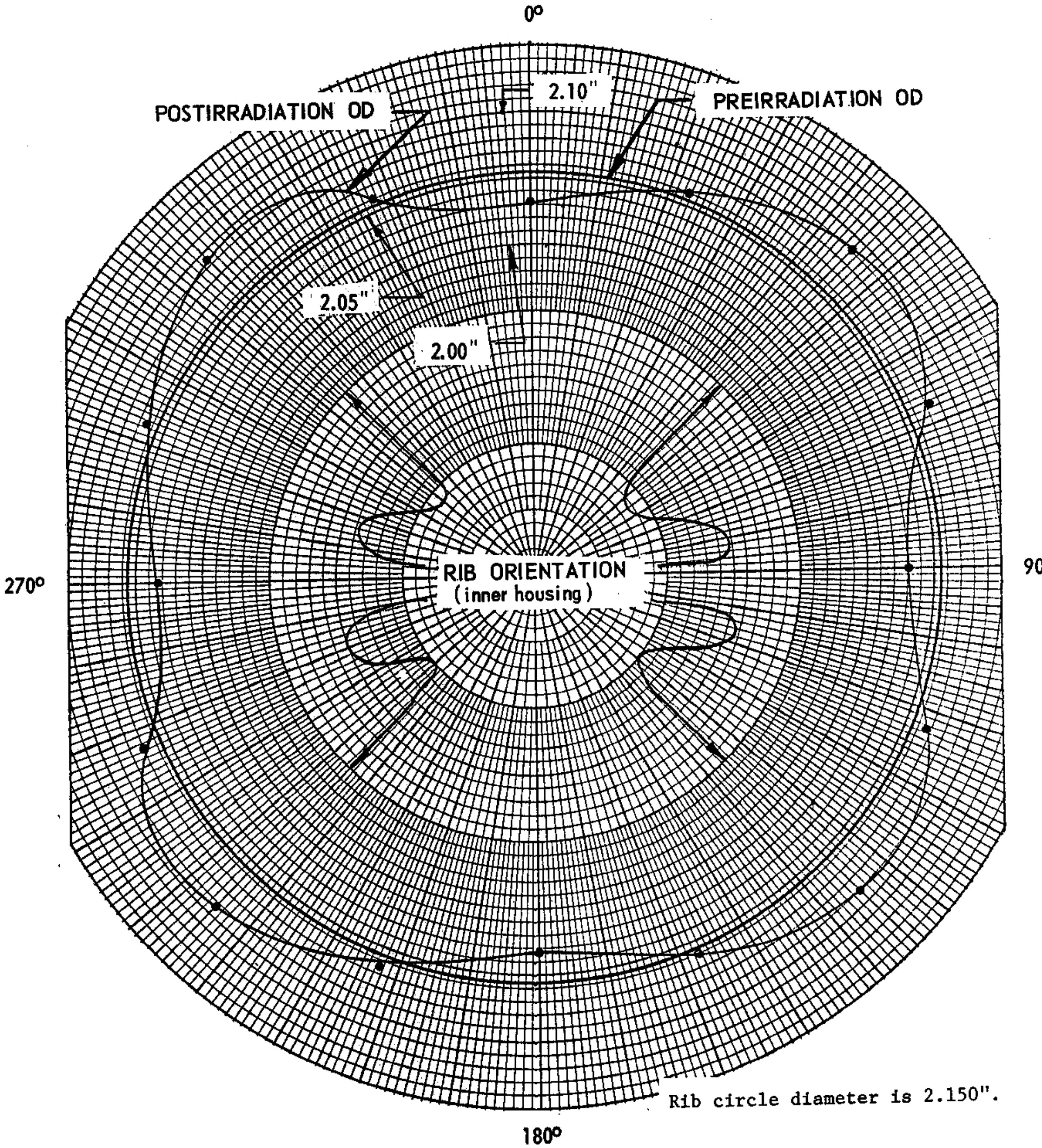
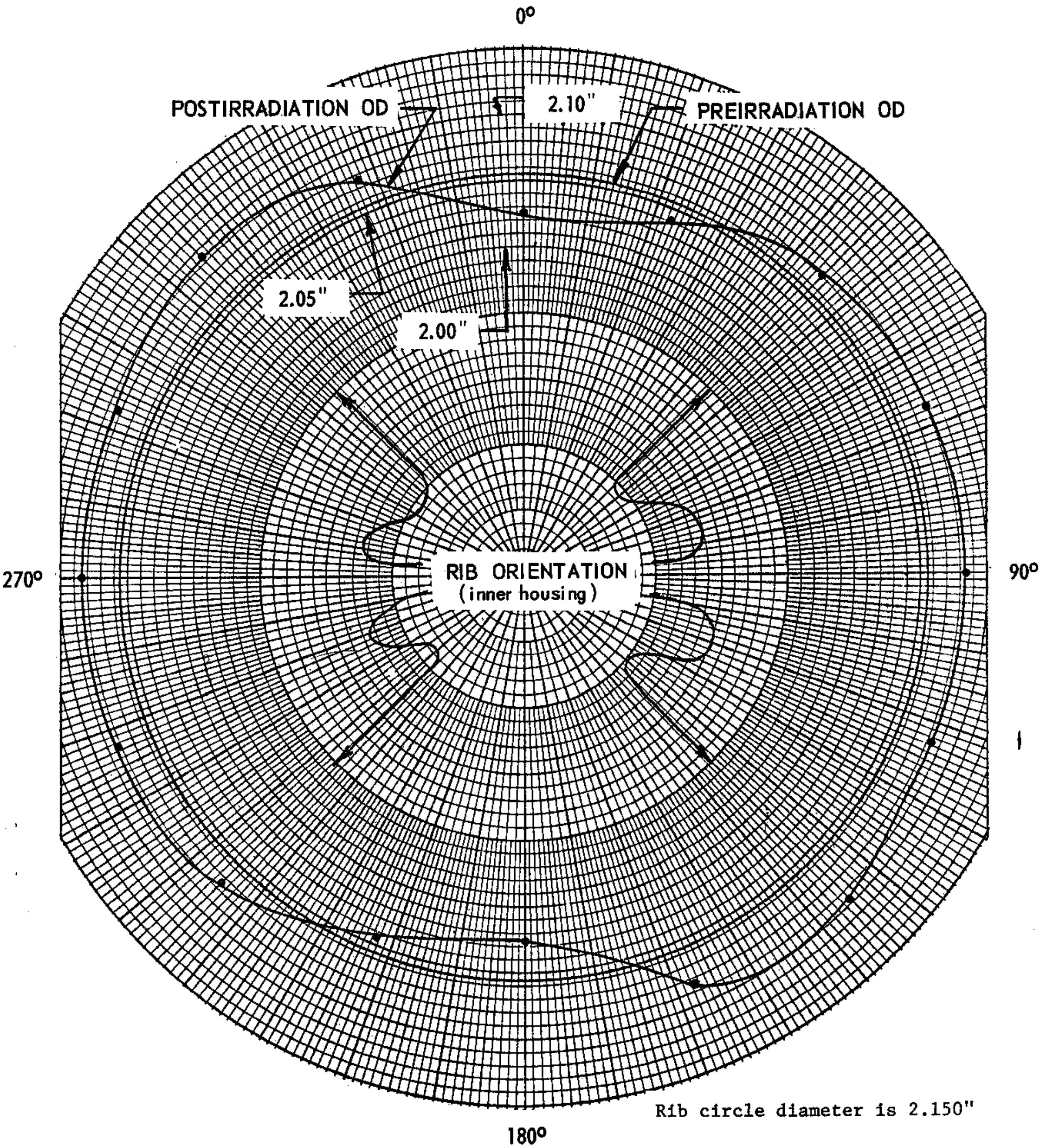


