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UO_2 FUEL ROD OPERATION WITH GROSS CENTRAL MELTING

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October 31, 1963

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FOREWORD

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The work described in this report represents the Joint U. S. - EURATOM effort which is in keeping with the spirit of cooperation in contributing to the common good by the sharing of scientific and technical information and minimizing the duplication of effort by the limited pool of technical talent available in Western Europe and the United States.

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ABSTRACT

A fuel assembly (EPT-12C) containing UO₂ pellets has been irradiated at extreme thermal performance conditions sufficient to produce gross central melting. Based on loop calorimetry, the peak surface heat flux at startup averaged over the four rods in the assembly was

$$1.3 \times 10^6 \text{ Btu/h-ft}^2 \left(\int_{0^\circ\text{C}}^{T_{\text{CL}}} k dT = 188 \text{ W/cm} \right).$$

The calculated peak surface heat flux of the maximum power rod was

$$1.4 \times 10^6 \text{ Btu/h-ft}^2 \left(\int_{0^\circ\text{C}}^{T_{\text{CL}}} k dT = 200 \text{ W/cm} \right).$$

Analysis indicates more than 70% of the UO₂ cross section was melted, at the peak heat flux location during operation. The thickness of the remaining, unmelted UO₂ rim adjacent to the cladding was approximately 40 mils (0.1 cm). During its initial 55 days of operation, the assembly attained an average burnup of 5300 MWD/T_e (1.4×10^{20} fissions/cm³) and sustained 22 significant power cycles without difficulty. An interim examination of the fuel assembly at the 5300 MWD/T_e (1.4×10^{20} fissions/cm³) burnup level showed the fuel rods to be visually and dimensionally identical to the pre-irradiation conditions.

The successful operation of the EPT-12 fuel assembly is the culmination of the results obtained from an earlier series of fuel assembly irradiations designed to investigate the feasibility and limitations in operating with UO₂ central melting. During these irradiations, it was found that:

1. Severe swelling of the cladding occurred when using long unsegmented fuel rods which contain solid UO₂ pellets, and
2. The basic cause of swelling was the volume increase of UO₂ during melting.

This limitation is also believed to have caused the failures reported by other experimenters during their attempts to operate with UO_2 central melting. Recognition of this limitation resulted in a change to hollow UO_2 fuel pellets to accommodate the UO_2 volume expansion during melting and to eliminate rod swelling. The results of subsequent irradiation experiments and the satisfactory operation of this final, extreme thermal performance fuel assembly have demonstrated that the interpretation of the clad swelling phenomenon was correct. The fabrication parameters, operating conditions, and clad swelling results obtained from all the pellet fuel assembly irradiations are shown in tabular form. The irradiation experience leading to the successful operation of the EPT-12C fuel assembly is described.

During the startup that followed the interim examination at $5300 \text{ MWD}/T_e$ (1.4×10^{20} fissions/ cm^3) burnup, assembly EPT-12C sustained both clad swelling and an actual fuel rod failure. However, the investigation has since shown that the failure was probably induced by too rapid an increase in the amount of central melting during reactor startup and, therefore, could be prevented by programmed startup procedures. On this basis, two rods from the assembly have now been replaced, and it is planned to continue the irradiation of the rebuilt assembly to a burnup of $20,000 \text{ MWD}/T_e$ (5.4×10^{20} fissions/ cm^3). This extended irradiation should identify any other time-dependent failure mechanism (unique to operation with central melting) that might limit life. An almost identical exploratory program is being performed to determine the feasibility of operation with central melting in compacted UO_2 powder fuel.

EURATOM PRESSURIZED WATER LOOP FUEL ASSEMBLY IRRADIATIONS

ASSEMBLY DESIGNATION

	<u>EPT-6*</u>	<u>EPT-8</u>	<u>EPT-10</u>	<u>EPT-COM</u>	<u>EPT-12C</u>
Fuel Enrichment, %	1.5 (4 rods)	2.2 (4 rods)	3.0 (4 rods)	1.5 (1 rod) 2.2 (1 rod) 3.0 (1 rod) 3.8 (1 rod)	3.8 (4 rods)
Fuel Type, Pellet	Solid	Solid	Solid	Hollow	Hollow
Fuel Column Length, inches	34 ± 1/4	34 ± 1/4	31 ± 1/4	Variable	29-1/2 ± 1/4
Fuel Rod Peak Surface Heat Flux (assembly av) Btu/h-ft ²	645,000	895,000	1,010,000	1,190,000 (3.8% rod)	1,320,000
Assembly Thermal Power, kW	200	260	310	293	360
Average Burnup, MWD/T _e	1,180	1,442	1,182	1,564 (4 rod av)	5,290
Irradiation Period	4/23/62- 5/19/62	6/1/62- 6/24/62	7/3/62- 7/19/62	9/15/62; 11/26/62- 12/16/62	1/29/63- 3/31/63
Number of Fuel Rods Exhibiting Clad Swelling	3	4	4	1	0
Maximum Fuel Rod Diameter Increase (mils)	47	53	74	13 (3.0% rod)	0

*The term EPT designates EURATOM Pellet Fuel Assembly; the numbers 6, 8, 10, and 12 designate the assembly nominal peak heat flux during irradiation × 10⁻⁵

NOMENCLATURE

T	=	Temperature
T_m	=	Melting Point Temperature
T_C	=	Centerline Temperature
T_s	=	Surface Temperature
k	=	Thermal Conductivity
σ_t	=	Tangential Stress
r	=	Mean Clad Radius
P	=	Internal Pressure
t	=	Clad Thickness.

I. INTRODUCTION

Current design practice for UO_2 fuel rods in commercial power reactors restricts the UO_2 central temperature to below the melting point [$\sim 5000^\circ\text{F}$ (2800°C)] at the maximum reactor power condition. The experimental basis for this restriction has been reported by Eichenberg, et al.⁽¹⁾ The majority of the irradiation experiments consisted of relatively small diameter, [0.375 inch, (0.954 cm)] short, [~ 4 to 5 inches (10 to 12 cm)] capsules, and were not specifically intended as an investigation of central melting. The experimental results were extremely variable but, in general, the upper thermal performance level specimens either failed or exhibited gross cladding deformations. (One of the results of the program to be described was to demonstrate that short capsules do not accurately reflect the consequences of operation with central melting in full-size fuel rods.) Only one experiment with an appreciable fuel column length [~ 10 inches (25 cm)] operating at central melting conditions was reported. This experiment, the X-1-e-3, resulted in a "very disastrous failure"⁽²⁾ after operating for 3 hours in the NRX loop at Chalk River. A large quantity of fuel was lost from the specimen and the cladding was destroyed over a length of 1-1/4 inches (3 cm). As a result of this experience, and without having identified the exact mechanism of failure, it was recommended that UO_2 fuel rod operation be restricted to less than central melting conditions. This center melt limit has been used throughout the industry to date. (On the basis of the pellet diameter, initial gap, UO_2 enrichment, and peak surface heat flux of 682,000 Btu/h/ft² (214 W/cm²) reported for the X-1-e-3,⁽¹⁾ UO_2 central melting definitely should not have occurred. Even considering the initial 0.005-inch (0.013-cm) defect in the capsule and the resulting internal steam atmosphere, melting would have been most unlikely. Nevertheless, the as-cut cross section of the capsule shows unequivocal central melting to a very large diameter. Unquestionably, the X-1-e-3 actually operated at a much higher heat flux than reported. The relatively high burnup value taken from the capsule fission monitor also supports this conclusion.)

