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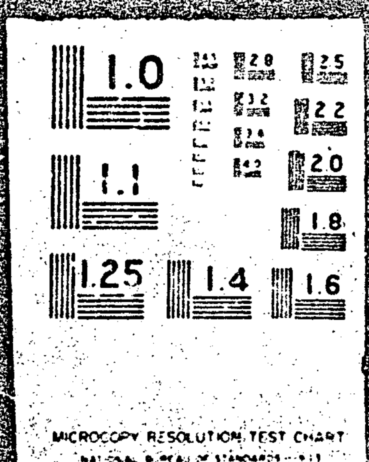

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16. Abstract

The design and successful operation of three tantalum alloy (Ta-8W-2Hf) clad uranium mononitride (UN) fuel pins irradiated for 8070 hr at 990° C (1815° F) is described. Two pin diameters having measured burnups of 0.47 and 0.90 uranium atom percent were tested. No clad failures or swelling was detected; however, postirradiation clad samples tested failed with 1 percent strain. The fuel density decrease was 2 percent, and the fission gas release was less than 0.05 percent. Isotropic fuel swelling, which averaged about 0.5 percent, was less than fuel pin assembly clearances. Thus the clad was not strained. Thermocouples with a modified hot zone operated at average temperatures to 1100° C (2012° F) without failure. Factors that influence the ability to maintain uniform clad temperature as well as the results of the heat transfer calculations are discussed.

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IRRADIATION OF THREE T-111 CLAD URANIUM NITRIDE FUEL PINS FOR 8070 HOURS AT 990° C (1815° F)

by Jack G. Slaby, Byron L. Siegel, Louis Gedeon, and Robert J. Galbo

Lewis Research Center

SUMMARY

The design and successful operation of three T-111 tantalum alloy (Ta-8W-2Hf) clad uranium mononitride (UN) fuel pins irradiated for 8070 hours at 990° C (1815° F) in the Plum Brook Reactor Facility is described. Two diameters of fuel pins were tested, a 1.822-centimeter (0.717-in.) diameter pin (reference diameter pin) and a 0.914-centimeter (0.360-in.) diameter pin (half-diameter pin). The reference diameter pin was run at the reference burnup rate of 2.73 percent in 50 000 hours, and the half diameter pin was run at twice the reference burnup rate. The fuel length was 5.72 centimeters (2.25 in.).

No T-111 clad failures were detected in these three fuel pins. A thermocouple well located in the end cap of one of the half-diameter fuel pins leaked. No increase in clad diameter was measured, and fuel swelling was less than the assembly clearance between fuel and clad. Thus the clad was not strained during irradiation. Ductility tests conducted on the irradiated clad samples failed at strains greater than 1 percent. The fuel density decrease was about 2 percent. The average amount of fuel swelling was about 0.5 percent on both the diameter and length of the UN pellets. The measured fuel burnup as determined by the mass spectrometer method was in good agreement with the calculated burnup based on heat-transfer considerations. The amount of fission gas released by the fuel was less than 0.05 percent. The maximum to minimum variations in axial and circumferential burnup values were in most cases less than 25 percent. Highly reliable type K thermocouples, which lasted the duration of the test without failure, were achieved by modifications to convential hot zone designs.

Factors that influence the ability to maintain uniform clad temperature are discussed, and their effect on the clad temperature is shown.

The results of the heat-transfer calculations are discussed, emphasizing that using a conduction path through a stagnant gas gap to remove heat from the fuel pin produces a uniform clad temperature only if the flux profiles do not change.

Thermal neutron shields were incorporated to achieve uniform power, and low fuel enrichment was used to minimize the radial self-shielding of the fuel.

INTRODUCTION

A technology program based on a fast-spectrum liquid-metal-cooled compact reactor concept for space-power application was conducted at the NASA Lewis Research Center. The investigation was keyed to a reference reactor design concept with a reactor power level of 1 megawatt (thermal), a coolant outlet temperature of 950° C (1742° F), and an operating life of 50 000 hours. Three reports (refs. 1 to 3) have been written describing the overall effort. The conceptual reactor design is described in reference 1, the nuclear design and associated initial experiments are discussed in reference 2, and the materials technology and fuel pin test programs are discussed in reference 3.

Part of the fuel-pin test program discussed in reference 3 was conducted in the NASA Plum Brook Reactor Facility. The purpose of the Plum Brook tests was to evaluate the fuel-pin design under conditions of temperature and total burnup of the reference reactor fuel pins. The evaluation of the fuel pins was to be determined by dimensional stability and the ability of the clad to contain both fuel and the fission products. In addition, the test results were to be used to augment and enhance modeling techniques for fuel swelling and clad strain.

The reference reactor fuel pin is clad with T-111, fueled with 38.1 centimeters (15 in.) of uranium mononitride, and has an outside diameter of 1.9 centimeters (0.75 in.). A thin tungsten liner mechanically separates the fuel and clad to prevent a chemical reaction between the fuel and clad. The goal of the reference design concept is not to exceed 1 percent maximum diametral fuel pin cladding strain in 50 000 hours at 990° C (1815° F).

Because of the impracticality of inpile testing for 50 000 hours, irradiations are being carried out over shorter time periods at accelerated test conditions. These tests incorporate higher burnup rates and/or thinner clad.

This report discusses the results including the postirradiation examination of three fuel pins irradiated for 8070 hours in the Plum Brook Reactor Facility. The testing of these three pins is a small part of multiple fuel-pin irradiation test program to investigate fission gas release, materials compatibility under irradiation, and fuel-pin dimensional stability. Two fuel pin diameters were tested - a reference-diameter pin run at the design burnup rate and two half diameter pins run at two times the design burnup rate. This report will also cover the test fuel-pin and capsule design, capsule instrumentation, fabrication techniques, test facility considerations, and heat transfer calculations.

FUEL PIN DIMENSIONS AND TEST CONDITIONS

The clad of the three fuel pins irradiated in the test program was the T-111 alloy (Ta-8W-2Hf). The fuel was uranium mononitride whose density was 94 percent of theoretical density. All fuel pellets were cored, and the fuel length in each fuel pin was 5.72 centimeters (2.25 in.). The short fuel length was to reduce cost and ease the problem of maintaining uniform axial power. Two fuel pin diameters were chosen for the three test pins; a reference-diameter fuel pin and two one-half-diameter fuel pins. Shown in table I are the nominal fuel-pin dimensions. The nominal clad temperature was 990° C (1815° F) with a desired maximum axial variation along the clad of ±40° C (±72° F). As shown in table I the two half-diameter fuel pins were run at burnup rates increased by a factor of two over the reference reactor nominal burnup rate. Consequently, the surface heat flux of both pin sizes was the same and the ΔT across the fuel in the half-diameter pin was one-half that of the reference diameter pin. It should be pointed out that the reference-reactor fuel-pin clad is 0.147 centimeter (0.058 in.) thick but that clad thickness was a variable in the test program. The clad thickness of the reference-diameter fuel pin discussed here was 0.1016 centimeter (0.040 in.). The half-diameter pins were scaled down by a factor of two; consequently, the clad thickness was 0.0508 centimeter (0.020 in.).

FUEL PIN AND CAPSULE ASSEMBLY

The intent of the fuel pin and capsule assembly design was to couple reliability and simplicity into an economical test article. It was felt that a stagnant gas gap would provide a reliable method of temperature reduction from the fuel pin clad to the flowing reactor cooling water surrounding the capsule. The overall fuel pin and capsule assembly design, which is the same for both diameter pins, is shown in figure 1. A brief description of the important design features are presented herein. For further information, reference 4 should be consulted. Reference 4 includes detailed fuel-pin and capsule fabrication procedures, weld qualification procedures, inspection procedures, and assembly procedures.

Fuel-Pin Components

The fuel-pin components are shown in figure 2. The clad, end caps (integral with thermocouple wells), and spherical spacers (used to accommodate axial fuel swelling) were fabricated from the T-111 alloy (Ta-8W-2Hf). The uranium nitride (UN) fuel pellets were fabricated by uniaxial cold pressing in steel dies with camphor as a binder, reference 5. The remaining parts shown in figure 2 were fabricated from tungsten. The purpose of these parts was to physically separate the UN from the T-111 clad end caps and thermocouple wells and thus prevent a reaction between fuel and clad (ref. 3). The tungsten liner which separates the fuel from the clad is a loose liner not integral with either the fuel or clad.

Capsule Components

The capsule components are shown in figure 3. The capsule body and end caps are stainless steel. The fuel pin cone spacers, which center the fuel pin within the capsule, were fabricated from T-111. These spacers were only 0.0127 centimeter (5 mils) thick in order to minimize axial heat conduction losses.

Component Assembly

A cutaway view of the assembled fuel pin and capsule referred to as a capsule assembly is shown in figure 1. There are 19 welds in each reference diameter capsule assembly and 17 in each half diameter capsule assembly. All the welds on the capsule assemblies are electron-beam (EB) welds with the exception of closure welds 7 and 9, which are gas tungsten arc (GTA) welds used to seal helium inside the fuel pin and inside the capsule.

Some important design and fabrication features incorporated in the capsule assembly are as follows:

(1) The fuel pin was completely assembled, EB welded, and backfilled with ultrapure helium and the closure weld made without exposing the fuel to air. This was accomplished in an EB welding chamber, which was modified so it could also be used as a dry box. Ultrapure helium in the weld-chamber dry box was continuously monitored for oxygen and water vapor. Thus, all assembly was performed under controlled environmental conditions.

(2) Closure welds were not made until oxygen and water vapor levels were each below 5 ppm.

(3) All assembly and welding procedures were fully qualified.

(4) Detailed documentation was kept of all procedures used and inspections made during fabrication and assembly of all fuel pins and capsules.

(5) High cleanliness standards were maintained from the time components were cleaned until fuel pins and capsules were assembled.

(6) Thermocouples were fabricated and assembled using stringent cleanliness standards to minimize moisture and air content. To minimize thermal expansion problems during thermal cycling, loose fitting wire, insulation, and sheath were incorporated in the hot zone of the thermocouple. The thermocouple fabrication is discussed in appendix A.

Thermocouples penetrated the capsule end caps and extended into the thermocouple wells, which were integral with the fuel pin end caps. The thermocouple junctions were approximately 3.81 centimeters (1.5 in.) apart. Eight fuel center thermocouples were used in the three capsules (the reference diameter capsule contained two four-wire thermocouples). In addition the reference diameter capsule contained a thermocouple that measured the temperature of the stagnant gas (gas thermocouple) between the fuel-pin end cap and capsule end cap. The junction of this thermocouple was located midway between the fuel pin and capsule end caps (see fig. 1). This thermocouple served as a low-temperature backup thermocouple in the event that the fuel center thermocouples failed.

TEST FACILITY AND ENVIRONMENT

The three capsules were clustered together in a triangular array: one reference diameter capsule with two half-diameter capsules. The arrangement of the capsules within the capsule holder assembly is shown in figure 4. The capsule holder assembly can move vertically about 25.4 centimeters (10 in.) within the reactor test location to adjust the capsule temperature during reactor operation. Vertical movement of the capsule assemblies as a unit is accomplished by using a vertical adjustable facility tube (VAFT) as described in reference 6. This is an electromechanical device for remotely positioning a capsule assembly holder relative to the reactor core.