FIG. 10 OUTSIDE DIAMETER OF 2ETWO-2 FUEL TUBE, SECTION 3
Measured at elevation corresponding to exposure of 6650 MWD/Te.
(See Fig. 7)



Rib circle diameter is 2.150"

FIG. 11 OUTSIDE DIAMETER OF 2ETWO-3 FUEL TUBE, SECTION 1
Measured at elevation corresponding to exposure of 6470 MWD/Te.
(See Fig. 7)

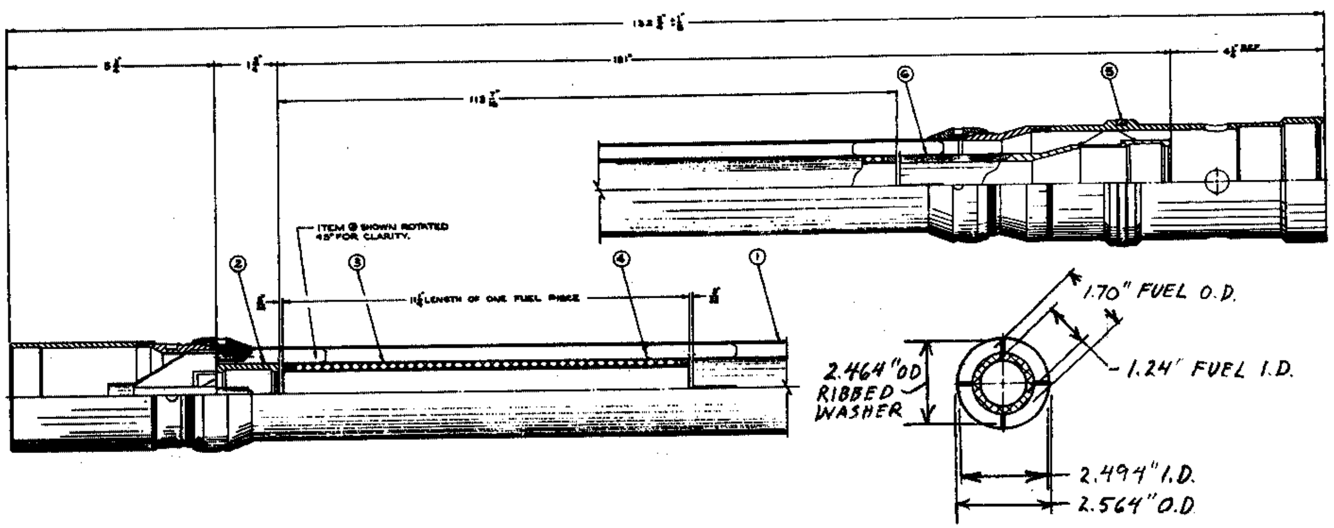


FIG. 12 SEGMENTED METAL TUBE (SMT) ASSEMBLY

(1) Outer Housing Assembly
(2) Bottom Spacer Weldment
(3) SMT - 1 Fuel Piece
(4) Ribbed Washer
(5) Piston Rings (HWCTR)
(6) Top Fitting