For the first generation of power reactors the central temperature limit did not impose too significant a hardship. Other considerations, particularly heat transfer, fission gas release, and a generally conservative design approach dictated fuel rod operating conditions considerably below the center melting limit. With succeeding reactors and the progression to fuel rods operating at higher specific power, the center melting limit has resulted in using smaller diameter fuel rods with consequent, less favorable fuel economy. As a result, it became very desirable to re-examine the validity of central melting as a design limit. The experimental program described here was undertaken with the broad objective of exploring the feasibility and limits of UO_2 fuel rod operation with central melting.

An orderly progression of experiments in the central melting regime was selected as the best approach to satisfy the over-all objective of the program. This plan offered the maximum return of experimental data, particularly with regard to identifying the basic failure mechanism (s).

associated with operating at central melting conditions, and minimized the risk of another severe in-pile failure. A sequence of fuel assembly irradiations was designed with each succeeding assembly operating at a significant increment in thermal performance level above the preceding one. The thermal performance level of the initial assembly was chosen to produce a small diameter zone of central melting over a short length of each fuel rod. The design and fabrication processes were identical for all the fuel rods to minimize the influence of fabrication variables on the results. The only fabrication change between assemblies was in UO_2 enrichment to obtain the desired thermal performance increments. The following additional general guidelines were observed to enhance the significance and applicability of the program results.

1. The fuel rod design was chosen to be typical of that used in present day power reactors, e.g., freestanding Zircaloy-2 cladding and an unsegmented, reactor length fuel rod with a fission gas plenum at the upper end.
2. Every possible care was exercised in the design, fabrication, and inspection of the fuel rods and assembly hardware, and also in the selection of the fuel rod external operating conditions to preclude failure from extraneous sources. Very conservative margins with respect to heat-transfer burnout limits were emphasized since this type failure could easily be confused with that originating from UO_2 melting.⁽³⁾
3. The irradiation loop instrumentation was upgraded significantly to minimize the uncertainty limits regarding the exact fuel rod operating conditions (Appendix A).
4. Extensive provisions were made for safe loop operation and subsequent decontamination in the event of a severe fuel rod failure.

The original program plan was to irradiate each fuel assembly in the series for a single reactor cycle up to the maximum thermal performance assembly operable without failure. This initial sequence was expected to identify any short time limitation on fuel life as a result of operating with central melting.

Based on the experience gained from the initial series, a selection was then to be made of a thermal performance level for the irradiation of an assembly to very high burnup, tentatively 20,000 MWD/ T_e (5.4×10^{20} fissions/ cm^3). This assembly would identify any time-dependent mechanism unique to operating with UO_2 central melting that would limit fuel life. The program included an identical irradiation plan with compacted powder UO_2 fuel rods. The powder fuel assembly irradiations, now in progress,^(4,5) were started much later. The experience gained from the pellet fuel irradiations has simplified the powder irradiation program. Only the experience from the UO_2 pellet fuel assembly irradiations are summarized here.

II. DISCUSSION OF RESULTS

A. CLAD SWELLING AND UO₂ MELTING

The irradiations of the initial three pellet fuel assemblies, EPT-6, EPT-8, and EPT-10 were made in consecutive GETR cycles. Post-irradiation examination of the assemblies started very shortly after their removal from the loop; only the minimum time necessary for decay of the assembly after-heat was allowed. By the time the EPT-10 assembly irradiation was terminated, the EPT-6 assembly post-irradiation examination had been completed, including metallography, and non-destructive examination of the EPT-8 assembly was also complete. At this time, it was apparent that the program could no longer proceed according to the original plan. The severe clad swelling observed in the initial two assemblies coupled with the early failure of the EPT-10 assembly - in all probability caused by the same phenomenon - precluded operating at higher thermal performance levels. Both an understanding of the clad swelling mechanism and a specific correction for the problem were required before resuming irradiations.

Taken as a whole, all the data collected in the post-irradiation examinations even this early in the investigation, indicated a direct relationship between the clad swelling and UO₂ melting. The bases for this conclusion were:

1. The correspondence of the maximum swelling zone to the peak heat flux zone (maximum region of melting) at startup,
2. The correspondence of threshold thermal performance level for swelling with that required to initiate melting, and
3. The metallographic appearance of the maximum swelling zone cross section from rod 6D.

The obvious mechanism for the swelling was the occurrence of a significant UO₂ volume increase accompanying the solid-to-liquid phase change. However, before the start of this program, the density change accompanying melting was reported to be small, probably less than 3%. (6) Thus, considering the original 95 to 96% density of the pellets, it would be impossible to generate a net fuel volume increase by melting.

Almost simultaneously with the occurrence of the fuel rod clad swelling during these irradiations, new measurements of the UO₂ volume change on melting were reported (7, 8) which indicated a much larger volume increase. Preliminary results suggested a value of 7% which subsequently refined to 9.6%. A volume change of this magnitude would be sufficient to generate clad swelling, although it was not immediately obvious how to rationalize the very large, localized diameter increases in the Pressurized Water Loop (PWL) fuel rods. Based

on earlier irradiations of almost identical capsules in the GETR Trail Cable facility, at essentially equivalent thermal performance levels, ^(9, 10) the swelling in the PWL fuel rods was totally unexpected. Careful dimensional measurements made on the Trail Cable capsules indicated no detectable clad growth.

In comparing the capsules to the PWL fuel rods to explain the difference in results, it became apparent that a mechanism for axial expansion of the molten UO_2 was inherent in the design of the capsule. This mechanism resulted from the combined effects of using hollow, Zircaloy end plugs and the flux peaking at the end fuel pellets. Because of the flux peaking, the end fuel pellets operated at a higher central temperature relative to those within the fuel column. Thus, the end pellets experienced central melting first and provided an axial path for UO_2 expansion when the remaining pellets within the fuel column subsequently melted. Axial expansion and movement of the fuel into the adjacent end plenum was observed in each of the capsules. ^(9, 10)

Conversely, significant axial fuel movement cannot occur in the reactor length, unsegmented PWL fuel rods. Because of the axial power distribution in the reactor, the very high thermal performance region containing molten UO_2 is confined from above and below by relatively cold, nonplastic fuel pellets. Based on previous experience, it is known that during high thermal performance operation, the fuel pellets are locked tightly in the fuel cladding and are incapable of independent axial movement. ⁽¹¹⁾ Thus, the UO_2 volume expansion appears to have been accommodated by axial movement in the capsules but could not be similarly compensated in the PWL fuel rods. With this explanation, the UO_2 volume expansion caused by melting seemed to be the probable mechanism for clad swelling.