Individual capsule temperature adjustment was made in two ways: The temperatures of the two half-diameter pins were adjusted relative to each other by rotating the capsule assembly holder relative to the reactor core. The temperature of the reference-diameter pin was adjusted relative to the two half-diameter pins by changing the shape and thickness of the hafnium shield attached to an aluminum shroud that housed the fuel pins.

The capsules were irradiated in the reflector region of the Plum Brook Reactor as shown schematically in figure 5, with the reference-diameter capsule facing the reactor core. The type of test reactor and the experiment location in the test reactor lead to nonuniform power generation in the test fuel pins. In this particular experiment there are three separate effects that lead to nonuniform power generation. One is the radial power variation caused by the thermal absorption (U^{235} self-shielding) when testing in a thermal reactor. The second is the circumferential power variation caused by reflector location and the perturbations of other test pins. The third is the axial power variation caused by end peaking in the test pins and by changes in the shape of the axial flux profile as the reactor control rods are withdrawn.

To achieve meaningful irradiation data, acceptable variations in power generation in the fuel pins were established. The following was used as the criteria for acceptable power variation:

Maximum radial power-density variation, max/min	1.5
Maximum circumferential power-density variation, max/min	1.3
Maximum axial power-density variation, max/min	1.5

Calculations were made to study the affect of thermal absorption (U^{235} self-shielding) when testing in a thermal reactor. The calculated radial power distribution across the fuel thickness of a fully enriched reference-diameter pin in a fast spectrum (the intended application is a fast-spectrum reactor) is flat as shown in figure 6 taken from reference 7. Also shown are the fission power profiles across different enrichment pins in the reflector region of the Plum Brook Reactor. The ratio of outside to inside power (max/min) is also shown on the curves. An enrichment as low as 5 percent still shows an outside to inside power drop off of 1.35 to 1.0. As the enrichment is increased significantly, the ratio becomes intolerable. The results of the mockup reactor tests conducted to determine the fuel enrichment corresponding to the required fission density, yet low enough to preclude large radial self-shielding effects are given in reference 8. Figure 7 shows the results of the average fission power densities for a range of enrichments for both the reference diameter and half-diameter fuel pins. These enrichment-power curves along with the self-shielding calculations and test-hole flux levels provided the basis for selecting the reference-diameter and half-diameter fuel pin enrichments as 5.6 and 8.2 percent, respectively.

Based on these enrichment-power measurements, additional mockup reactor (MUR) tests were conducted to obtain the axial and circumferential power density variations. The maximum-to-minimum power variations without shielding measured in the mockup tests were approximately 2.5 for the axial power variation and approximately 1.6 for the circumferential variation. The axial measurements excluded 0.635 centimeter (0.25 in.) from each end of the fuel because the variations at the ends were greater than the acceptable values set for the experiment. More MUR tests were performed to reduce the maximum-to-minimum power variation by using a thermal neutron shield. A description of these tests is presented in reference 8. From these tests a shield configuration was selected for the experiment. Using the shield, the measured maximum-to-minimum fission power density for the axial variation was reduced from 2.5 to 1.5, and the circumferential variation was reduced from 1.6 to 1.3. The maximum-to-minimum variation for the axial profile did not include 0.635 centimeter (0.25 in.) from each end of the fuel. The neutron flux increases sharply in this region due to flux end peaking.

A review of the MUR test results indicated that any additional flux flattening could be accomplished only by a more complicated design. Therefore, the testing for power uniformity was terminated with the shield design selected for this experiment (fig. 8). The bottom portion of the shield extends 180° around the cluster of three fuel pins, the upper portion covers about 40°. Additional neutron end shielding in the form of a disk was located at the bottom end of each fuel pin. The main neutron shield was attached to an aluminum shroud, which housed the fuel pins in the reactor test hole, protected the fuel pins during handling, and was designed to channel the primary coolant water flow uniformly over the capsules. A photograph of the shroud with attached neutron shield is shown in figure 9.

The power density data obtained from the mockup tests were then used in the heat-transfer calculations to determine the size of the helium heat-transfer gas gaps between the fuel pin and capsule.

HEAT TRANSFER

A fission power-density variation existed in the radial, circumferential, and axial directions of the fuel. (This was discussed in the section on Test Facility and Environment.) The axial variation was time dependent and changed during a reactor cycle as shown in figure 10. Consequently, the combination of a set of fixed heat-transfer gap dimensions and a changing axial power profile produced a clad temperature profile that also varied with time. The purpose of the heat-transfer calculations was to determine a compromise set of gap dimensions in order to achieve the minimum variation in clad temperature.

A computer code, Steady State Heat Transfer Program (STHTP) (ref. 9), was used to calculate the helium gap dimensions. This program was designed to handle three-dimensional steady-state heat-transfer cases involving internal heat generation, temperature-dependent thermal conductivity, radiation heat transfer, and constant film and contact coefficients. The calculations were simplified by using two-dimensional r, z calculations using average values of power density in the r, θ plane. This simplification did not significantly affect the magnitude of the fuel-pin clad temperature.

The relatively large heat loss at the ends associated with the short lengths made it necessary to vary the helium gap along the length. Increasing the helium gap at the ends of the fuel pin prevented the clad temperature at the ends from dropping too low.

To reduce machining costs, the capsule was fabricated with steps in the helium gap. Three steps were used. Additional steps did not significantly improve the temperature profiles. Figure 11 shows a schematic layout of the reference diameter pin with a minimum radial gap at the center of the fuel pin of 0.508 millimeter (0.020 in.). Figure 12 shows the calculated temperature profiles for a reference-diameter pin using fission power density profiles (fig. 10) for the beginning and end of reactor cycle. Gamma heating in both the fuel and clad was used in the calculations. The gamma heating in the fuel was about 15 percent of the total power generation in the fuel. The average temperature of the clad throughout the cycle was taken as the average of the beginning and end of reactor cycle profiles.

The corresponding calculated thermocouple "readings" for a reference-diameter fuel pin at the beginning and end of a reactor cycle condition are also shown in figure 12. However, this figure is plotted for the conditions of beginning of life and does not take into account fuel swelling. During irradiation the clearance gap decreased as the fuel swelled. Consequently, the end of life (EOL) fuel-center-thermocouple readings are reduced to maintain the same average clad temperature. Listed in table II are the temperature drops across the fuel, clearance gap, and clad for both the reference and half-diameter pins for the conditions of beginning of life (BOL) and end of life (EOL). BOL data assumed a 0.032-millimeter (1.25-mil) radial gap and EOL data assumed a 0.0064-millimeter (0.25-mil) radial gap.

The half-diameter radial fuel pin geometry was scaled down from the reference diameter pin by a factor of two. The fission power density in the half-diameter pin was about twice that of the reference diameter pin. Thus, the heat flux and the midplane helium gap dimensions were the same for both pin sizes.

RESULTS AND DISCUSSION

Postirradiation Examination

This report describes the design of fuel pin assemblies irradiated in the Plum Brook Reactor Facility and presents the operating history of three fuel pins irradiated for 8070 hours. The value of the fuel-pin irradiation experiment is determined from the postirradiation examination results and the utilization of these results to enhance the technology for space power reactors. Curtailment of NASA nuclear programs prevented a complete postirradiation examination. This section of the report covers the results of the fuel-pin examination relative to clad swelling, clad integrity, fuel swelling, fuel burnup, and fission gas release. However, in order to facilitate the postirradiation discussion a summary of the fuel pin operating conditions are shown in table III.

Fuel pellet dimensions and density measurements. - Preirradiation and postirradiation fuel pellet dimensional measurements were compared for the half-diameter fuel pins. These values are given in table IV. There were six fuel pellets in each fuel pin. The maximum values of the fuel pellet diameters and lengths are also given along with the average values. All of the fuel pellets in the reference diameter pin contained axial cracks, and all but the top pellet fell apart during fuel stack disassembly. As a result the measurements reported for the reference diameter fuel pellets are for the top pellet. No cracks were observed in the tungsten liner.

Density measurements were made before and after irradiation with a mercury pycnometer. The irradiated fuel density changes are listed in table IV. On the average the fuel densities decreased by approximately 2 percent compared with the preirradiation data.

Clad integrity. - The capsules and fuel pins were punctured to check for the presence of fission gas. In two of the three pins fission gas was found only after puncturing the fuel pin, indicating that the pins did not leak. In the third pin, a half-diameter pin, fission gas was found after puncturing the capsule indicating a fuel pin leak. The leak was located in the thermocouple well of the fuel pin end cap containing the closure weld. The results of a liquid bubble type of leak test indicated a leak emanating from the base of the thermocouple well viewed from the fuel side of the end cap. The thermocouple well is an integral part of the end cap. Two smaller leaks were also observed in the face of the end cap (side also facing the fuel) in the vicinity of the seal weld closure. The seal weld was also leak tested. No leaks were detected in the top of the seal weld (side not exposed to the fuel). A photomicrograph of a metallographic section taken through the top end cap (thermocouple well wall and closure weld with filler wire) is shown in figure 13. The approximate magnification is X40. Note the leak path from the fuel side of the closure hole through the wall of the thermocouple well.

The clad did not leak on any of the three fuel pins. All clad appeared bright and shiny when removed from the capsules. The clad measurements before and after irradiation showed that the clad diameters did not increase. In fact, the irradiated clad was about 0.0076 millimeter (0.3 mil) smaller in diameter. Part of this diameter decrease, approximately 0.0025 millimeter (0.1 mil), can be attributed to the final acid etch. No measurements were made after the final cleaning and anneal as per the cleaning specifications. The measurements showed that the clad did not increase in diameter; yet the fuel pellets increased up to a maximum of 0.056 millimeter (2.2 mils) or 0.7 percent. This paradox can be explained by the amount of clearance provided between the fuel and the clad. Assembly clearances were provided so that the tungsten liner and fuel pellets could be assembled into the fuel pin clad under dry box conditions. Thus at room temperature there was 0.114 millimeter (4.5 mils) of diametral clearance between fuel and clad on the reference-diameter pin and 0.089 millimeter (3.5 mils) of diametral clearance on the half-diameter pins. At temperature, the beginning of test diametral clearance between clad and fuel was about 0.0635 millimeter (2.5 mils) for each size fuel pin. Even using the maximum fuel pellet swelling measurements, the fuel was not in complete contact with the clad during the irradiation. Thus, it can be concluded that the clad was not strained during irradiation.

Clad ductility tests. - We concluded that during the irradiation the fuel did not strain the clad. However, it was felt that an evaluation should be made to determine the ductility of the clad after irradiation. The postirradiation clad ductility was evaluated from expansion tests conducted on ring samples taken from the clad.