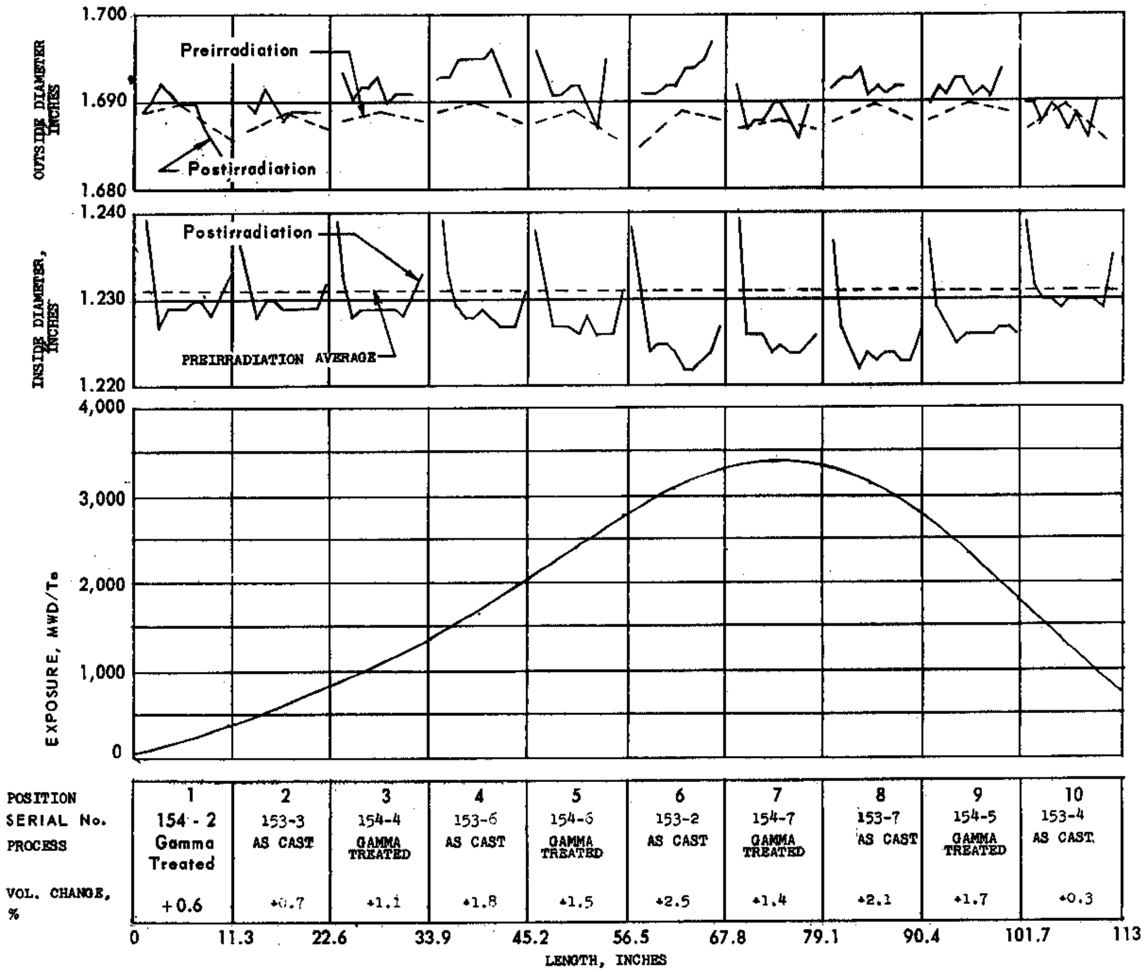


FIG. 13 HWCTR TEST ASSEMBLY SMT 1 – 2

Postirradiation Measurements after 3300 MWD/Te U Exposure

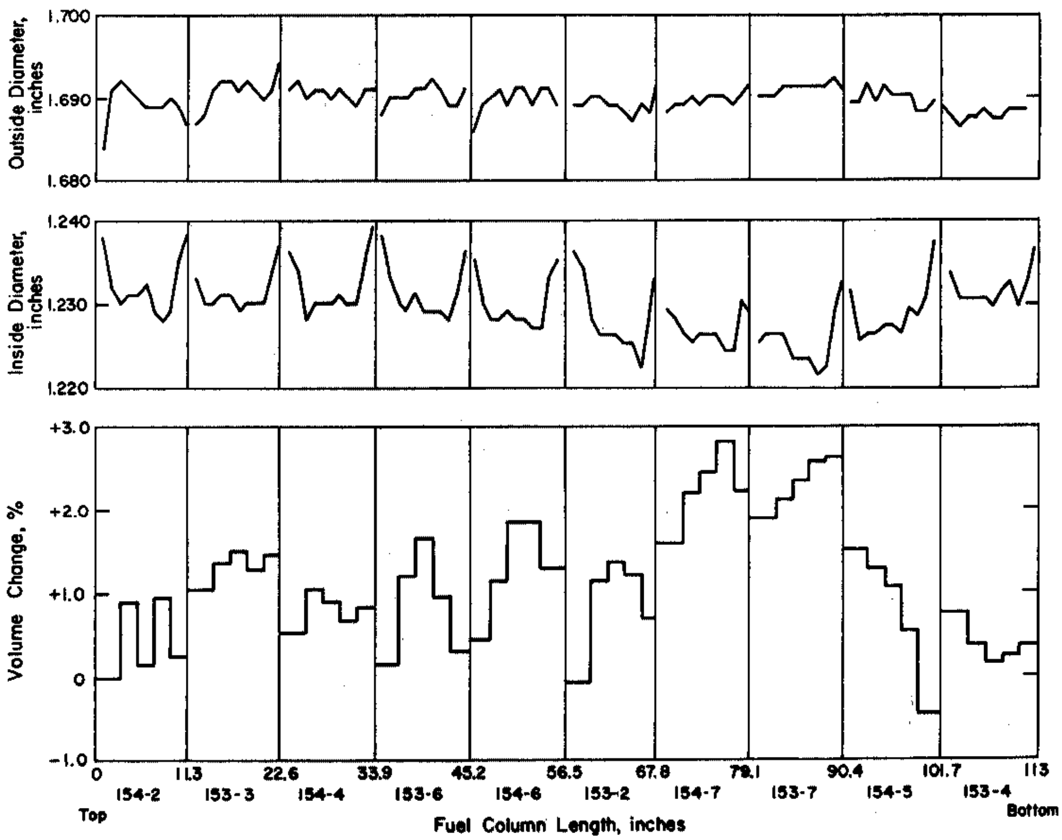


FIG. 14 HWCTR TEST ASEMBLY SMT - 1 - 2
Final Postirradiation Measurements

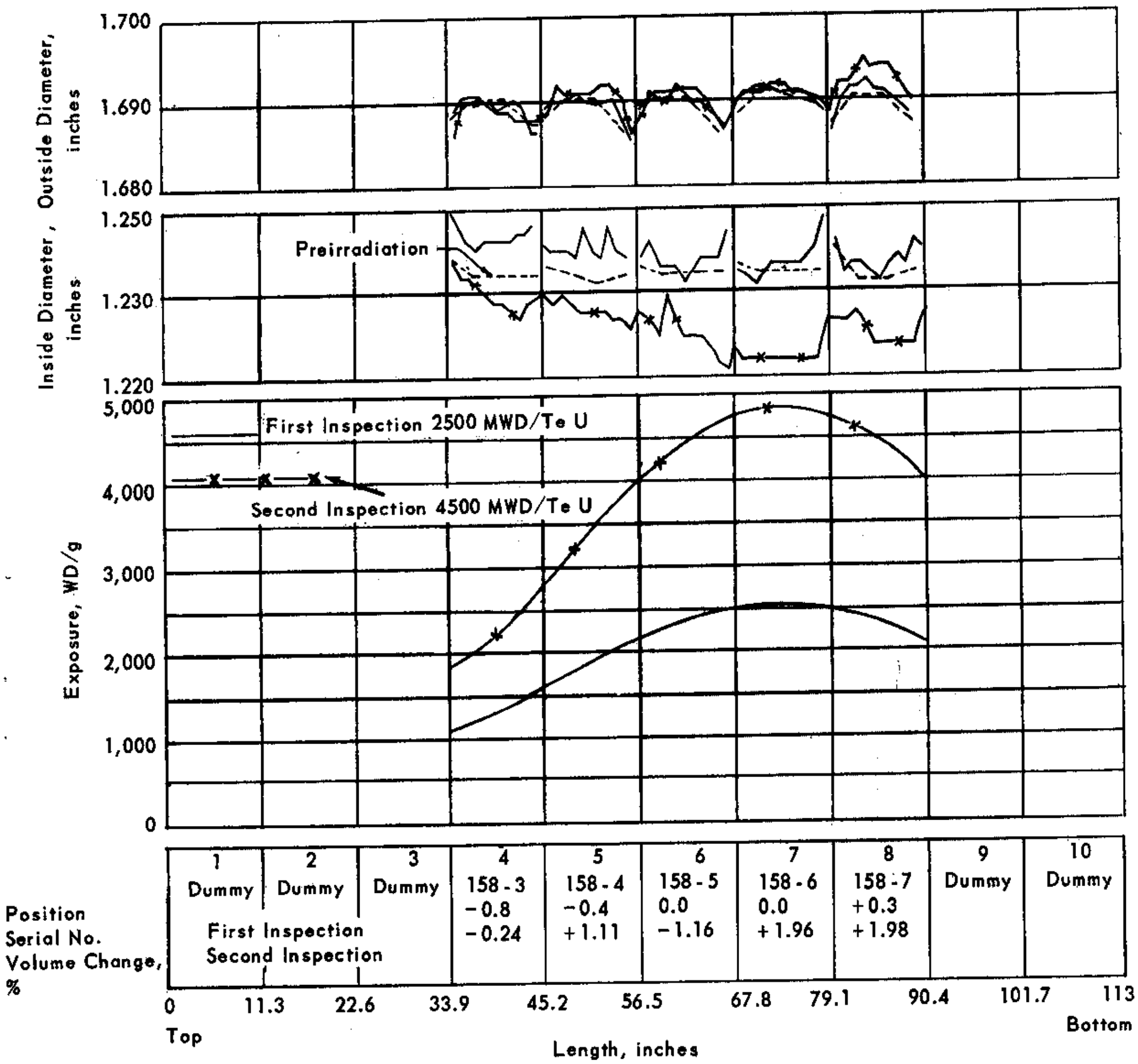


FIG. 15 HWCTR TEST ASSEMBLY SMT-1-3