B. EFFECT OF FUEL LENGTH

The capsule experience also provided an insight to the mechanism by which the UO_2 volume expansion could generate the very large, localized diameter increase in the PWL fuel rods. Once the clad swells slightly at the axial location for initial melting, i. e., the peak heat flux zone, this same cladding zone is less resistant to further swelling because of its larger diameter. ($\sigma_t = Pr/t$). Consequently, the volume increase resulting from additional melting above and below the initial molten zone is transmitted hydraulically and concentrated in the initial zone of swelling. Thus, even a slight, net volume increase of the fuel on melting can generate a very large, localized diameter increase if the length of the molten zone is appreciable. For this same reason, the dimensional changes observed in short length, capsule experiments with central melting are not representative of those that occur in reactor length, unsegmented rods.

C. PREVENTION OF CLAD SWELLING

The different performance of the capsules and the PWL fuel rods relative to clad swelling suggests a simple, if somewhat uneconomical, solution to the clad swelling problem. The fuel

rods could be simply subdivided into capsule length segments. If the segment lengths are sufficiently small so that each segment operates with a relatively uniform power over its length and if each segment is provided with appropriate axial expansion volume, clad swelling should be positively eliminated. While decidedly unattractive compared to a "through" rod design, this solution positively prevents axial fuel relocation and, therefore, the necessity for very carefully programmed reactor startups with "through" rods as discussed later.

The EPT-COM assembly irradiation and the supporting capsule irradiations were undertaken with the twofold purpose of verifying the suspected clad swelling mechanism and demonstrating a solution to the problem other than segmenting. It was evident that if the UO_2 volume expansion on melting was responsible for clad swelling, then the provision of free volume within the fuel and accessible to the molten core should prevent clad swelling. The method of providing the free volume was to use hollow UO_2 fuel pellets. In addition to fabrication ease, the use of hollow pellets has the advantage of providing a relatively large, interconnected free volume, directly adjacent to the point of initial fuel melting. Since the exact amount of free volume required was unknown, the interconnected void would permit excess molten UO_2 to flow axially from the zone of maximum melting to a less severely melted zone.

The significant clad swelling produced in capsule 8 (Section XI) provided considerable confidence that the correct swelling mechanism had been identified. The fact that swelling could be deliberately generated by preventing axial fuel movement was strong confirmation that the UO_2 volume expansion on melting is the basic clad swelling force. The diameter increase in capsule 8 was several times larger than that previously observed in a capsule of this original diameter with freestanding clad. Although the diameter increase was not nearly so large as that in the PWL fuel rods, it is understandable from the previous discussion on the effect of fuel length on swelling.

In addition to the objectives already mentioned, the purpose of the EPT-COM assembly irradiation was to investigate the effect of repetitive power cycling on clad swelling. The swelling in the previous assemblies was believed to have occurred almost entirely during the initial rise to full power, but the possibility of a ratcheting mechanism with repetitive power cycling could not be completely discounted. For this reason, the EPT-COM irradiation was performed in two steps, i. e., a single, short cycle of full-power operation followed by examination and then a full cycle of irradiation with power cycling.

Even though the 10E rod indicated some swelling after the initial cycle of irradiation, it was very evident that the swelling was only slightly greater than that necessary to prevent removal of the rod from its shroud tube, i. e., approximately 10 mils, (0.025 cm). This, coupled with the complete absence of swelling in the 12E rod, confirmed the effectiveness of the hollow pellet concept in significantly reducing clad swelling for at least the initial power cycle. Both of these rods operated at thermal performance levels equivalent to those for the fuel rods in the EPT-10

assembly which had shown severe clad swelling. The condition of the assembly following the subsequent entire reactor cycle of irradiation, including four full power cycles, provided almost complete confidence that a ratcheting mechanism was not involved. None of the previously unswollen rods developed swelling during the second irradiation. The maximum additional swelling that could have occurred in rod 10E was 3 mils (0.008 cm) and, in all probability, the rod was unchanged from its first post-irradiation examination. The slight swelling experienced in rod 10E undoubtedly indicates that the free volume provided by the 0.085-inch (0.22-cm) diameter central hole was marginally insufficient for the extent of melting experienced at the 10E thermal performance level.

As a result of the almost completely favorable results from the EPT-COM assembly irradiation and the supporting capsule experiments, the resumption of the original irradiation program appeared feasible. The final, maximum thermal performance fuel assembly (EPT-12C) to be irradiated was enriched to operate at a nominal peak surface heat flux of 1.2×10^6 Btu/h-ft² (377 W/cm²). Because of the greater volume of UO₂ melting expected in this assembly as a result of its higher (than rod 10E) thermal performance level and the slight clad swelling experienced in rod 10E, the diameter of the central hole in the EPT-12C fuel pellets was increased to 0.140 inch (0.360 cm). The free volume provided by this larger core is approximately 8% of the total fuel volume compared to the slightly less than 3% provided by the 0.085-inch (0.22-cm) hole in the EPT-COM fuel rods. The satisfactory irradiation of the EPT-12C assembly at extreme thermal performance conditions and to relatively high burnup with extensive power cycling is considered a conclusive demonstration of the effectiveness of hollow fuel pellets in preventing clad swelling. The EPT-12C fuel rods were both visually and dimensionally indistinguishable from their pre-irradiation condition during the interim examination after 5300 MWD/T_e (1.4×10^{20} fissions/cm³) burnup.

D. AXIAL FUEL RELOCATION - STARTUP POWER PROGRAMMING

Even though the EPT-COM assembly operated through several full-power cycles without apparent effect on clad swelling, the concern about power cycling was not completely eliminated. The power cycles with the EPT-COM assembly were rapid, i. e., scrams followed by power recovery within 1 hour. Under these conditions the flux peak location during power recovery was not lower than that before scram and usually was higher. The favorable results from the EPT-COM assembly relative to power cycling may only be applicable to this specific type of cycling. Based on the observations of axial fuel relocation, the molten fuel appears to flow down the rod and concentrate in the peak heat flux zone at startup. The use of initially hollow pellets will, if anything, promote this axial fuel relocation. In addition, the shifting flux profile in the GETR probably intensifies the fuel relocation further. As the flux profile shifts upward during a reactor cycle, more fuel melts at the upper end and becomes available to fill and re-solidify in the lower end of the rod. Thus, the axial fuel relocation associated with the initial reactor cycle of operation appears to produce an unfavorable redistribution of fuel and void,

despite the initial coring. If the reactor is then refueled, moving the flux peak to the lower end of the rod, clad swelling may very well occur during the next startup. The point of initial melting will be in a very high density zone of UO_2 with very little available free volume for expansion. Unless the molten UO_2 can move axially up a very small central void, the free volume now concentrated at the upper end of the rod will be ineffective in accommodating UO_2 expansion. The startups following a refueling appear to be the critical time for hollow pellet fuel rods; at this time, conditions are most conducive to clad swelling.