Control tests were also conducted on unirradiated ring specimens. These ring specimens were from aged (2862 hr at 1038° C (1900° F) in a vacuum) T-111 cladding. The clad was cut into rings about 4.8 millimeters (3/16 in.) wide for evaluation. The cuts were made with a dry hacksaw blade in the hot cell atmosphere using a niobium - 1-percent-zirconium holding block to facilitate cutting the small ring samples. The saw cuts were filed and dry sanded to remove microcracks at the cut surfaces. The intent of the dry cutting and sanding was to keep moisture out of the T-111. The ring expansion or ductility test was run in an atmosphere of flowing dry argon. Strain was induced in the ring by forcing a hardened tapered steel plug into a stainless-steel body at a constant rate resulting in a radial strain in the ring of approximately 7.6×10^{-5} centimeter (3×10^{-5} in.) per second.

The expansion rings for both unirradiated-aged and irradiated samples were prepared and tested under the same conditions and at approximately the same time.

The test results are as follows: The half-diameter specimen (east pin with end cap leak) was evaluated first. The irradiated clad failed after an expansion of 1-percent. A second clad ring was vacuum heat treated 1 hour at 1038° C (1900° F). The ring expansion test was run, and the clad was found to be ductile as were the unirradiated rings; that is, no failure occurred at the maximum strain of 5 percent. It is believed that only hydrogen would be removed by a 1 hour vacuum heat treatment at 1038° C (1900° F). Furthermore, previously run out-of-pile tests indicated that this heat treatment would restore ductility to hydrogen contaminated T-111. Therefore, the embrittlement of the irradiated material is attributed to hydrogen.

A probable source of hydrogen is the n, p reaction with the nitrogen in the UN fuel according to the following equation from reference 10.

$$_{7}N^{14} + _{0}n^{1} \rightarrow _{6}C^{14} + _{1}H^{1}$$

Generated hydrogen can become trapped in the capsule by the nature of the method of testing. More specifically, the capsule design incorporated a stagnant gas gap for heat removal from the fuel pin to the capsule water-cooled walls, which were at 66° C

(150° F). The cold capsule wall along with nonflowing heat-transfer gas kept the hydrogen generated from escaping. A calculation was made for the irradiation time of 8070 hours, and assuming uniform distribution throughout the clad, the resultant amount of hydrogen would be approximately 3 ppm by weight. Tests run with 1 ppm of hydrogen caused embrittlement of T-111. Recognizing that possible hydrogen embrittlement may influence the test results, the ring expansion tests were continued on the remaining two pins (the reference-diameter pin and half-diameter west pin). The as-irradiated reference clad expanded 1.5 percent before failure; the half-diameter west clad expanded 3 percent before failure. Additional specimens from the clad were vacuum heat treated 1 hour at 1038° C (1900° F), and the expansion test was repeated. The ductility in these cases, however, were essentially unaffected by the vacuum heat treatment. The reference-diameter clad failed after 1.5 percent expansion; the half-diameter west pin failed after 4 percent expansion.

One further test was conducted on the clad from the reference-diameter pin. Another ring was out from the clad and vacuum heat treated for 1 hour at 1400° C (2550° F). This heat treatment restored the ductility (same as the unirradiated controls). There is conjecture that possibly oxygen contamination could have resulted in the loss of ductility. The vacuum heat treatment at 1038° C (1900° F) was not high enough to redistribute the oxides concentrated in the grain boundaries. The possible source of oxygen could have been instrumentation leads and insulation or from the vacuum heat-treat furnace (possible contamination of furnace). However, preliminary data from other pins being evaluated as part of this overall program (about 10 pins tested) indicate that without exception the clad ductility was restored to its unirradiated condition after a 1-hour heat treatment at 1038° C (1900° F).

In summary, the clad from the investigation of three fuel pins was brittle after 8070 hours of irradiation. One half-diameter clad had full ductility restored after a vacuum heat treatment of 1 hour at 1038° C (1900° F). A reference-diameter clad had complete ductility restored after a vacuum heat treatment of 1 hour at 1400° C (2550° F). There was not enough clad available to further evaluate the other half-diameter pin (4 percent expansion after 1 hr vacuum heat treatment at 1038° C (1900° F)). Further investigation was precluded by the termination of NASA nuclear work.

Neutron fluence. - Associated with the clad ductility tests was the fluence measurement incident on the fuel pins. The total fluence incident on the fuel capsules was determined by counting flux wires attached to the capsule holder and adjacent to each capsule. The wire used for the thermal flux monitors was 99 percent aluminum and 1 percent cobalt. Stainless steel was used for the fast flux monitors. The thermal fluence nvt was 3.1×10^{20} neutrons per square centimeter on the half-diameter size pins and 2.7×10^{20} neutrons per square centimeter on the reference diameter pin. The fast fluence, with energy greater than 0.1 MeV, was in the range of 1.4 to 1.5×10^{20} neutrons per square centimeter for both pin sizes.

Burnup analysis. - Cross-sectional wafers were cut from selected positions of the fuel for destructive analysis. Some wafers were used to determine the average axial variation in burnup. Samples were taken from some of the wafers with a microdrill for determination of circumferential burnup. The primary method used to determine burnup of the fuel was the change in the U^{236} to U^{235} ratio obtained from the mass spectrometer analysis. The burnup results are summarized in the following table:

Pin designation	Average burnup, percent		Maximum to minimum burnup ratio		Calculated fission gas release[a], percent
	Determined by mass spectroscopy	Calculated using center fuel temperature	Circumferential	Axial	
Half diameter, east	0.89	0.7[illegible]	1.22	1.13	0.05
Half diameter, west	.92	.780	1.30	1.25	.02
Reference diameter	.47	.468	1.15	1.47	.02

[a]Based on krypton-85.

The maximum to minimum variations in the axial and circumferential burnup values were in most cases less than 1.25. This was well within the acceptable values as stated in the section Test Facility and Environment.

Both the measured and calculated burnup values for the reference-diameter or controlled capsule were the same. There is, however, a difference of about 10 percent between the measured and calculated burnup values for the half-diameter pins. This can be attributed to several causes: the method used to calculate the burnup, changes in the hafnium shield effectiveness, and changes in the relative burnup between the pins. The method used to calculate the burnup was based on heat-transfer considerations using the measured fuel center thermocouple reading in the calculation. However, no correction was made for fuel swelling which was greater for the half diameter pins. This increased swelling decreased the gap (thermal resistance) between the fuel and clad. Thus, for the same flux, the half-diameter fuel center temperature decreased with time relative to the reference pin temperature. A burnup calculation based on this temperature, the value of which is given in the burnup tabulation, was lower than the measured value as expected. However, the calculational error, by not taking into account fuel swelling, accounts for only one half of the burnup difference.

Part of the remaining difference can be attributed to time changes in the hafnium shield effectiveness, which predominantly effects the behavior of the reference pin. For example, the capsules were designed so that at startup the power density in the half-diameter pins should be twice that of the reference pin. However, as the shield burns up, the half-diameter pins run cold relative to the controlled reference pin. Further, as the fuel burns up, the relative power density changes between the reference- and the half-diameter pins (the initial enrichments were 5.6 percent for the reference and 8.2 percent for the half diameter). As a result, the half diameter pin burnup is less than twice that of the reference pin.

Gamma scan. - Gamma scanning affords a rapid, nondestructive method of determining axial burnup variation in irradiated UN fuel pins. However, the data must be normalized to mass spectrometer burnup values in order to determine the absolute value of a gamma-scanned axial burnup profile. For example, if several mass spectrometer determinations are done on one fuel pin then the results should be applicable to other pins of the same composition.

Gamma scanning along the fuel-pin length for axial burnup determinations was conducted both with and without the T-111 clad. In the case of scanning without the cladding, the fuel pellets were removed from the clad, stacked on a rod and inserted into a chuck positioner located within the gamma scan facility. For both axial scans (with and without clad), fission product niobium-95 was used as the monitor.

The burnup value at a given axial location was determined within an error of 7 percent at the 2σ level when compared with the burnup value determined on the mass spectrometer.

Fission gas release. - Fission gas release values were calculated from the measured average burnup and the amount of krypton-85 activity detected in the gas sample taken at the initial puncture of gas containment. In the case where the pin had a leak in the thermocouple well, the fission gas was still contained in the capsule. The amount of fission gas released was small when its contribution to gas pressure buildup was considered.

Neutron Radiography

Neutron radiography is an important technique available for nondestructive examination of the fuel-pin capsule. The fuel capsule assemblies were radiographed before irradiation and at selected times based on in-pile operation. Typical neutron radiographs are shown in figure 14. As an example, neutron radiographs are used to nondestructively detect fuel-pin clad swelling by viewing the heat-transfer gaps between the fuel-pin outside diameter and the capsule inside diameter. Because of the arrangement of the three capsules in the capsule holder (see fig. 4), one radiograph was insufficient to view the gaps from all three capsules. Three radiographs were taken at three orientations to provide views of each capsule. Gas-gap dimensions were measured by scanning the neutron radiograph negative with a recording microdensitometer. The details of this technique are reported in reference 11.

Neutron radiographs were taken before irradiation, after 4910 hours, and at the completion of testing (8070 hr). No fuel-pin clad swelling was indicated from any of the radiographs, and these observations were later verified during the postirradiation examination. The only change indicated in the neutron radiographs was the downward shift of the fuel stack caused by deformation of the spherical spacers provided for accommodating axial fuel expansion.

Fuel Pin Temperature Variation

The three fuel pins were irradiated in the same reactor test hole with all three pins controlled by one vertical positioning device. This arrangement precluded independent neutron flux and temperature control of each fuel pin. We recognized and accepted the temperature variation in the test pins that resulted from factors over which we had no control. However, because of the importance of temperature we felt that it was necessary to know the clad temperature variation. Consequently, each fuel pin contained at least two fuel center thermocouples penetrating the fuel pin end caps and extending into the fueled region. In this manner we were cognizant of the temperature variations that existed. The axial distance between the thermocouple junctions located in the fuel was about 3.8 centimeters (1.5 in.). The best that could be accomplished with this arrangement was to maintain two of the three fuel pins within the desired temperature band of 80 K (144° F). Factors that precluded uniform temperature on all three test pins can be summarized as follows:

(1) Reactor power varies from cycle to cycle. As an example, a thermal neutron shield - the purpose of which is to provide uniform power distribution in the fuel pin - was effective only if the mode of reactor power did not change from cycle to cycle. In this experiment the reactor power did change.

(2) Changes in core loading can locally influence the flux shape. Also periodic cadmium shim rod reloading affects the flux level.

(3) One of the most significant, yet uncontrollable, perturbations was the influence of other experiments. Temperature changes as large as 60 K (108° F) were observed on individual pins resulting from the insertion or removal of other experiments.

(4) Fission gas leakage in one capsule decreases the thermal conductivity of the helium heat-transfer gas. Thus, the fuel pin with a leak operates hotter relative to the nonleaking pins.

Figure 15 shows the six thermocouple plots for the three fuel pins that comprised the capsule assembly. Along the abscissa, both the reactor cycle numbers and the cumulative irradiation time (in hr) are shown. The temperature plots for the reference-diameter fuel pin are shown at the bottom of the figure. The temperature plots for the smaller pins are labeled half-diameter east, and half-diameter west. The east and west designation refers to the relative position of these fuel pins with respect to the reactor core.