Postirradiation Measurements, after 2500 and 4500 MWD/Te U Exposure

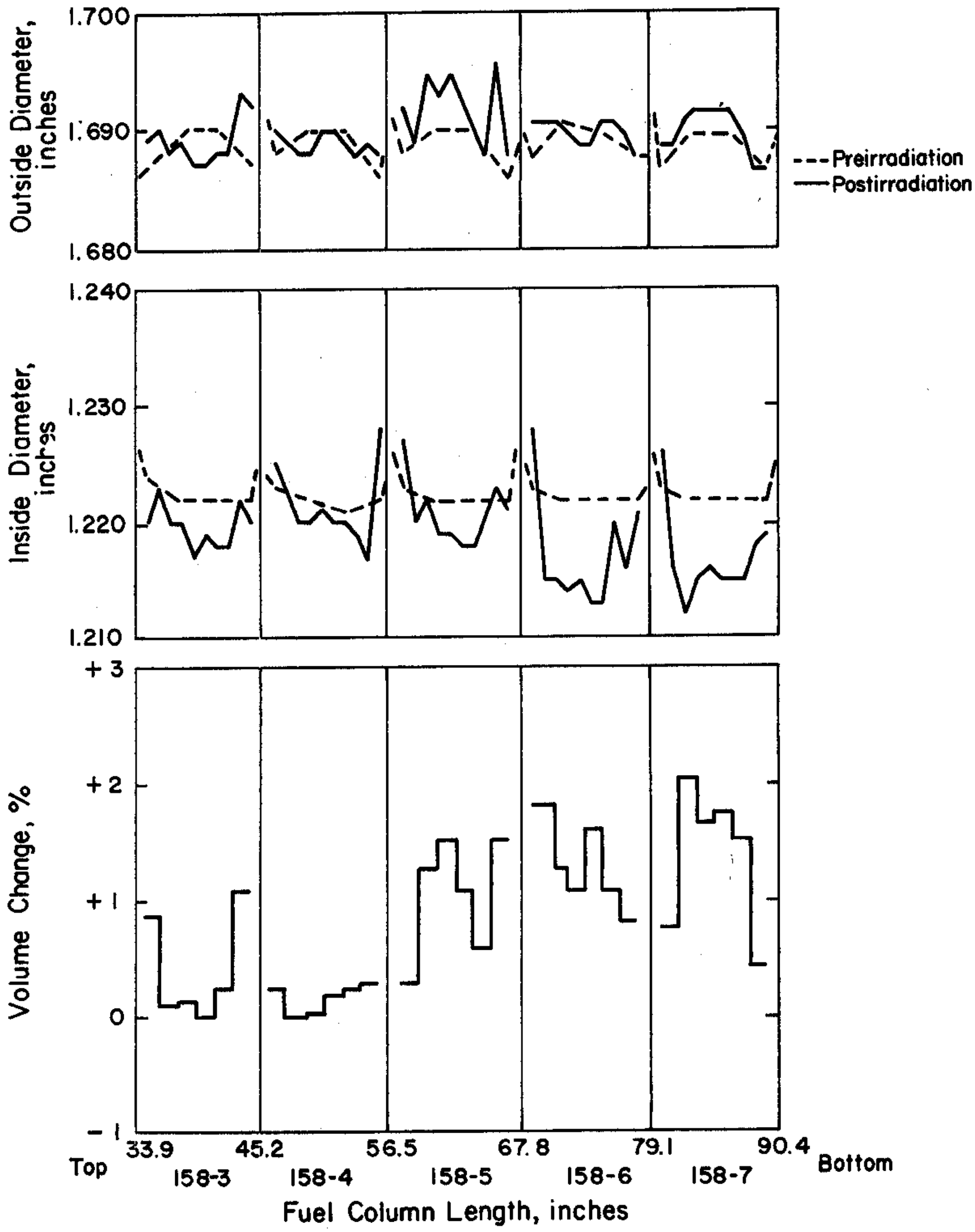


FIG. 16 HWCTR TEST ASSEMBLY SMT-1-3
Final Postirradiation Measurements

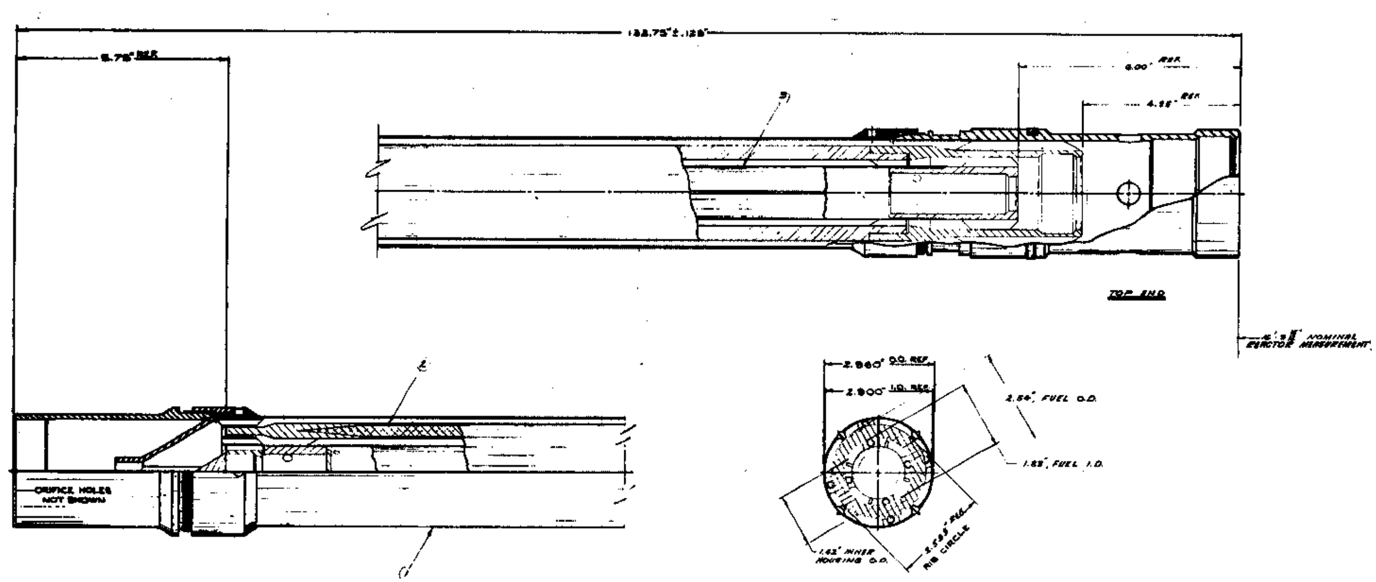


FIG. 17 THORIUM - 1.4 wt% URANIUM FUEL ASSEMBLY

(1) Outer Housing Tube Assembly
(2) Fuel Tube Assembly
(3) Inner Housing Assembly