Recognition of this problem resulted in the adoption of a procedure for a very carefully controlled reactor startup. This procedure was instituted with the first cycle of irradiation on the EPT-12C assembly. It appeared that even if the zone of initial melting in the UO_2 was isolated from the available free volume by unmelted UO_2 in between, the UO_2 expansion could be accommodated, if the melting occurred at a slow enough rate. With temperatures sufficient to produce melting at the lower end of the rod, the unmelted UO_2 above this zone would still be operating at a sufficiently high central temperature to exhibit significant plasticity. (12, 13) With a slow enough rate of melting, the expansion could be accommodated by deforming the unmelted UO_2 upward into the available void volume. The internal pressure required to deform the plastic UO_2 at this slow rate would probably be less than that necessary to produce radial clad swelling.

The basic objective in establishing the reactor startup procedure was to prevent any sudden, large increases in the internal volume of the fuel rod, i. e., step increases in power. To accomplish this objective, the maximum reactor power rise increment was prescribed as 1 MW. The power rise rate was limited to 1 MW/15 min interval. The power programming was instituted at a power level below that required to initiate central melting in the assembly fuel rods. Power programming was specified for all startups following a reactor refueling, provided the predicted flux peak location after refueling was 3 inches (8 cm) or more below that existing at the previous shutdown. Of the 22 power cycles on the EPT-12C assembly; 3 required programmed startups following the above criteria. The remaining 19 power cycles were scrams followed by rapid power recovery without refueling, similar to the power cycles with the EPT-COM assembly.

E. JUSTIFICATION FOR POWER PROGRAMMING

Power programming was instituted purely as a precautionary step based on the observed fuel redistribution and probable consequences with respect to clad swelling. There was no direct experimental evidence to indicate the necessity of power programming; therefore, the extent to which the programmed startups contributed to the satisfactory operation of the EPT-12C assembly was unknown. However, the subsequent failure of the EPT-12C assembly provided some tentative evidence supporting the need for power programming with unsegmented fuel rods.

The EPT-12C assembly exhibited no detectable clad swelling as a result of the initial two cycles of irradiation. The dimensional examination was performed 1 week before inserting the

assembly for the third cycle of irradiation. The assembly failed during the third cycle startup at a power level of 26 MW. When again examined, all four rods showed some clad swelling. The failure was a split in the maximum swelling zone of the 12D rod. When the loop records were examined, the temperature rise instrumentation indicated that the element had experienced an approximate 4 MW step increase in power, about 1-1/2 hours before the failure. Power programming had been very carefully followed both before and after this incident. The large increase in assembly power resulted from a rapid correction of the GETR power level to bring the indicated power on nuclear instruments in conformity with that indicated by calorimetry. The indicated correction on the reactor instruments was also 3 to 4 MW. The discrepancy between nuclear and calorimetric power indications is normally much smaller.

The simultaneous occurrence of clad swelling with the large power rise increment indicates a relationship between the two incidents. New cored pellet fuel rods essentially identical to the original EPT-12C rods have since been operated at higher thermal performance levels without clad swelling after correctly programmed startups. No other explanation has been found for the sudden development of clad swelling in the original EPT-12C rods. This experience, coupled with the original observations that led to power programming, appears to be reasonable grounds for considering the procedure mandatory for long unsegmented fuel rods with appreciable central melting. The startup power programming will be continued for all subsequent fuel assembly irradiations under this program. The exact procedure to be followed will be modified as dictated by additional experience. The previously specified startup rate, although arbitrarily selected, appears to be the maximum that can be recommended.

F. UO₂ THERMAL CONDUCTIVITY

It should be re-emphasized that the assumptions in the analysis to determine the UO₂ thermal conductivity were deliberately chosen to produce a high result. An estimate was desired of the maximum UO₂ thermal conductivity that can be inferred from these irradiations. Precise evaluation of the conductivity of the pellet-to-clad interface is expected to result in a few watts per centimeter decrease in the

$$\int_0^{T_m} k dT$$

values, particularly for those sections that experienced severe clad swelling.

The most consistent results are those calculated by using the porosity termination boundary in the UO₂ metallography as the molten boundary indicator and the fuel rod heat flux at end of life without correction for clad swelling (Table 18). The average value of the integral from 0°C to melting is 94.3 W/cm is based on an examination of ten sections with a confidence limit of 2.9 of W/cm (95% probability). In calculating this average, two metallographic sections

(No. 8600 and 8748) were neglected, both of which indicate very low values for

$$\int_0^{T_m} k dT$$

Both metallographic sections were taken at axial locations near the upper end of the fuel rods; one from rod 8A and the other from rod 12E. The probable significance of the results from these two metallographic sections is discussed in Section III-G.

When the heat flux values in the above analysis are corrected to reflect the expected increases accompanying clad swelling, the average value of the conductivity integral increases from 94.3 to 97.6 W/cm. In addition, the 95 % confidence limit of the average increases from 2.9 to 4.5 W/cm which indicates considerably less consistency in the individual values. Although it appears logical to expect a higher surface heat flux at the swollen sections because of their larger fuel volume, the magnitude of the increase may be less than the proportionality to diameter assumed. The density of the fuel filling the molten sections is generally less than the average fuel density along the rod, i. e., a greater fraction of the fuel is above the melting point. Since the sections with swelling are also the sections for which the interface conductivity is believed to be the most conservative, one effect may partially offset the other. In other words, the surface heat flux may actually increase because of clad swelling; but the improvement in the conductivity of the pellet-to-clad interface from increased contact pressure is compensatory so that essentially the same temperature levels in the UO_2 are maintained. A similar effect was observed in the earlier short time capsule irradiations for thermal performance levels below that sufficient to produce central melting.^(9, 10) Over a surface heat flux range of nearly 300,000 Btu/h-ft² (94 W/cm²), the interface conductivity varied almost linearly and resulted in a virtually constant pellet surface temperature. Finally, there is evidence to indicate that axial convective heat transfer occurs within the molten fuel column. This axial heat flow would tend to smooth out any sharp variations in fuel rod surface heat flux attributable to dimensional variations and would also tend to reduce the fuel rod peak surface heat flux values below that calculated on the basis of the axial flux profiles. Most of the metallographic sections examined are taken from peak heat flux locations. On this basis, the conductivity integral based on the initial rod diameter is considered the more reliable value, i. e., 94.3 ± 2.9 W/cm.

In Table 17, it can be seen that almost the identical average value for the conductivity integral was obtained by using the autoradiographic results. The average value based on ten metallographic section autoradiographs is 95.6 W/cm with a 95% confidence limit of ± 5.1 W/cm. It should be especially noted that the observed autoradiograph boundary is believed to represent the maximum limit of melting in the sections at any time during life and the integral values were calculated on the basis of the maximum heat flux that occurred at the section. Thus, even though the autoradiographs indicate the extent of melting at a different time than that

indicated by the metallography, the calculated UO_2 thermal conductivity is the same. This correspondence provides considerable confidence that the selected autoradiograph characteristics do define the UO_2 molten boundary.