The regions of most interest are the 80 K (144° F) desired temperature bands for each fuel pin shown by the shaded area. The absolute level of this band for the reference-diameter fuel pin is different from the level of the other two bands for the following reason: The reference- and half-diameter pins have different power densities, fuel thicknesses, and assembly clearances, thereby resulting in different fuel center temperature readings for the same desired clad temperature.

The hottest thermocouple was the one used for controlling the temperature of the capsule assemblies.

Listed at the top of figure 15 are factors that resulted in temperature variations between the fuel pins. The temperature variations of the fuel pins in the third reactor cycle (number 110) were considered to be as good as we could achieve. Figure 16 shows the temperature plots of the six fuel center thermocouples as well as the gas thermocouple plot. The gas thermocouple is a low-temperature backup couple located midway between the fuel pin and capsule end caps. The temperature plots were encouraging, and no further shield or rotational changes were contemplated. However, early in next reactor cycle, the hafnium power-flattening shield became dislodged from the capsule holder shroud. A large power increase occurred in the reference-diameter capsule - approximately a factor of two at the location of the bottom fuel center thermocouple and approximately a factor of 1.2 at the location of the top center thermocouple. The calculated temperature at the bottom fuel center thermocouple resulting from the shield dropping off was approximately 1920 K (2996° F (above the melting point of type K thermocouples). The reference-diameter capsule four-wire bottom fuel center thermocouple failed during this temperature transient which lasted about 2 minutes. The experiment was automatically withdrawn during this temperature excursion. After a shield-attachment redesign was incorporated, the experiment was again irradiated. The bottom reference diameter fuel pin temperature was now determined from a relationship established between the bottom center thermocouple and gas thermocouple before the shield fell off. This calibration is discussed in appendix B. A significant observation from this irradiation is the large axial temperature variation that occurs over a distance of 3.8 centimeters (1.5 in.) - the distance separating the fuel center thermocouples. In many other experiments large temperature variations are not observed because the fuel pins usually do not contain thermocouples at different axial positions. In this test a determined effort was made to achieve uniform temperature on three fuel pins in the same test hole; yet, there were uncontrollable factors that precluded continuous operation throughout the test duration at uniform temperature. The fact that a test pin operated at off-design conditions did not negate the experiment. The off-design results can also be used to evaluate fuel-pin performance.

Thermocouple Performance and Reliability

Eight fuel center thermocouples were located in the capsules. Six operated for the duration of the test. Two were subjected to temperatures above their melting point when a neutron shield fell off a capsule resulting in a failed thermocouple. We feel that the success of the thermocouples was due to the fabrication techniques used to form the ungrounded 5.08-centimeter (2-in.) long high-temperature zone and hot junction. (See fig. 17.) Extreme care was taken to insure a loose fit between the wires, sheath, and ceramic - both axially and radially. Minimization of moisture content was also stressed. Thermocouple assembly and fabrication procedures are discussed in appendix A.

The two bottom fuel center thermocouples on the reference pin were lost because of an overtemperature of the fuel pin when the thermal neutron shield surrounding the capsule came off. These thermocouples have operated for 8070 hours between a range of 1150 to 1400 K (1610° to 2060° F) (see table III for hours of operation at various temperatures) and have been subjected to 82 thermal cycles. A thermal cycle has been defined as any decrease in temperature in excess of 300° C (540° F). Most of these cycles result from normal withdrawal and insertion during reactor operation required for reactor refueling. However, some of these cycles are a result of a reactor scram or fast setback where the thermal cycles are severe. The resistances of the thermocouples both wire to wire and wire to sheath were periodically checked over the duration of the irradiation. No appreciable change in the resistances between the wires and sheath occurred after the initial change. The initial change was attributed to moisture absorption in the connector, which occurred after the experiment was placed in the reactor tank (see appendix A - THERMOCOUPLE PREPARATION).

Calibration of Fuel Center Thermocouples

Postirradiation calibration of two fuel center thermocouples from the half-diameter fuel pin (designated as the west pin) was conducted. In brief, the results showed that the maximum temperature decalibration was a 27° C (49° F) lower thermocouple temperature indication at 1000° C (1832° F). The lead to sheath (insulation) resistances of the irradiated couples were as high as the unirradiated control. It is felt that the decalibration was due to compositional changes in the thermoelements. The estimated thermal and fast ($E > 0.1$ MeV) neutron fluences in the temperature gradient region of the thermocouples are 3.1×10^{20} and 1.4×10^{20} neutrons per square centimeter, respectively.

CONCLUSIONS

The following concluding remarks are divided into two categories: conclusions based on the postirradiation examination and conclusions based on the design, fabrication, and 8070-hour operation of three fuel pins.

Postirradiation Remarks

1. No T-111 clad failures were detected in the three fuel pins tested. A thermocouple well located in the end cap of one half-diameter pin leaked. No other leaks were detected and no increase in clad diameter was measured.

2. Expansion ring ductility tests conducted on irradiated clad specimens failed at about 1 percent strain, indicating an embrittlement of the clad. However, irradiated clad rings that were vacuum heat treated for 1 hour at 1038° C (1900° F) were as ductile as unirradiated control specimens. The clad embrittlement is not believed to result from irradiation damage.

3. The average increase in diameter of the fuel pellets was less than 0.5 percent, and the average increase in fuel pellet length was slightly greater than 0.5 percent. The fuel density decrease was about 2 percent. The fuel swelling was not sufficient to close the clearance gap between fuel and clad; consequently, the fuel did not strain the clad.

4. The fuel burnup as determined by the mass spectrometer method was in good agreement with the calculated burnup based on heat-transfer considerations. The amount of fission gas released was less than 0.05 percent.

5. Maximum to minimum variations in axial or circumferential burnup values were in no case greater than 50 percent and in most cases less than 25 percent.

6. Axial gamma scanning of fuel pins appears to be an economical nondestructive method by which to supplement mass spectrometer burnup measurements.

Design, Fabrication and Operation of Fuel Pins

1. A simple fuel-pin - capsule design combined with a reasonable amount of quality assurance provided a reliable test article for inpile testing.

2. Three fuel pins, of two different sizes and burnup rates, were operated near the designed clad temperature bands even though attached to the same capsule holder positioner. To achieve this condition of operation within or near a specified temperature band, the following information was needed:

a. Mockup reactor test data to determine fuel pin flux profiles, gamma heating values and flux shield requirements.

b. Heat-transfer calculations to determine gap sizing for heat removal from the capsule to provide uniform clad temperature.

c. Quality assurance on all components from the fuel pins and capsules.

3. Standard type K thermocouples, with selected modifications to the high-temperature zone, provided dependable operational life (longer than 8000 hr at temperatures up to 1400 K (2060° F).

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, June 22, 1973,
503-25.

APPENDIX A

THERMOCOUPLE PREPARATION

The total length of the thermocouples from junction to reactor penetration is 8.9 meters (26 ft), however, less than 5.08 centimeters (2 in.) in the region of the junction, is exposed to any appreciable temperature level. Because of this and for economy reasons, the leads were fabricated using conventional fabrication procedures with minor deviations to minimize the possibility of contamination. However, extreme care was taken in the fabrication of the 5.08-centimeter (2-in.) high-temperature zone. A brief description of the thermocouple fabrication procedures follows, a more detailed description of the hot-junction fabrication may be obtained in reference 4. The 8.9-meter (26-ft) stainless steel sheathed thermocouple leads were fabricated using the highest purity crushable alumina ceramic (99.7 percent min. purity) commercially available. This ceramic was baked out at a minimum temperature of 427° C (800° F) for at least 4 hours prior to assembly. To minimize moisture and air content, the leads were backfilled with dry, high-purity argon, and the ends sealed before drawing. The high-temperature zone of the thermocouple was fabricated with adequate clearances between the wires, the sheath, and the ceramic both axially and radially to minimize stresses on the wires due to relative thermal expansion. Approximately 5.08 centimeters (2 in.) of the sheathed lead was stripped back and a loose fitting, hard-fired, high-purity alumina or beryllia (depending on availability) insulation was placed over the wires before forming the junction. All work was performed in a humidity controlled clean room and followed by a 760° C (1400° F) vacuum bakeout of the junction including a minimum of 0.61 meter (2 ft) of the leads. The furnace was back filled with high-purity dry argon before removal of the leads. The hot junction of the thermocouple (fig. 17) was tantalum sheathed because of the operating temperature of the thermocouple (1400 K (2060° F)) and the compatibility problems between the stainless steel and the T-111 alloy at these temperatures. A stainless-steel transition section was electron beam welded to the tantalum sheath covering the hot junction on one end and to the stainless-steel sheath of the thermocouple lead on the other. The diameter of the weld joint between the tantalum tube and stainless-steel transition was limited to 3.175 millimeters (1/8 in.) outside diameter because of thermal expansion differences. The transition section was then electron beam welded to the stainless-steel capsule and provided the seal between the thermocouple lead and the capsule.

All the thermocouple leads from the three capsules clustered in the capsule holder within a test hole were brought out of the test hole through a stainless steel flex hose. The other end of the flex hose had an electrical connector welded to it. The sheathed thermocouples contained inside the flex hose terminated with the thermocouple wires soldered to the pins of this connector. The connector was then potted with an epoxy compound to prevent the exposed ends of the thermocouple leads from absorbing moisture and to provide a seal to prevent reactor primary water from leaking out of the reactor tank through the pins of the connector. The connector end of flex hose, which fits through a penetration on the top of the reactor tank, was sealed by an O-ring. The epoxy used on the connector was not completely impervious to moisture, and the resistance between the leads to ground dropped from the megohm range to the kilohm range. However, the resistance did not drop low enough to appreciably shunt the electromotive force (EMF) generated by the thermocouple. This problem was remedied on later capsules by bringing the leads out of the reactor tank through compression type insulating fittings. All resistances remained in the megohm range using this method.

APPENDIX B

CORRELATION BETWEEN BOTTOM FUEL CENTER AND GAS THERMOCOUPLE

The neutron shield surrounding the capsule assembly came off during the fourth irradiation cycle. This resulted in an overtemperature on all the fuel pins and a melt down of both bottom fuel center thermocouples on the reference diameter capsule (this would be expected since the shield has the greatest affect on the bottom half of the reference diameter pin). The reference diameter capsule had a bottom gas thermocouple, the junction of which was located approximately midway between the fuel pin and capsule end caps (see fig. 1). The purpose of this gas thermocouple was to serve as a backup in the event the fuel center thermocouples did not last for the full duration of the test. A correlation had been established (before failure) between the gas and bottom fuel center thermocouples. This correlation was established in the following manner.