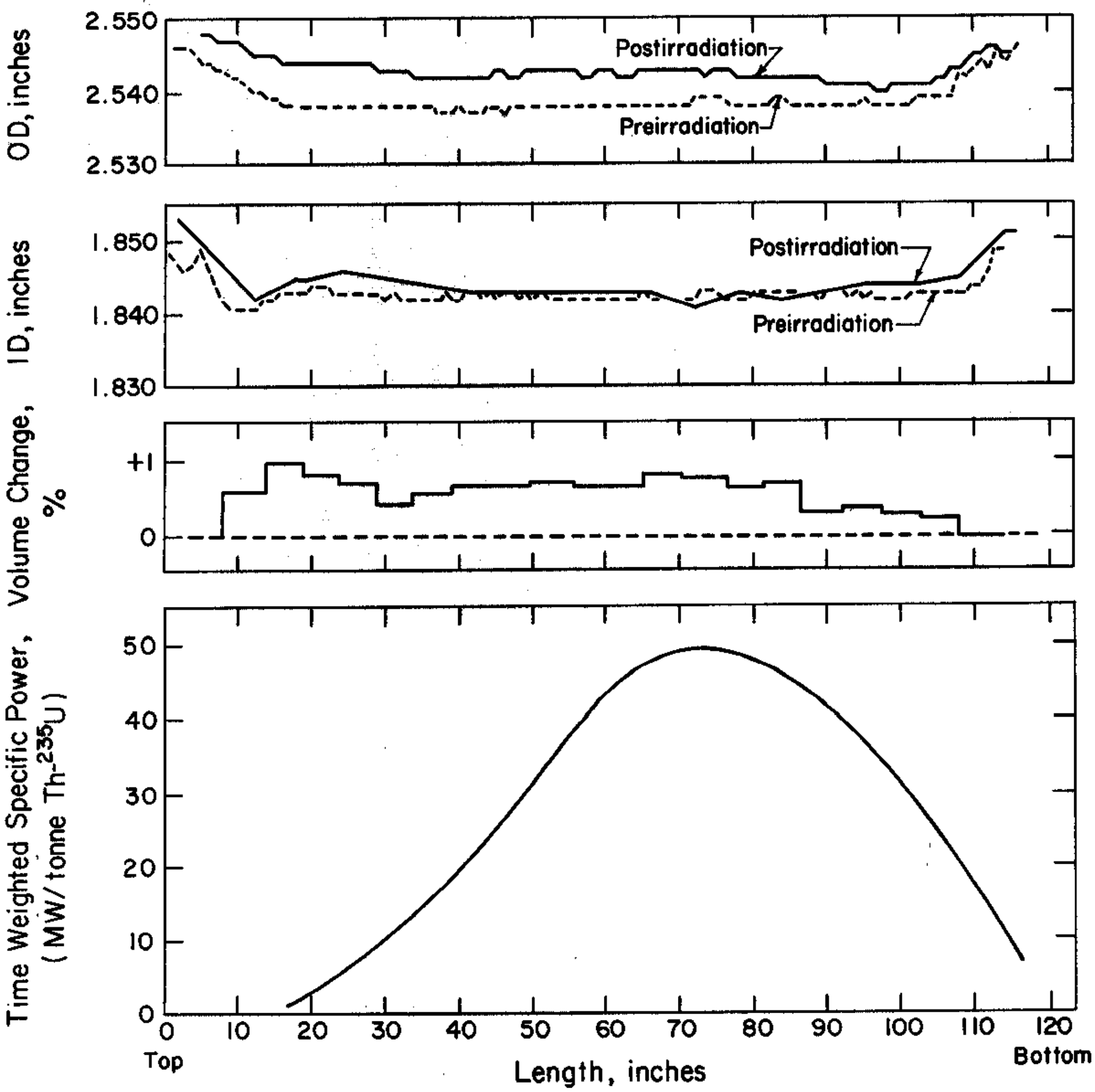


FIG. 18 THORIUM – 1.4 wt% ^{235}URANIUM METAL FUEL TUBE – HWCTR ASSEMBLY NO. TMT 1 – 2, Postirradiation Measurements

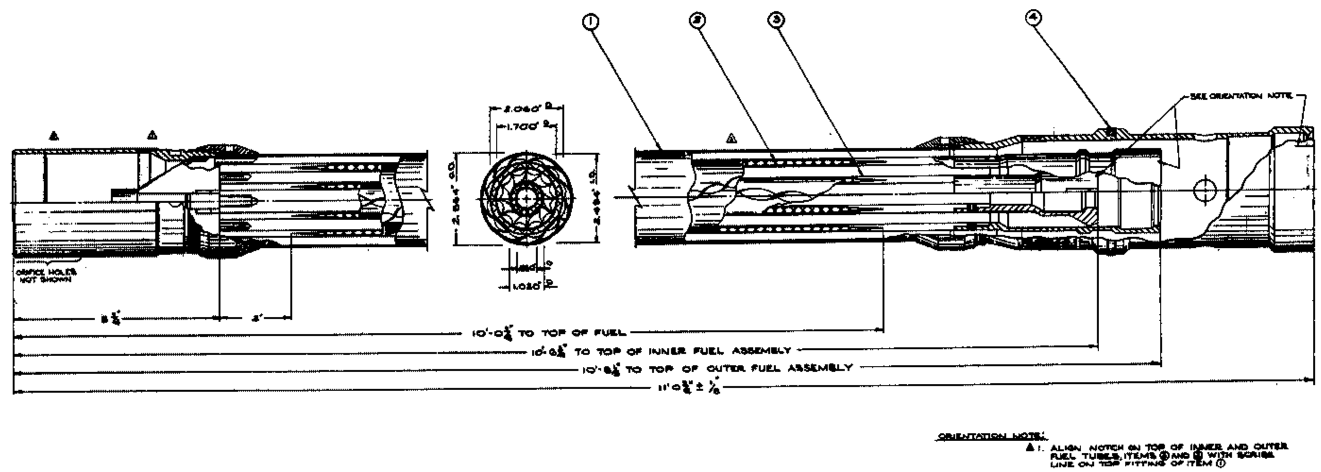
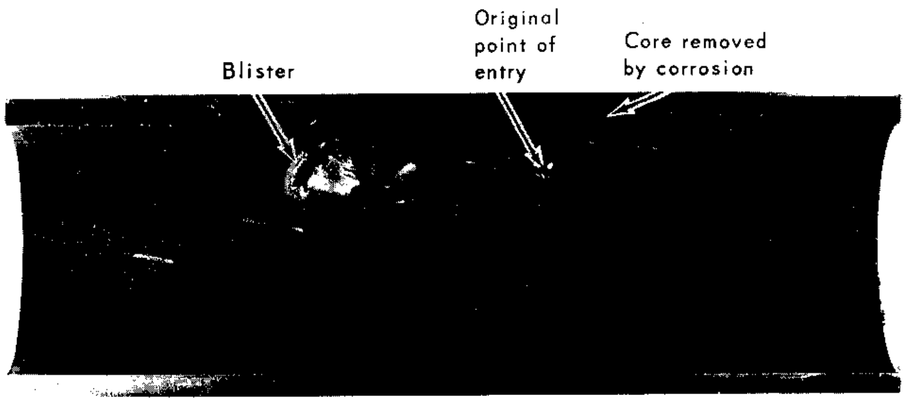


FIG. 19 THIN-WALL NESTED TUBE (TWNT) FUEL ASSEMBLY

(1) TWNT Outer Housing Subassembly
(2) Outer Fuel Tube Subassembly
(3) Inner Fuel Tube Subassembly
(4) Piston Ring (HWCTR)

PLATE DPSTF-1-5179

FIG. 20 THIN-WALL INNER AND OUTER FUEL TUBES
Note twisted ribbon spacers



NEG. 47349

Failed section of element showing the areas of mechanical wear against the ribbon, the blister, and an area where the core was removed.

NEG. 47347

The opposite side of failed section showing the blister.

NEG. 47955

Areas of mechanical wear on inner surface of housing tube. Maximum penetration at worn area is 32 mils.

FIG. 21 OUTER FUEL TUBE SURFACE OF TWNT ASSEMBLY WHICH FAILED IN OUT-OF-PILE FLOW TEST AFTER 15 DAYS IN WATER AT 250 gpm, 260°C, AND 1000 psig $\sim$1X