By recognizing that the above value of approximately 95 W/cm for the UO_2 thermal conductivity integral may decrease slightly by refining the conductivity value for the pellet-to-clad interface, the result is very close to that determined previously from short-time capsule irradiations.⁽¹⁴⁾ The capsules were irradiated for 20 minutes and indicated an integral conductivity value of 86 W/cm based on the time average surface heat flux over the total irradiation period. If the capsule results are calculated on the basis of the maximum instantaneous heat flux during irradiation (maximum duration approximately 3 minutes) the indicated UO_2 thermal conductivity integral becomes 93.3 W/cm.^(9,10) Thus, the results from both the short-time capsule irradiations and these fuel rods with appreciable burnup indicate nearly the same value for the UO_2 thermal conductivity integral, i. e., 92 to 96 W/cm. Considering the uncertainties in the analysis, the results cannot be considered significantly different. It is very evident that any improvement in UO_2 thermal conductivity as a result of extended irradiation and columnar grain formation is slight and does not exceed a few watts per centimeter. This result contradicts the large improvement at elevated temperatures reported by Bates.⁽¹⁵⁾

As a further confirmation of the value of the integral for melting, the UO_2 conductivity integral indicated by the onset of clad swelling (UO_2 melting) can be compared to that from metallography and autoradiography. Extrapolation of the least squares curve to the abscissa indicates the onset of clad swelling at a value of 92 W/cm, in excellent agreement with that from metallography and autoradiography. In these calculations, the interface conductivity also was assumed to be 2000 Btu/h-ft²-°F. There may be some finite volume of UO_2 melting required before it is reflected in clad swelling. Consequently, the integral value of 92 W/cm from the clad swelling data also represents an upper limit for the integral. These results compare very closely with the more recent results obtained by other investigators. Bain⁽¹⁶⁾ has reported a value of 91 to 92 W/cm for stoichiometric UO_2 ; Duncan⁽¹⁷⁾ has reported a value of 93 W/cm; Robertson, et, al.⁽¹⁸⁾ have reported a value of 95 to 96 W/cm, and Williamson and Hoffman⁽¹⁹⁾ have reported a value of 89 W/cm for the integral from 0°C to melting. All the values are considerably lower than the value of 125 W/cm (calculated) reported by Bates⁽¹⁵⁾ for 95% dense UO_2 . Considering all these results, it is concluded that the true value of the conductivity integral lies in the range of 90 to 96 W/cm. The most probable value is closer to 90 W/cm. To ensure that central melting and the associated clad swelling problem are avoided in solid pellet, UO_2 fuel rods, the maximum operating thermal performance level should not exceed

$$\int_0^T k_{\text{UO}_2} dT = 90 \text{ W/cm}$$

These conclusions for the value of the UO_2 thermal conductivity integral have been coupled with (a) all the available out-of-pile data on UO_2 thermal conductivity, ⁽²⁰⁾ and (b) data on the conductivity at elevated temperatures derived from grain growth studies ⁽¹⁰⁾ in irradiated capsules to provide a best estimate of the actual UO_2 conductivity curve as a function of temperatures. This curve is shown in Figure 1. The curve was obtained by making the usual least squares fit to the data with the general equation $k = \frac{A}{B + T} + CT^3$. However, one additional constraint was applied to the curve fitting process. The integral of the specific conductivity curve was forced to equal 90 W/cm to reflect these experimental results for the value of the integral. The equation of the curve best satisfying both of the above requirements is

$$k_{\text{UO}_2} = \frac{38.24}{T + 129.4} + 4.79 \times 10^{-13} T^3$$

where T is in °K and k is in W/cm-°C.

From Figure 1 it is evident that there must be some conductivity upturn at elevated temperatures to satisfy both the low temperature data and the integral. Apparently, a new conduction mechanism becomes significant since lattice conduction ($k = A/B + T$) alone would continue to decrease asymptotically. The form of the assumed conductivity equation follows that proposed by Bates ⁽¹⁴⁾ in which internal radiation ($k = CT^3$) is assumed to be responsible for the apparent elevated temperature conduction increase. However, other investigators ⁽²¹⁾ have proposed that the elevated temperature increase is from electronic conduction ($k = C \exp(-D/T)$). The mechanism underlying the conductivity upturn has not been resolved.

G. AXIAL CONVECTIVE HEAT TRANSFER IN MOLTEN UO_2

In calculating average values of the UO_2 thermal conductivity integral, two metallographic sections were neglected. Both these sections were obtained from locations very near the top of the fuel column: one from rod 8A and the other from rod 12E. Originally, these two sections had been expected to be the most definitive regarding the UO_2 conductivity integral because of their location and consequent thermal performance history (Section X-C-3). However, the integral values obtained from these two sections are so low that they actually fall outside the probability distribution of all the other values with greater than 99% confidence level. Because of the methods of calculation used, it is almost impossible for this result to be in error. Once the correct value for the average thermal performance level of a particular fuel rod is established, the values at the various axial locations depend only on the accurately known axial power distribution. Considering the uncertainties involved in determining the average power level of the fuel rods, it is expected that there will be greater variation between sections in different rods than between sections from the same rod. The discrepancy found in these two sections caused all the axial locations for the various sections to be redetermined, as well as the assembly axial location relative to the GETR core to be remeasured. The results of these remeasurements

Figure 1. UO_2 Thermal Conductivity vs Temperature for 95% Theoretical Density

indicated that the quoted axial location values for the sections are accurate to $\pm 1/4$ inch (0.6 cm) and that the assembly axial location relative to the GETR core is actually $1/2$ inch (1.3 cm) higher than was originally believed. This latter result actually reduced the integral values for the two sections a slight amount which increased the discrepancy.

The analyses to determine the effect of burnup and xenon poisoning on the axial flux distributions (Appendix B) were also undertaken to reconcile the results obtained from these two sections. Again the corrections were much too small to affect the basic result, and it was finally concluded that the discrepancy could not be attributed to errors or inapplicable data.

The basic difficulty found in these two sections is a much larger zone of UO_2 melting than would be expected from calculations based on the clad heat flux. The calculated heat flux for the section from rod 8A is actually considerably below that necessary to initiate central melting. Nevertheless, the presence of appreciable central melting in the section is definite and unequivocal. It was finally realized that if there were internal circulation of the UO_2 within the molten fuel column, the circulation could convect heat axially up the rod and account for the evident higher thermal performance levels at the upper end. This effect could also account for the relatively abrupt termination of the molten region evident in section 8600 from rod 8A. The flux profile along the fuel rod in this zone varies almost linearly and the boundary of the molten zone would also be expected to vary smoothly. Instead, there is a rapid neck down of the molten boundary at the upper end of the molten zone. Finally, internal fuel circulation could well explain the progression in gamma scan profiles discussed here. The circulation would tend to redistribute the fission products uniformly over the total length of the molten zone and could lead to the very nearly flat activity profiles observed in the EPT-12C assembly fuel rods.