A thermal neutron shield change was made between the first and second cycles (reactor cycles 108 and 109), and a rotational change between the second and third cycles of irradiation (reactor cycles 109 and 110). No changes were made between the third and fourth cycles (reactor cycles 110 and 111); so reproducibility could be verified between two cycles with no changes. The results are shown in figure 18. As can be seen, the relationship between the gas and fuel center thermocouple is dependent on reactor operating power and shield design. The maximum deviation of the data from the lines shown in figure 18 is ±20 K (±36^{o} F). As can be seen, the difference between the two thermal neutron shields at the same reactor power causes a 43 K (77^{o} F) shift in the correlation. Further, a change in reactor power from 60 to 50 megawatts causes an additional 20 K (36^{o} F) shift in the correlation. This is probably due to gamma heating differences. The gas thermocouple sensitivity is such that, for approximately every 1 K (1.8^{o} F) change in gas thermocouple temperature, there is about a 3 K (5.4^{o} F) change in the bottom fuel center thermocouple temperature. There were no shield or rotational changes between the third and fourth cycles (reactor cycles 110 and 111), and good reproducibility was obtained from the available data.

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TABLE I. - FUEL PIN DIMENSIONS AND OPERATING CONDITIONS

	Pin designation	
	Reference diameter	Half diameter
Number of pins	1	2
Pin diameter, cm; in.	1.822; 0.717	0.914; 0.360
Pin tungsten liner thickness, mm; mil	0.127; 5	0.076; 3
Clad thickness, cm; in.	0.1016; 0.040	0.0508; 0.020
Nominal cold diametral clearance[a] between clad and fuel, mm; mil	0.114; 4.5	0.089; 3.5
Relative fuel burnup rate	[b]1	2
Fission power density[c], W/cm^3	141	288
Nominal fuel outside diameter, cm; in.	1.580; 0.622	0.788; 0.310
Nominal fuel inside diameter, cm; in.	0.511; 0.201	0.279; 0.110

[a]At temperature, the beginning of test diametral clearance was approximately 0.0635 mm (2.5 mils) for each size fuel pin.
[b]Nominal burnup rate; corresponds to $5 \cdot 10^{12}$ (fissions/sec)/cm^3
[c]Fission power density is based on 176 MeV/fission in the Plum Brook Reactor.

TABLE II. - TEMPERATURE DROP ACROSS FUEL PIN FOR BEGINNING AND END OF LIFE CONDITIONS

Pin designation	Time of test (a)	Fuel temperature drop °C	Pellet-clad clearance gap temperature drop °C	Clad temperature drop °C	Total temperature drop	
					°C	°F
Reference diameter	BOL	70	44	14	128	230
	EOL	70	10	14	94	169
Half diameter	BOL	32	40	7	79	142
	EOL	32	8	7	47	84

[a]BOL indicates beginning of life; EOL, end of life.

TABLE III. - SUMMARY OF FUEL PIN OPERATING CONDITIONS

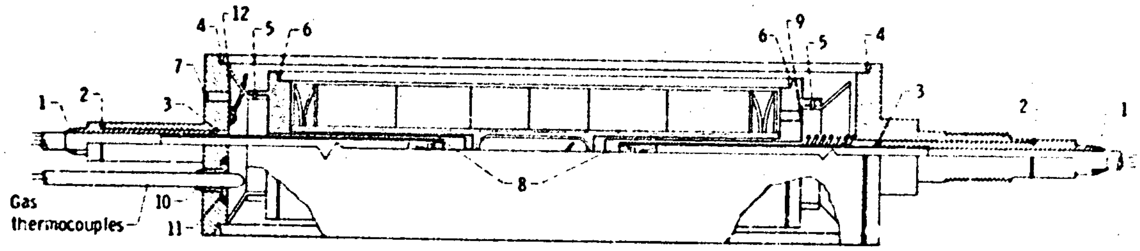
	Fuel pin designation		
	Half-diameter pin, east	Half-diameter pin, west	Reference diameter pin
Fuel centerline average temperature at top thermocouple, K[a]	1271	1260	1364
Fuel centerline average temperature at bottom thermocouple, K[a]	1242	1247	1370
Fuel centerline average temperature fuel pin, K[a]	1257	1254	1367[b]
Total hours of irradiation	8070	8070	8070
Burnup[c], percent	0.785	0.780	0.468
Burnup[d], fissions/cm^3	$2.59 \cdot 10^{20}$	$2.57 \cdot 10^{20}$	$1.54 \cdot 10^{20}$
Thermal excursions	82	82	82
Hours of irradiation at average fuel pin centerline temperatures between -			
1350 and 1425 K	108	331	6579
1250 and 1350 K	4201	3435	1491
1150 and 1250 K	3683	4178	----
<1150 K	78	126	----

[a]Average clad temperature at beginning of life (BOL) and end of life (EOL) is 79 and 47 K lower, respectively, than average fuel centerline temperatures for half-diameter fuel pins and 128 and 94 K lower for reference-diameter fuel pin.
[b]Calculated after first 937 hr, based on gas thermocouple calibration (see appendix B).
[c]Calculated, based on MUR measurements and heat-transfer calculations.
[d]Calculated based on average fuel center temperatures.

TABLE IV. - FUEL PELLET MEASUREMENTS

	Pin designation		
	Half-diameter, east	Half-diameter, west	Reference diameter (a)
Average change in diameter, ΔD, mm; mil	0.036; 1.4	0.030; 1.2	0.015; 0.6
Change in fuel pellet diameter, $\Delta D/D$, percent	0.445	0.387	0.1
Average change in length, ΔL, mm; mil	0.051; 2.0	0.043; 1.7	0.025; 1.0
Change in length, $\Delta L/L$, percent	0.525	0.456	0.3
Maximum change in diameter, ΔD_{max}, mm; mil	0.056; 2.2	0.043; 1.7	---------
Maximum change in diameter, percent	0.711	0.55	---------
Average decrease in density, $\Delta\rho/\rho$, percent	2.1	1.9	2.0

[a]Measurements taken on one pellet only; others were cracked.



1 Thermocouple sheath to thermocouple well (EB weld)
2 Thermocouple well to capsule end cap (EB weld)
3 Thermocouple well, stainless steel to tantalum transition (EB weld)
4 Capsule end cap to capsule tube (EB weld)
5 Fuel pin spacer to fuel pin (EB tack weld)
6 Fuel pin end cap to fuel pin tube (EB weld)
7 Capsule final closure (GTA weld)
8 Thermocouple well, tantalum plug to tantalum tubing (EB weld)
9 Fuel pin final closure (GTA weld)
*10 Gas thermocouple sheath to adaptor (EB weld)
*11 Gas thermocouple adaptor to capsule end cap (EB weld)
12 Positioning wire to capsule end cap (GTA weld)

Figure 1. - Weld schematic for capsule assemblies. (Asterisk denotes reference-diameter capsule only.)

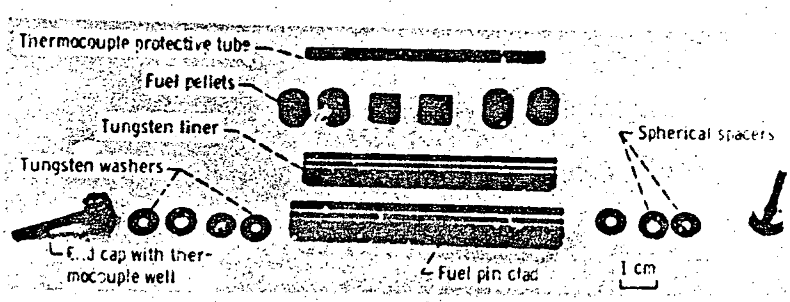


Figure 2. - Fuel pin parts for half-diameter pin.

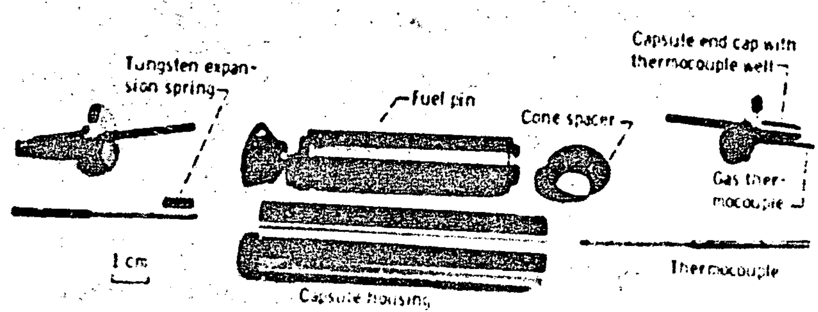


Figure 3. - Reference-diameter capsule ready for final assembly.

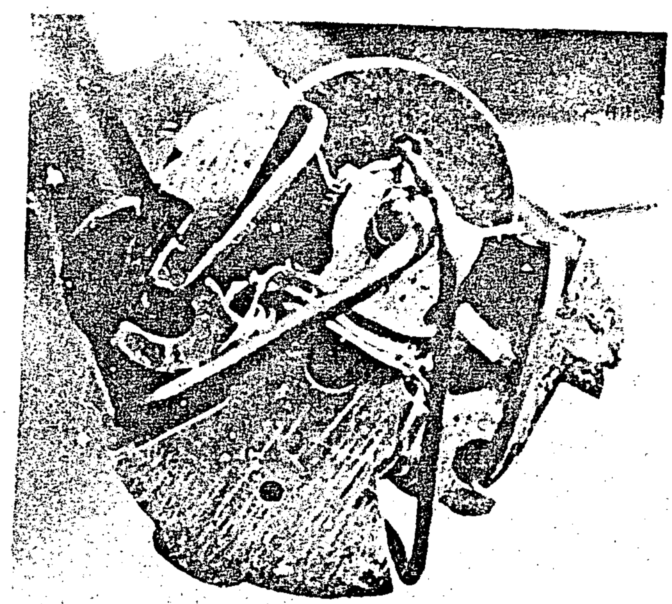


Figure 4. - Three capsules in capsule holder assembly.