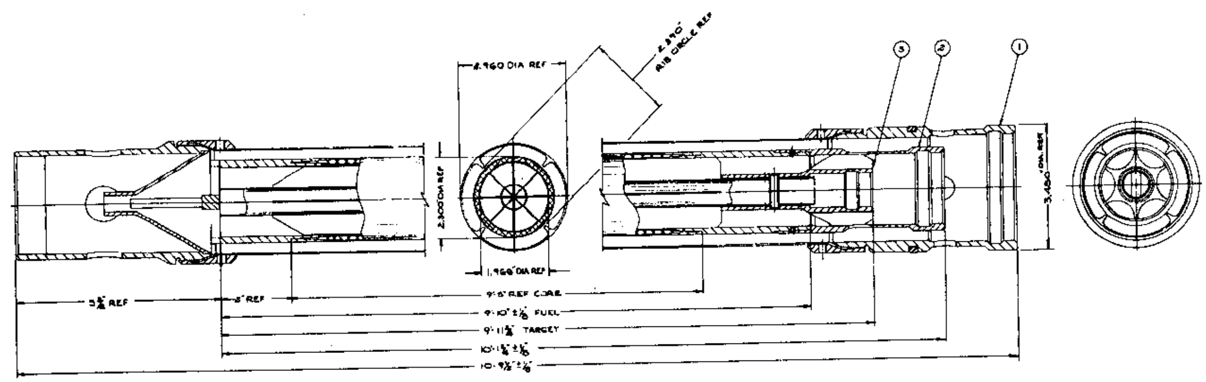


FIG. 22 HWCTR DRIVER ASSEMBLY

(1) Housing Assembly HWCTR Fuel Elements
(2) Fuel Assembly – Driver HWCTR
(3) Target Assembly HWCTR

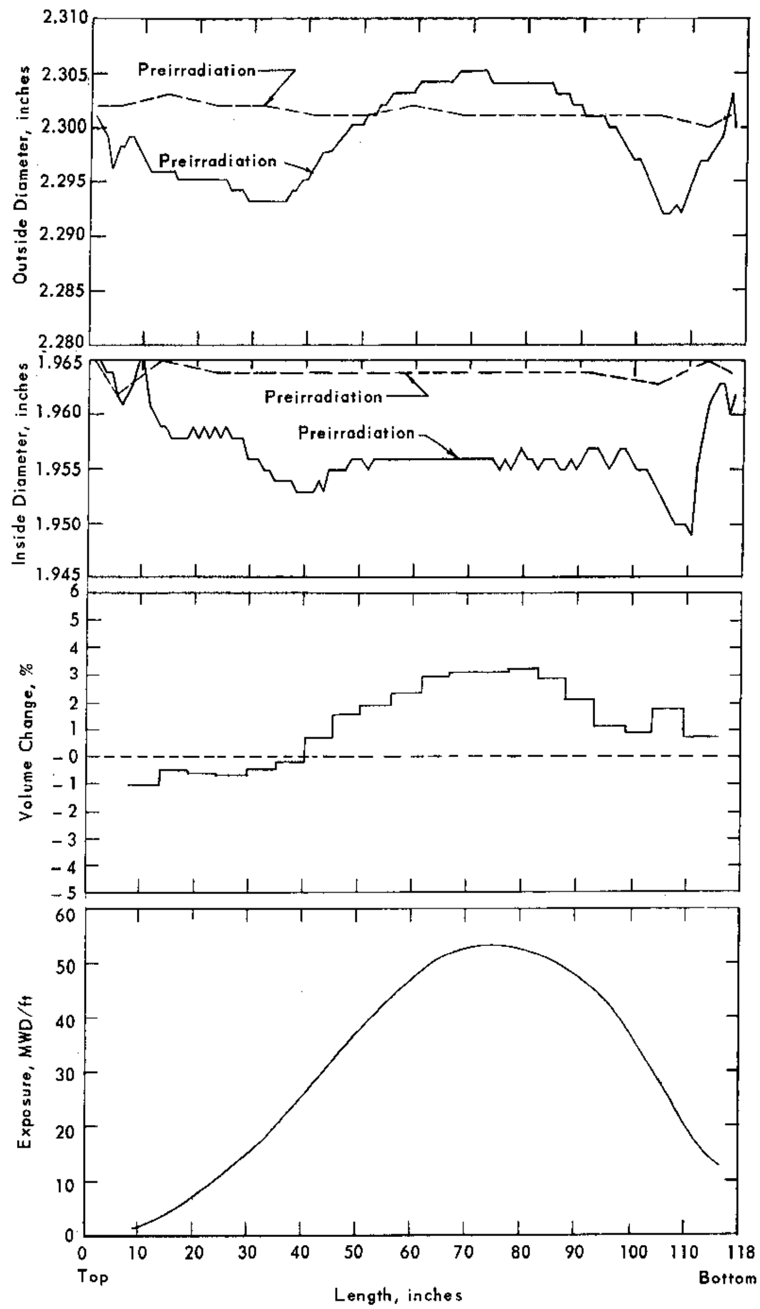


FIG. 23 HWCTR M-1 DRIVER ASSEMBLY NO. 15, FUEL TUBE NO. 22
Interim Postirradiation Measurements

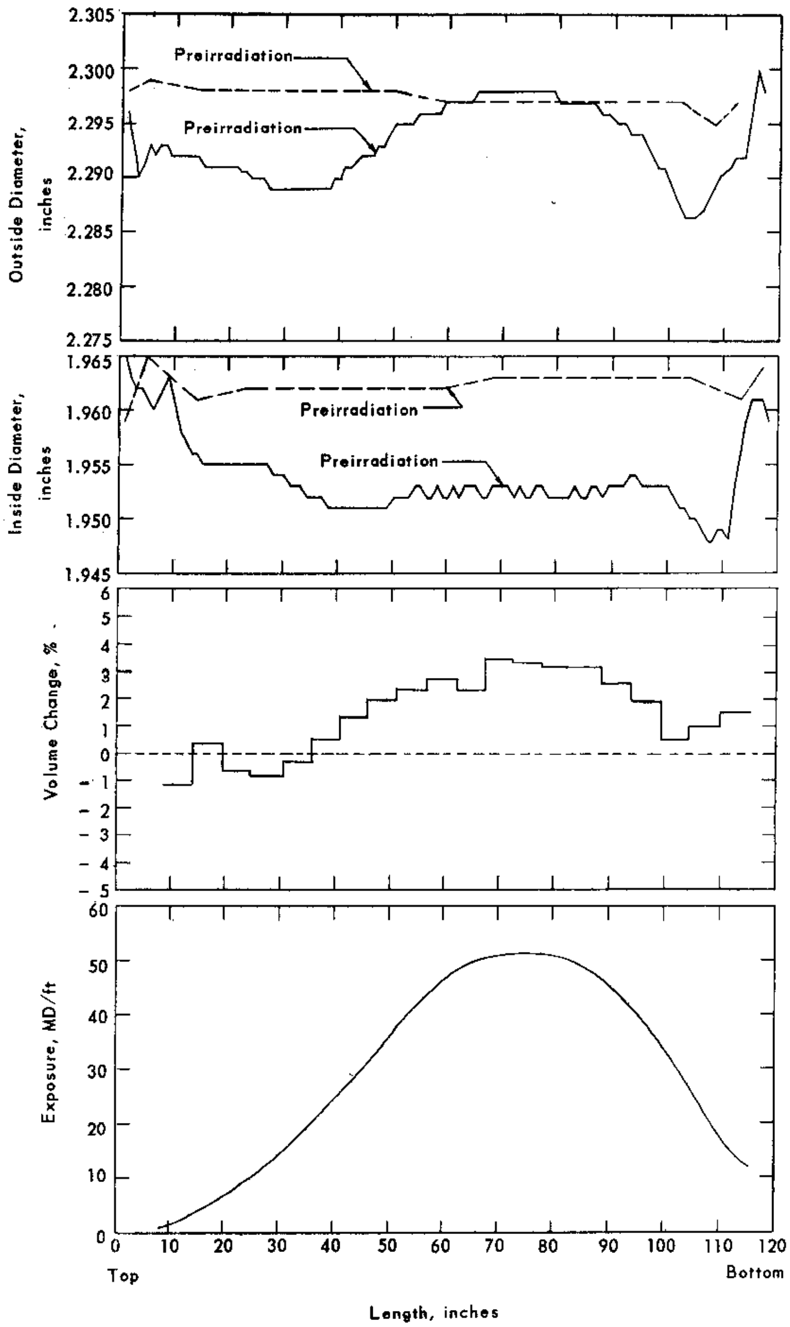


FIG. 24 HWCTR M-1 DRIVER ASSEMBLY NO. 11, FUEL TUBE NO. 48
Interim Postirradiation Measurements