The existence of the postulated axial convection of the molten fuel is by no means proved, but it is the only mechanism known that will adequately explain the large degree of UO_2 melting at the upper end of these fuel rods. The existence of this mechanism was proposed independently by Williamson and Hoffman⁽¹⁸⁾ to account for similar effects in their central melting tests. If the mechanism does occur, other anomalies should be readily apparent in post-irradiation analyses. For example, the radial position of the boundaries for equiaxed and fine columnar grain growth should not vary axially as rapidly as indicated by the flux distribution. The axial variation in fission product and plutonium concentrations and U-235 to U-238 isotopic ratio should also be less than that predicted from the flux profiles. Additional work is in progress to verify the expected anomalies and thereby provide more positive evidence for existence of axial heat convection in the molten UO_2 .

H UO₂ VOLUME EXPANSION ON MELTING

The UO₂ volume expansion during melting derived from the experimental data of this program is 7%. The experimental uncertainty is approximately 50% or $7 \pm 3.5\%$. The experimental uncertainty is almost entirely attributable to the variance in the slope of the experimental volume expansion curve. The uncertainties in the analytical portion of the volume expansion determination are essentially negligible in comparison.

This result is in reasonable agreement with the value of 9.6% from the direct volume expansion measurements of Christensen.^(7,8) Christensen's value has an apparent experimental uncertainty of about 30% or $9.6 \pm 3.0\%$. Thus, each value is well within the uncertainty band of the other, and both values confirm a substantially larger volume expansion during melting than previous results.⁽⁶⁾

The large uncertainties in both of these measurements are the direct result of determining the values from the difference of two considerably larger numbers. Even though both numbers are measured accurately, the relative uncertainty in the difference is greatly magnified. To obtain a more accurate measurement of the volume expansion, an experimental technique must be devised to eliminate the necessity for relying on differences.

III. CONCLUSIONS

The experimental results obtained from the UO_2 pellet fuel irradiations are summarized in the following conclusions:

1. Pellet UO_2 fuel rods can be successfully operated with gross central melting, provided the fuel rods are properly designed initially.
2. When operated at thermal performance levels sufficient to produce central melting, severe clad swelling occurs in long, unsegmented fuel rods containing solid UO_2 pellets.
3. The clad swelling mechanism has been identified as the volume expansion of UO_2 on melting. These irradiations indicate a UO_2 volume increase on melting of $\sim 7\%$.
4. Short length capsules with relatively uniform axial power distributions do not generally experience the severe clad swelling that occurs in long fuel rods. The UO_2 volume expansion in this instance is accommodated by UO_2 movement into the available axial free volume.
5. The provision of internal free volume in the fuel which is accessible to the molten core, has been demonstrated to prevent clad swelling. Specifically, the use of hollow UO_2 fuel pellets with 0.140-inch (0.360-cm) central hole (equivalent to 8% additional free volume) has been shown to prevent clad swelling for operation of 0.500-inch (1.27-cm) fuel pellets at

$$\int_0^{T_G} k dT$$

values up to 200 W/cm.

6. There has been no evidence of any failure mechanism other than clad swelling that will prevent fuel rods with appreciable central melting from operating to long life.
7. The appearance of the UO_2 in post-irradiation microstructures is determined primarily by the operating conditions just before shutdown. Evidence of prior operation at higher thermal performance levels, e. g., at levels sufficient to produce significant amounts of UO_2 central melting, can be completely eliminated from the microstructure by subsequent operation at lower thermal performance.

8. Considerable evidence has been accumulated regarding the identification of a molten boundary in UO_2 microstructures and in autoradiographs. For relatively mild temperature gradient conditions, a band of enlarged, almost spherical porosity appears in UO_2 heated to just below the melting point. The abrupt disappearance of this porosity is believed to mark the transition to melting. Central voids produced by UO_2 melting have been observed with diameters out to, but not beyond, the inside edge of this porosity band.
9. Analyses of these experiments indicates that the value for the UO_2 thermal conductivity integral

$$\int_{0^\circ\text{C}}^{T_m} k dT$$

lies in the range of 90 to 96 W/cm with the lower value more probable. To ensure no central melting and associated clad swelling, the maximum thermal performance level for solid pellet, UO_2 fuel rods, expressed by the parameter,

$$\int_0^{T_m} k dT$$

should not exceed 90 W/cm.

10. Both radial and axial relocation of fission products and axial relocation of fuel is apparent in fuel rods that have operated with appreciable central melting. However, physics analysis and the actual operating experience in these irradiations indicate that the axial fuel relocation does not significantly affect reactivity in a thermal core.
11. Under certain conditions, axial fuel relocation can displace the zone of initial melting from the required free volume for expansion, even for fuel rods with initially hollow fuel pellets. For this reason, startups of unsegmented fuel rods that have previously operated with appreciable central melting must be carefully programmed to limit the rate of UO_2 melting and prevent clad swelling.
12. In addition to the axial relocation of fuel, which occurs primarily during solidification at shutdown, there is evidence of axial circulation and heat convection within the UO_2 molten fuel zone during operation. This convection appears to flatten the axial power distribution from that expected on the basis of axial neutron flux profiles.

13. Fission gas release data determined by post-irradiation rod punctures indicate a maximum release of approximately 62% of theoretical for rods operating up to 1.4×10^6 Btu/h-ft² peak surface heat flux, i. e. , somewhat lower than expected on the basis of prior experience. Determination of fission gas release by post-irradiation sampling may be unrealistic for fuel rods with appreciable central melting. Significant percentages of the filler gas may be trapped in closed pores during the molten UO₂ solidification and, thus, be undetectable in the post-irradiation measurement.

IV. EVALUATION OF RESULTS

Before starting this program, four possible mechanisms had been postulated to explain the failures that had occurred during attempts to operate under central melting conditions:

1. Excessive fuel rod internal pressure as a result of high fission² gas release,
2. Excessive clad strains caused by the UO_2 volume increase with melting,²
3. Clad melting from contact with molten UO_2 escaping through cracks² in outer rim of unmelted UO_2 .
4. Chemical reaction between the UO_2 and the clad at the interface because of the high interface temperature produced by high specific power operation.