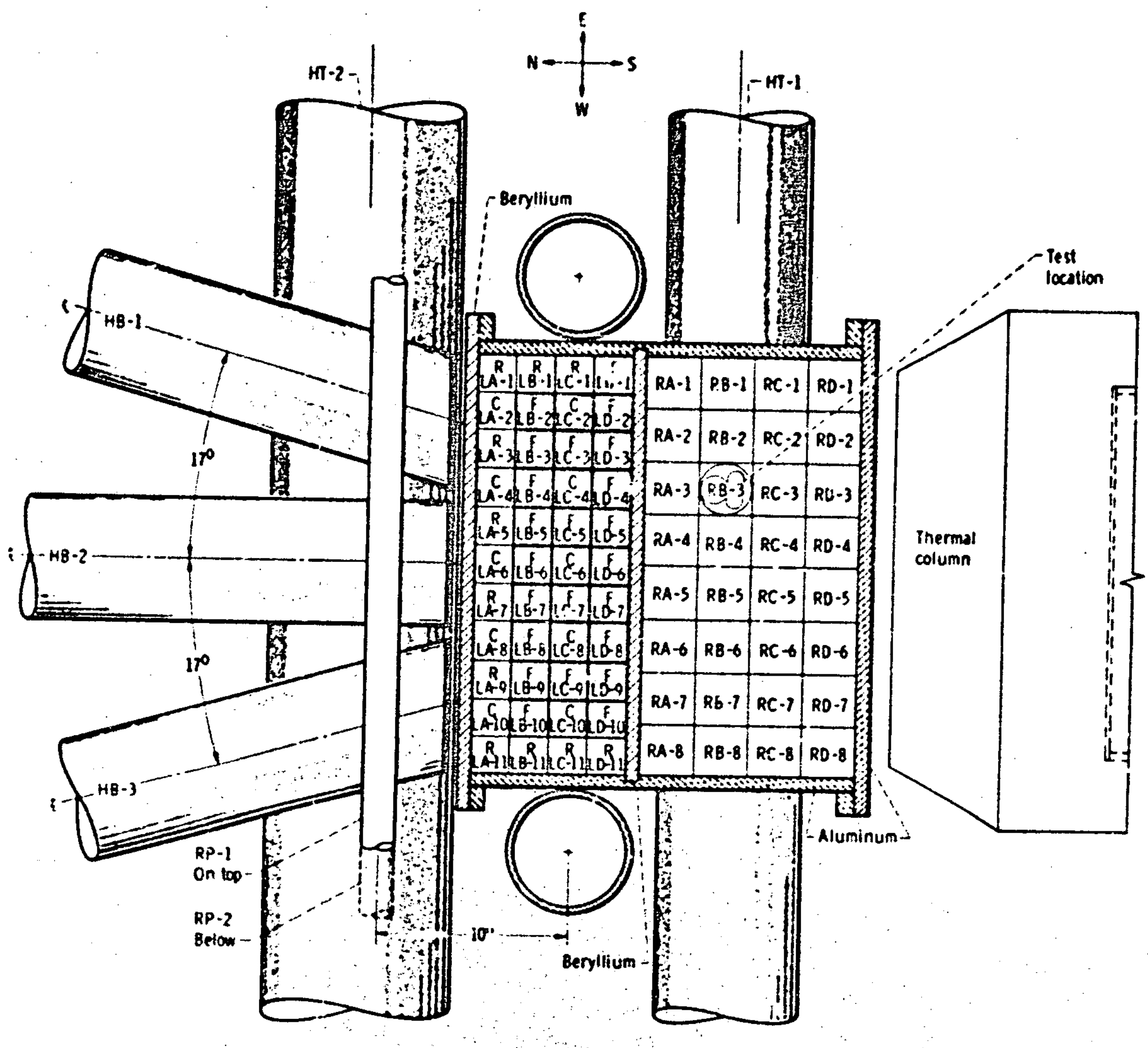


Figure 5. - Schematic horizontal section of Plum Brook reactor facility.

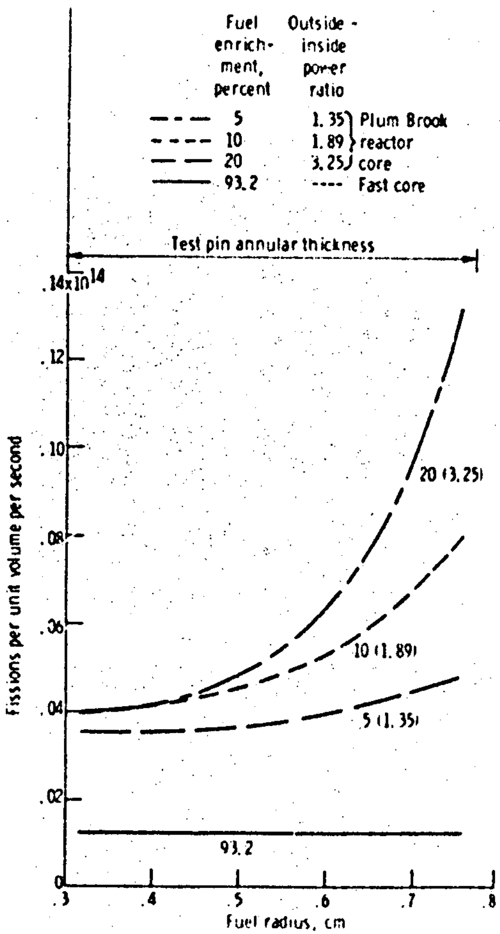


Figure 6. - Absolute fission rates for fast core and PBR core region.

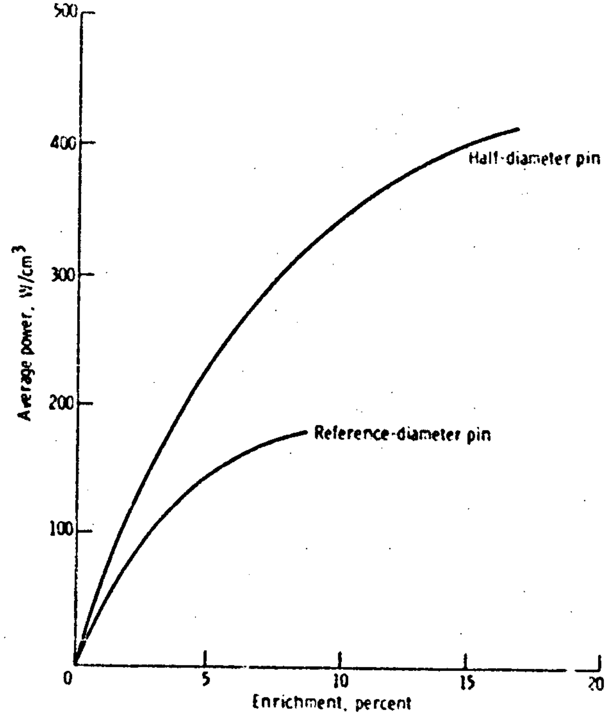


Figure 7. - Power as function of enrichment for reference-diameter and half-diameter fuel pins.

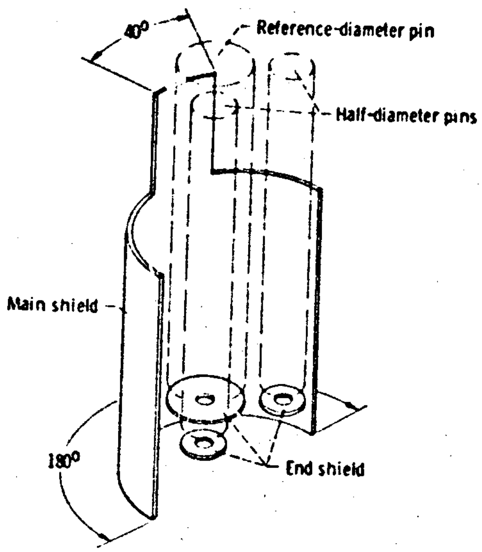
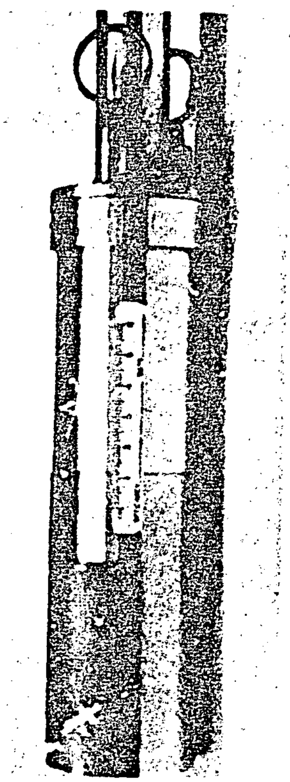


Figure 8. - Neutron shield.



igure 9. - Capsule shroud with attached neutron shield.

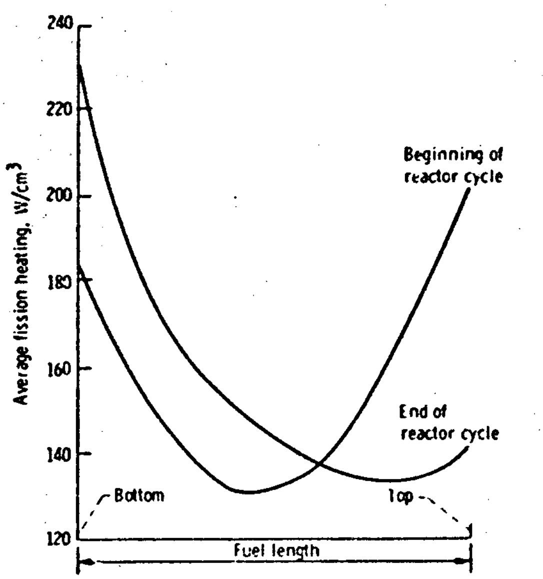


Figure 10. - Typical variation in average axial fission heating for reference-diameter pin at beginning and end of reactor cycle.

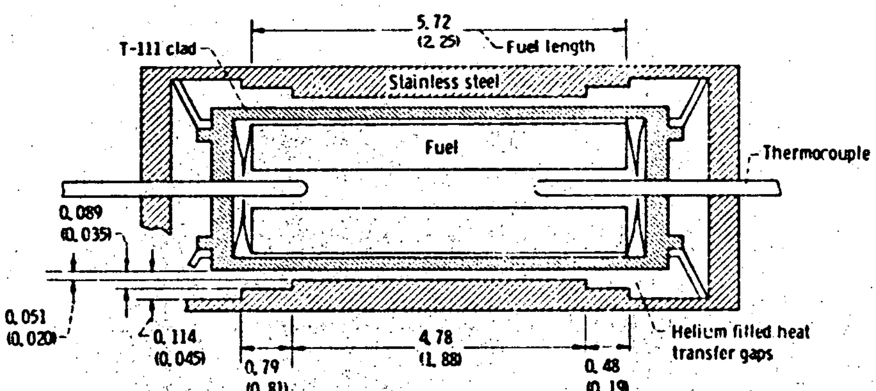


Figure 11. - Schematic of reference-diameter fuel pin. Helium gap sized for an average T-111 clad temperature of 982° C (1800° F). (All dimensions are in cm (in.).)