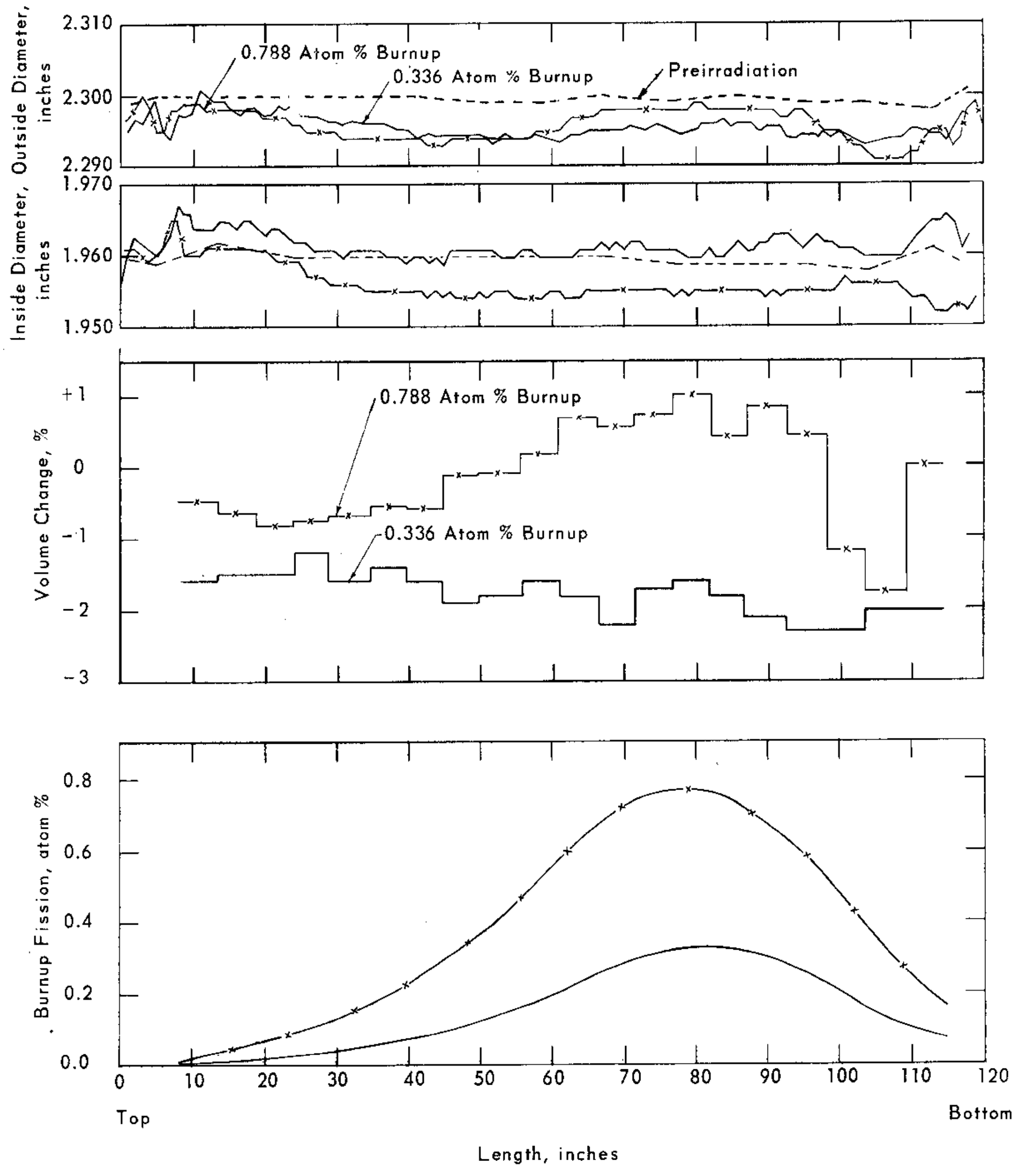


FIG. 25 HWCTR M-2 DRIVER ASSEMBLY NO. 22, FUEL TUBE NO. 1
Interim Postirradiation Measurements

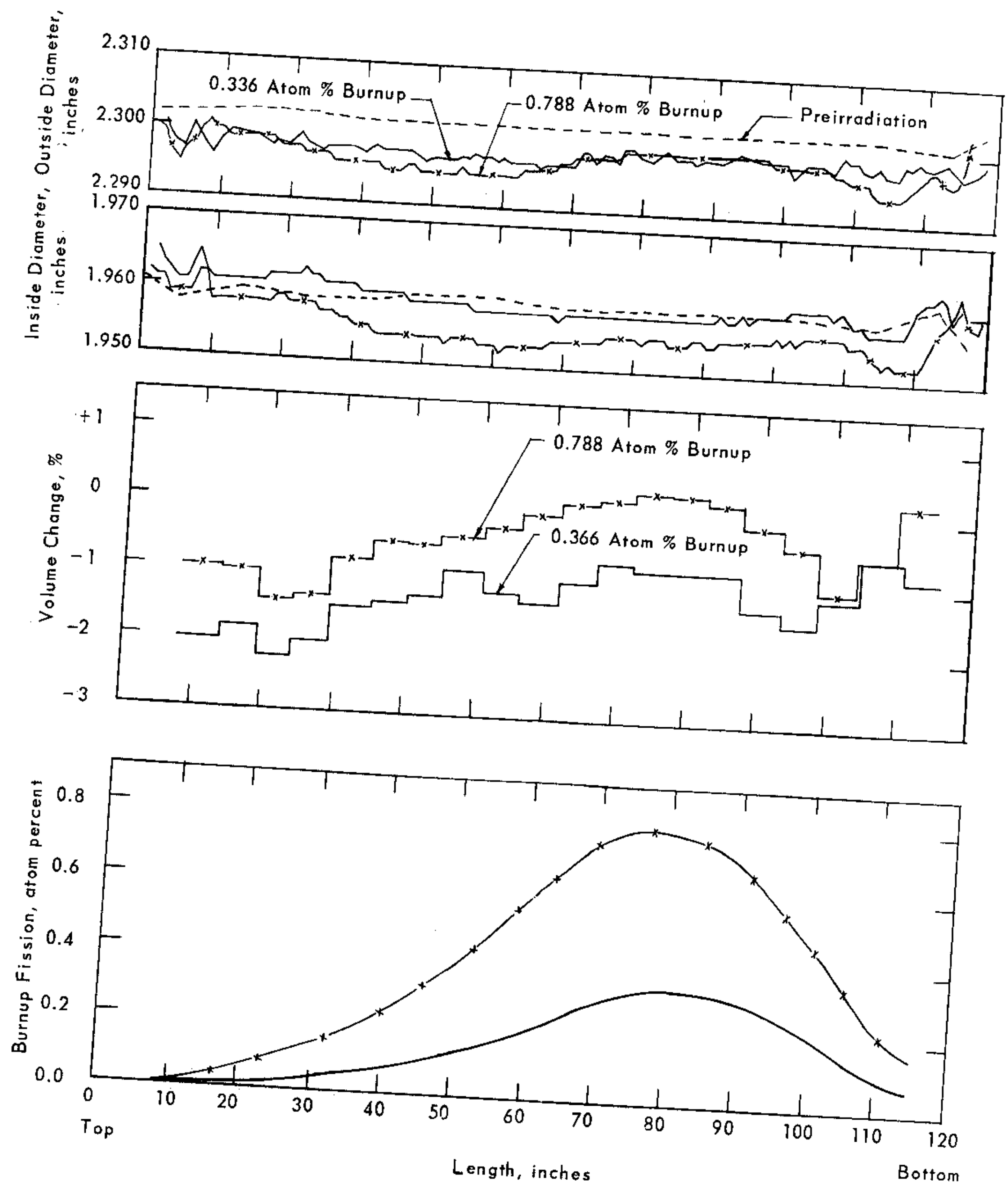


FIG. 26 HWCTR M-2 DRIVER ASSEMBLY NO. 19, FUEL TUBE NO. 18
Interim Postirradiation Measurements