The first of these mechanisms may have been important in the previous failures with UO_2 central melting, but it is not a real limit since the gas release can be handled with adequate initial fuel rod design. The second mechanism was originally proposed to explain the X-1-e-3 failure mentioned in the Introduction, but could not be reconciled with the then available data on UO_2 expansion with melting.⁽⁶⁾ These data indicated less than 3% expansion during the phase change. Subsequent measurements have shown the expansion to be approximately 7 to 10%.^(7, 8) The early results from this program coupled with the more recent UO_2 volumetric expansion data have conclusively demonstrated that the second failure mechanism is the significant failure mechanism insofar as the initial attainment of central melting operating conditions is concerned. This mechanism was also undoubtedly responsible for the X-1-e-3 failure. However, it has been demonstrated in this program that the second failure mechanism is not an insurmountable limit. It, too, can be handled by appropriate initial fuel rod design. Thus, only the third and fourth mechanisms remain as potential limitations to UO_2 fuel rod operation with central melting.

The successful irradiation of the EPT-12 assembly to 5300 MWD/ T_e (1.4×10^{20} fissions/ cm^3) burnup and through 23 power cycles provides considerable confidence regarding the third postulated failure mechanism. Either the mechanism does not actually function, or it functions so infrequently it has not yet been observed. With only four fuel rods in the EPT-12 assembly, any indication of the frequency of occurrence is unlikely. However, the experience with the identical series of assembly irradiations using compacted powder fuel rods is very significant to an evaluation of the third failure mechanism. With compacted powder, countless paths are available for molten UO_2 to flow out to the cladding. At this time, 85% dense, vibratory compacted fuel rods have been operated up to peak surface heat fluxes of 1.45×10^6 Btu/h-ft² (446 W/cm^2) without failure.⁽⁵⁾ The burnup attained is 10,000 MWD/ T_e (27×10^{20} fissions/ cm^3) and over 30 power cycles have been experienced. The successful attainment of these conditions

is believed to be sufficient to reach a preliminary conclusion on the third failure mechanism. In the absence of clad swelling, this postulated mechanism of failure does not appear to be of real concern.

The fourth failure mechanism cannot be evaluated accurately at this time. Very careful metallographic examination of the clad-to- UO_2 interface in Rod 12E from the EPT-COM assembly has shown no evidence of a chemical reaction after 2014 MWD/ T_e (0.54×10^{20} fissions/cm³) burnup at peak surface heat fluxes up to 1.15×10^6 Btu/h-ft² (364 W/cm²). However, this failure mechanism should be a definite function of the operating time and may become important at higher burnups. The irradiation of the EPT-12 assembly will be continued to 20,000 MWD/ T_e (5.4×10^{20} fissions/cm³) to evaluate this mechanism as well as to determine whether there is any other, unsuspected failure mechanism (unique to operating under central melting conditions) that will limit fuel life. On the basis of the experience to date, and assuming proper fuel rod design and care in operation to account for the UO_2 volume increase on melting, the capability of UO_2 fuel rods to operate satisfactorily with gross central melting to very high burnup seems very probable. The EPT-12 assembly is believed to represent an accelerated test of the central melting phenomenon because of its extreme thermal performance conditions. The 5300 MWD/ T_e (1.4×10^{20} fissions/cm³) burnup achieved with this assembly should provide confidence for high burnup operation of fuel rods with less severe central melting. Of course, in any application, the fuel rod external requirements must also be met, i. e., adequate heat-transfer conditions and a suitable corrosion environment for the specific cladding used.

V. IRRADIATION CONDITIONS

The fuel irradiations are being performed in the GETR - PWL. A detailed description of the loop is available in Technical Manual, Rev. No. 1, GEI-80866. The PWL is primarily a carbon steel system with the exception of the in-pile facility tube which is Type-300 series stainless steel. The location of the loop relative to the GETR is shown in Figure 2. The loop can be operated with a maximum coolant pressure of 1500 psi ($1 \times 10^6 \text{ kg/m}^2$) at 600°F (316°C) and a 500-kW element can be tested at the normal coolant operating temperature of 525°F (274°C). The loop is equipped with two centrifugal pumps in series that develop a combined, shutoff flow head of 185 psi ($1.3 \times 10^5 \text{ kg/m}^2$) and a maximum loop flow rate of 135 gpm (8.5 liters/sec) with the present design test element installed. The PWL including its auxiliary systems is shown in Figure 3.

The nominal coolant conditions selected for the performance of these experiments are:

Pressure	1000 psia ($7 \times 10^5 \text{ kg/m}^2$)
Saturation Temperature	545°F (285°C)
Fuel Assembly Inlet Temperature	344°F (173°C)
Fuel Assembly Outlet Temperature	~370°F (188°C)
Flow Velocity at Fuel Rods	16 to 34 ft/sec (292 to 621 m/min.)
pH (LiOH regenerated demineralizer)	9.5 to 10.0
Dissolved H ₂	25 to 65 cc/kg

These conditions were carefully chosen to minimize the possibility of a fuel rod failure from external causes. The coolant chemistry conditions provide an optimum corrosion environment for the Zircaloy-2 cladding while eliminating the tendency of corrosion products to deposit on heat transfer surface of the fuel rods. This problem is most significant in a predominantly carbon steel system and required considerable study and experimentation prior to operation with fuel in the loop. Under the above coolant conditions, there has been no perceptible crud deposition on the fuel rods during operation. The post-irradiation visual appearance of the fuel surfaces is indistinguishable from the pre-irradiation condition as shown in Figure 4.

The extremely suppressed coolant inlet temperature was selected to prevent a heat transfer burnout failure, even at the highest thermal performance operating conditions anticipated. Under the chosen conditions, fuel rod cooling is entirely in the forced convection regime with no nucleate boiling. The indicated amount of subcooling actually provides a safety factor of 1.5 relative to the onset of nucleate boiling at the reactor 125% overpower condition with 10% allowance for experimental uncertainties.⁽³⁾ The heat transfer operating regime for these assemblies is shown in Figure 5. The specific operating conditions for each fuel assembly are described in Section VIII-B.

Figure 2 Pressurized Water Loop Location in GETR

Figure 4 Post-Irradiation Appearance of Rod 12A, Assembly EPT-12C

Figure 5. Pressurized Water Loop Heat Transfer Regime

VI. FUEL ASSEMBLY DESIGN

The fuel assembly design for these irradiations is shown in Figure 6. All the assembly hardware is fabricated from Type-304 stainless steel. The assembly is installed in the PWL facility tube with the lifting tee upward. The assembly rests on, and is supported by, a 45-degree annular seat built into the facility tube 8 inches (20 cm) below the bottom of the GETR* core. The assembly was designed so that the bottom end of the fuel column within the test fuel rods is at the same elevation as the bottom of the GETR core when the fuel rods are at a uniform temperature of 350°F (177°C). Subsequent analysis and measurements have shown that the bottom end of the fuel rod end plug is at essentially the same elevation as the bottom of the GETR core while operating and the fuel column is 1 inch (2.5 cm) above this elevation. Coolant flow enters from above the assembly, and exerts a downward force that holds the assembly in position. In addition, there is a spring-loaded, hold-down rod between the assembly lifting tee and the facility tube closure plug that exerts additional downward force and positively prevents assembly rotation during operation. Approximately 95% of the coolant flow passes down through the fuel rod annuli, while the remaining 5% is bypassed around the outside of the assembly to cool the externally insulated, facility tube wall.