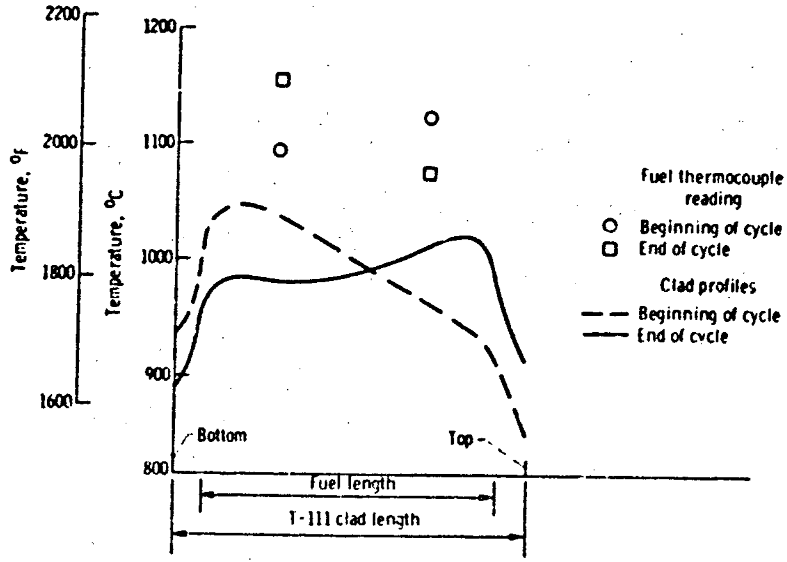
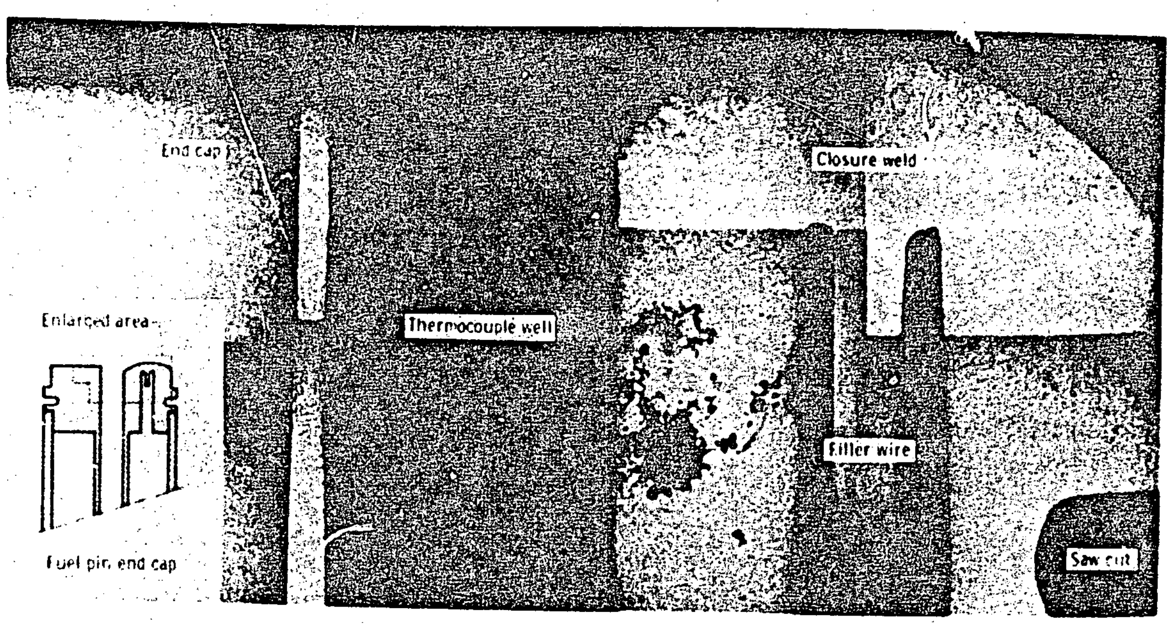
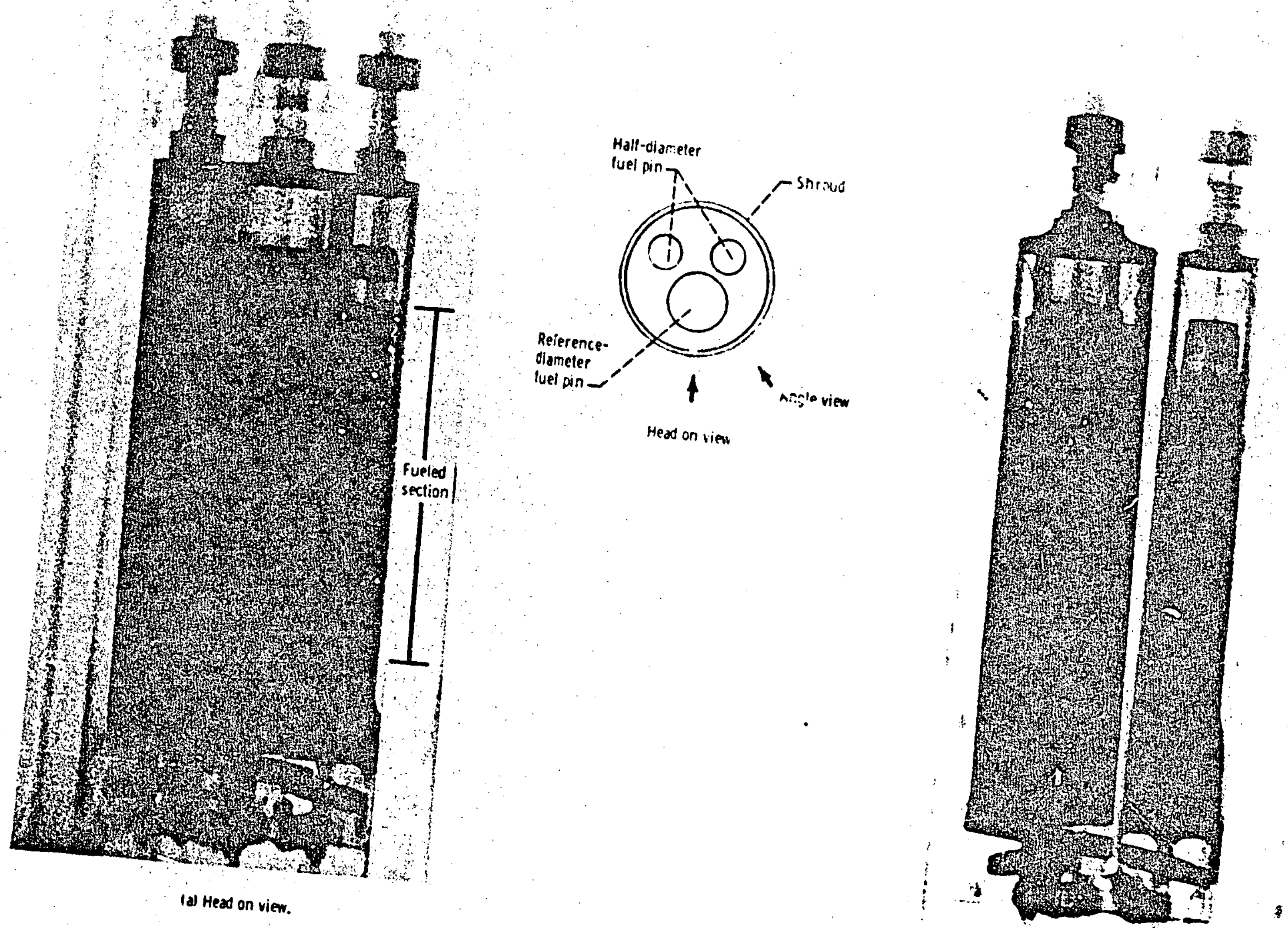


Figure 12. - Typical calculated clad temperature profiles of reference-diameter pin with two fuel thermocouple readings for beginning and end of reactor cycle.



CD-11435-15

Figure 13. -Photomicrograph of fuel pin end cap showing leak path between thermocouple well and closure hole. X40.



(a) Head on view.

(b) Angle view showing reference-diameter fuel pin and one half-diameter fuel pin.

Figure 14. - Neutron radiographs of irradiated fuel pin test assembly. Fueled section length, 5.715 centimeters (2.25 in.).

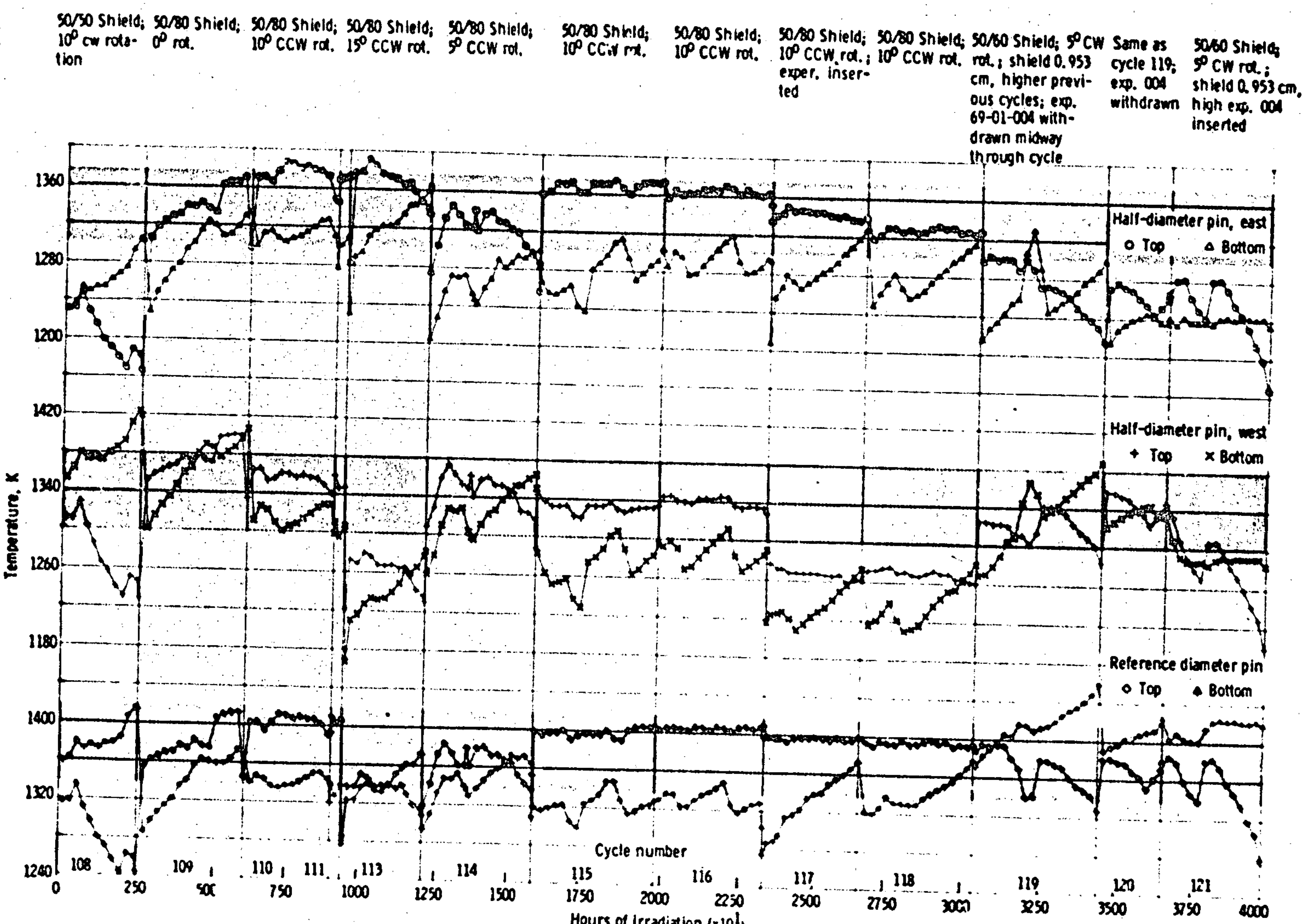


Figure 15. - Thermocouple reading for reference-diameter fuel pin and two half-diameter fuel pins.