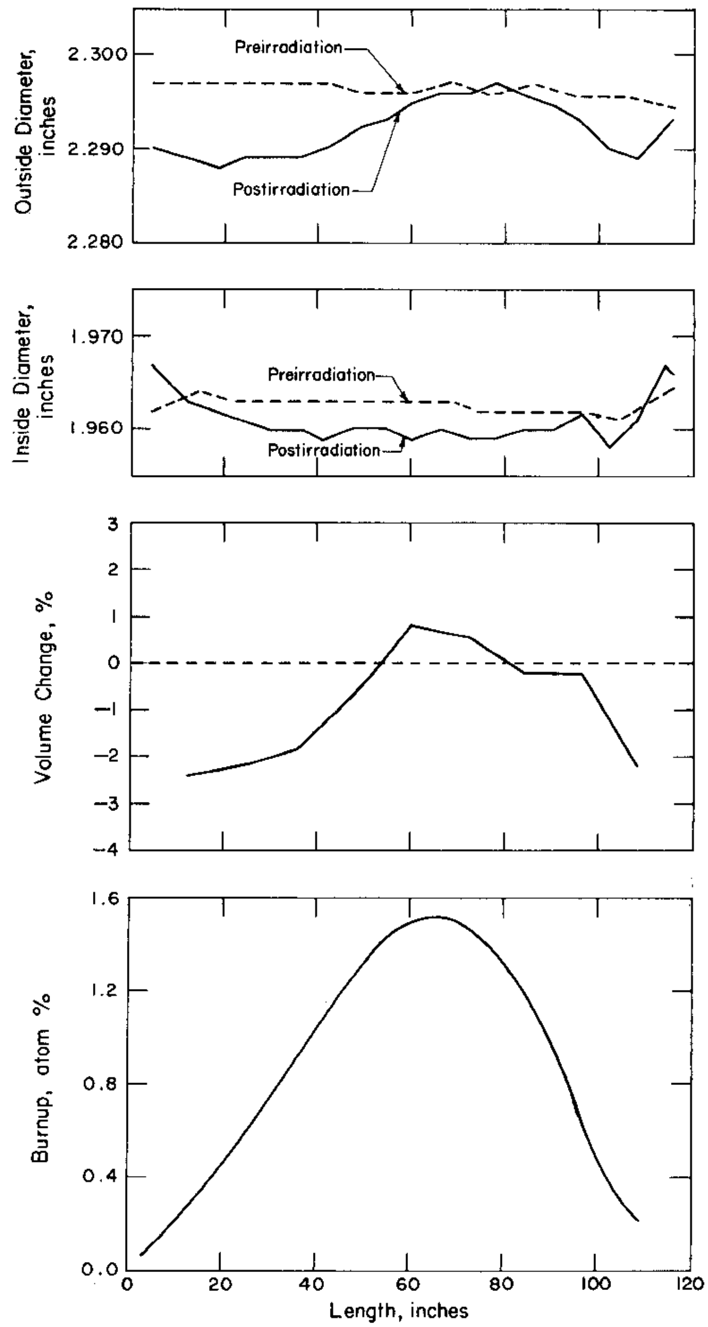


FIG. 27 HWCTR M-2 DRIVER ASSEMBLY NO. 22, FUEL TUBE NO. 1
Final Postirradiation Measurements

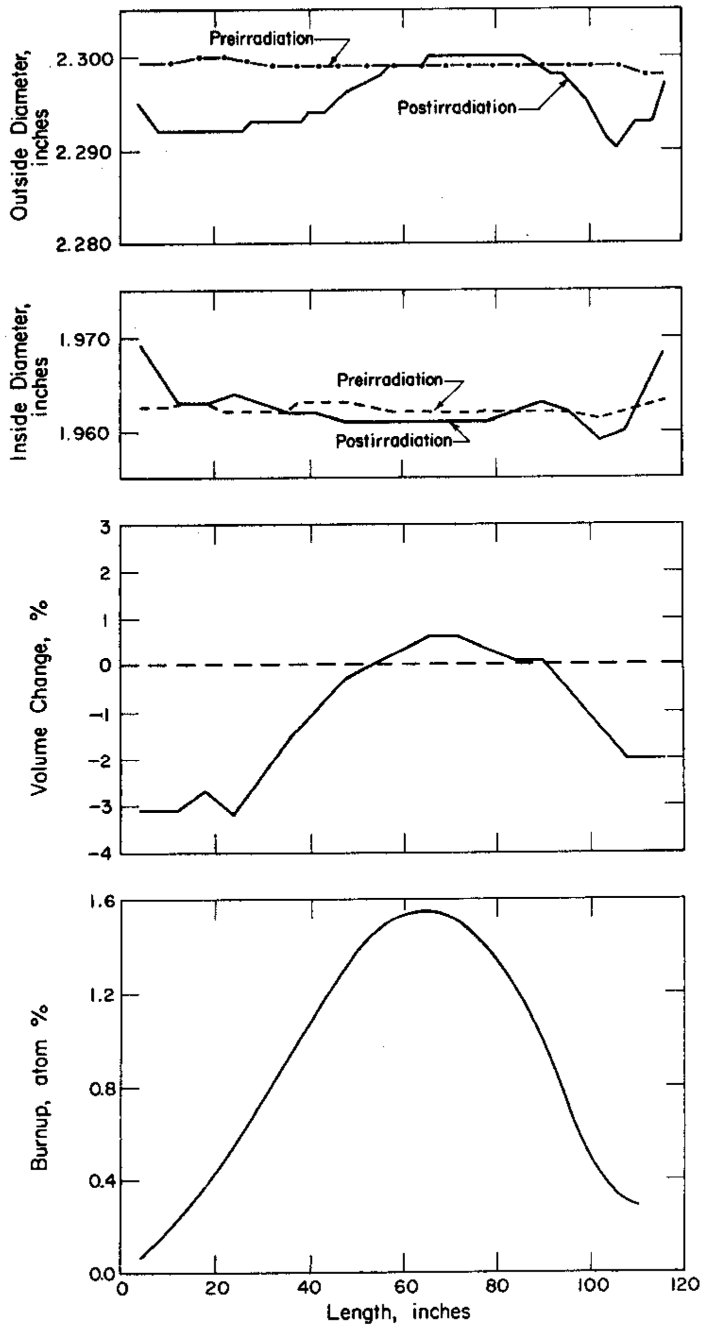


FIG. 28 HWCTR M-2 DRIVER ASSEMBLY NO. 19, FUEL TUBE NO. 18
Final Postirradiation Measurements