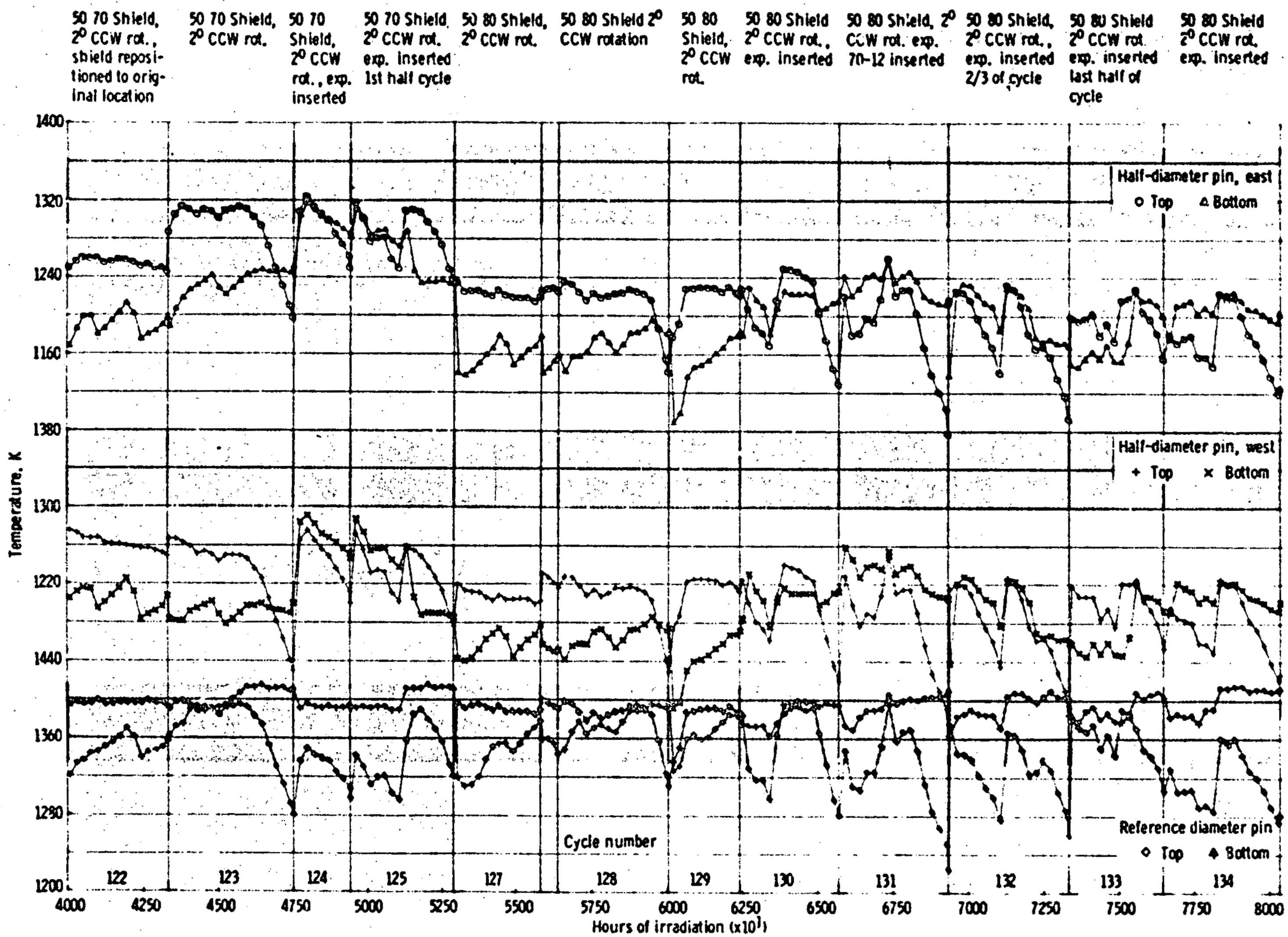


Figure 15. - Concluded.

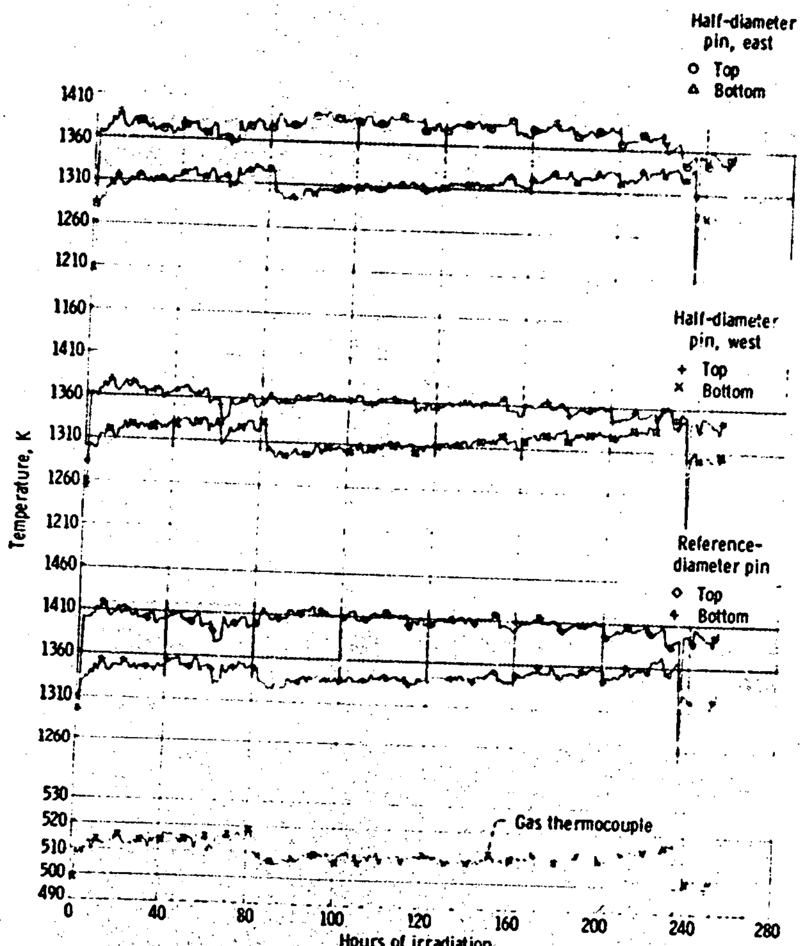


Figure 16. - Thermocouple readings for reactor cycle 110.

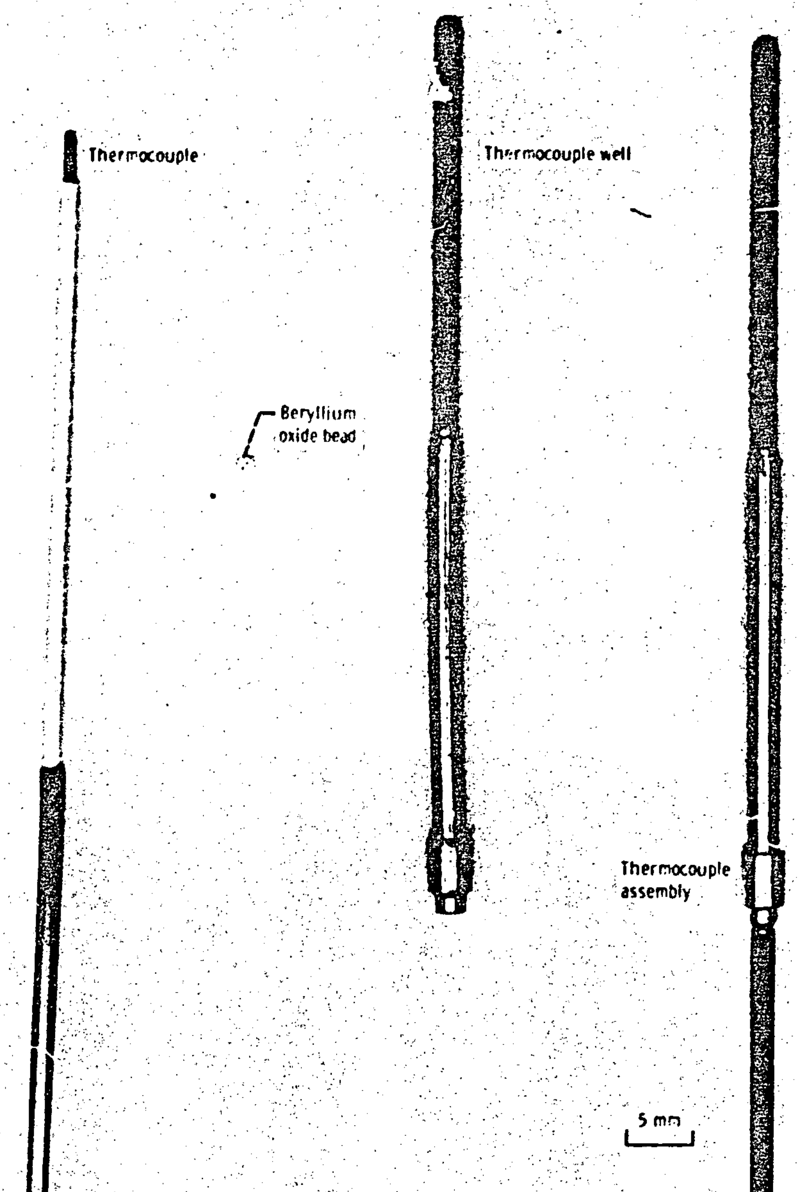


Figure 17. - Fuel center thermocouple assembly.

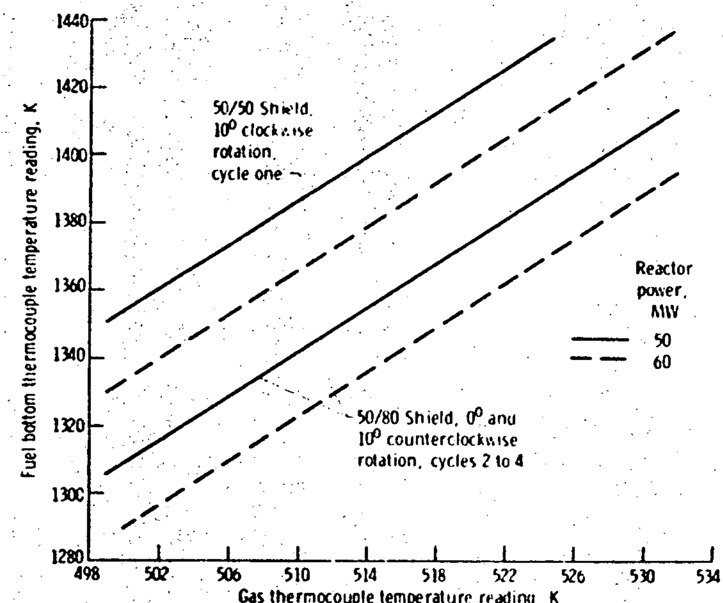


Figure 18. - Correlation between fuel center and gas thermocouple for reference-diameter fuel pin.

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