Each of the four fuel rods is contained in its own concentric flow tube to maximize flow velocity and to minimize flow variations between rods and around the rod periphery. Lateral spacing between the fuel rod and the shroud tube is maintained by five sets of spacers located 9 inches (22 cm) apart along the length of the shroud tube. Each set of spacers contains four spacer pins projecting inward from the wall of the shroud tube at 90-degree circumferential spacing between individual pins. The radial orientation of each spacer pin set is rotated 45 degrees from the set immediately above. Each individual spacer pin is a hairpin loop, formed from 0.072-inch (0.18 cm) stainless steel wire, and inserted in the shroud tube wall from the inside. The plane of the hairpin loop is parallel to the flowstream. The diametral clearance between the fuel rod and the spacer pins is nominally 10 mils (0.025 cm) with a tolerance of plus 5 mils (0.013 cm) minus 0 mil.

The shroud tubes are bowed inward approximately 70 mils (0.18 cm) toward the assembly centerline at their axial midpoint. The fuel rods when initially installed assume this same bow and are thus forced in relatively tight contact with at least one spacer pin in each set. The objective of the above arrangement was to prevent vibration-induced wear or fretting attack at the spacer bearing points on the clad. Despite the very high flow velocity and the previous history of vibration problems in PWL, (22) this spacing arrangement has been extremely satisfactory. In the absence of fuel rod dimensional changes, the spacer bearing points on cladding are barely detectable. The most evident spacer bearing point on the EPT-12 assembly rods after 55 days of operation is also shown in Figure 4. The appearance suggests a slight indentation of the cladding without wear or fracture of the initial Zircaloy oxide film.

* General Electric Test Reactor

Each fuel rod is supported in its shroud tube by a crosspin through the upper end plug. The rod is free to expand axially down from this point. The crosspin meshes with the notches in the upper end of the flow tubes. The inside ends of the crosspins from all four rods are locked in place by a double locknut on the assembly lifting stud. This arrangement permits the removal of the individual fuel rods with ease and rapidity. All four rods are freed by loosening and backing off the two locknuts. An individual rod may then be removed by simply lifting it 1/8-inch (0.3 cm) to clear the notches in the shroud tube, rotating it 90 degrees to clear the crosspin from beneath the locknut, and then lifting it straight up the remaining distance.

A second, shorter crosspin is included in the lower fuel rod end plug as a safety precaution. This pin prevents the fuel rod from escaping through the lower end of the shroud tube in the unlikely event that the upper crosspin failed. The presence of this lower crosspin necessitates slight rotation of the fuel rod during insertion and withdrawal to clear the sets of lateral spacing pins. The filter on the outlet end of the assembly is also a safety precaution and is intended to trap large particles of fuel or cladding in the event of a gross fuel rod failure. The filter is removable and is basically a cone formed from 10-mesh, stainless steel wire cloth. The general appearance of the EPT-6 assembly before irradiation is shown in Figure 7, and is typical of all subsequent assemblies.

1559-1

Figure 6 Fuel Assembly Design

V17520
TOP OR INLET END
SHOWING LIFTING TEE

V17524
FUEL ELEMENT TOP END
PLUG AND LOCKING PIN

V17523

V17521
RESTRAINING BAND
AROUND FLOW TUBES

V17522
NOSE PIECE WITH SCREEN
AT THE BOTTOM OR EXIT END

Figure 7 Pre-Irradiation Appearance - Assembly EPT-6

VII. FUEL ROD DESIGN AND FABRICATION

A. GENERAL

The fuel rod design used in these irradiations is shown in Figure 8. The rod is unsegmented and is 39 inches (99 cm) in over-all length. The cladding is nominally 0.565-inch (1.433 cm) o. d. by 0.030-inch (0.076 cm) wall Zircaloy-2. The fuel column is nominally 34 inches (86 cm) long and is composed of approximately sixty five, 0.5-inch (1.27 cm) o. d. by 0.5-inch (1.27 cm) high, sintered UO_2 pellets. The unfueled length at the upper end of the rod serves as a fission gas plenum and contains a compression spring to prevent fuel column separation during handling. The over-all fuel column length was decreased [to a minimum of 29-1/2 inches (75 cm)] in the higher thermal performance fuel rods to obtain additional fission gas plenum volume. The need for additional plenum volume was based on the possibility of irradiating these higher thermal performance rods to a very high burnup level, i. e., approximately 20,000 MWD/ T_e (5.4×10^{20} fissions/ cm^3). The plenum is designed to accommodate 100% fission gas release with a maximum internal pressure of 1000 psi ($7 \times 10^5 \text{ kg/m}^2$) at end of life.

The UO_2 sintered pellets were fabricated of ceramic grade powder converted from UF_6 by the Davison Chemical Company. The original UF_6 gas was obtained at the precise enrichment required for the experiments. The analytical results supplied by the vendor for the UO_2 powder are listed in Table 1. The as-received UO_2 powder was ball milled for 10 minutes while adding 1-1/2% wax binder and then blended for 5 minutes to add 0.2% Sterotex die lubricant. The powder was then cold pressed to a green pellet density of 5.9 gm/cc with a die size allowance for 18% shrinkage during sintering. The green pellets were dewaxed in hydrogen for 4 hours at 450 C and then sintered in hydrogen for 4 hours at 1600°C. After sintering, densities were determined for three pellets from each sintering boat of approximately 50 pellets by using the water immersion method. The pellets were then centerless ground with water cooling to the desired final o. d. and dried for 2 hours at 100°C after grinding. The end faces of the pellets were not ground. The maximum variation in pellet height around the periphery was approximately 6 mils (1.5 cm). All four enrichments of UO_2 pellets were processed at the same time under identical conditions. The analytical results obtained on sample pellets from each UO_2 enrichment are listed in Table 2.

The Zircaloy-2 tubing used for the fuel rod cladding was fabricated by Reactive Metals and received in the annealed condition. The physical and chemical properties of the tubing as-received are listed in Table 3. The tubing was 100% ultrasonically inspected, cleaned in HF-HNO_3 solution and autoclaved in steam at 1500 psi ($1.05 \times 10^6 \text{ kg/m}^2$) and 750°F (400°C for 24 hours. The tubing was inspected dimensionally and the lower end plug welded in place under an inert atmosphere. The end plugs were machined from Zircaloy-2 bar stock. After

TABLE 1. Chemical Analysis - UO₂ Ceramic Grade Powder

Element	Impurity Content, ppm			
	Enrichment %			
	1.5	2.2	3.0	3.8
Al	2	<2	<2	<2
B	0.20	<0.20	0.21	<0.20
Cd	<0.10	<0.10	<0.10	<0.10
Ca	26	33	50	72
C	43	55	65	25
F	49	50	42	20
Fe	10	12	22	14
Si	4	13	12	12
Cl	<25	<25	<25	<25
Cr	3	12	14	20
Cu	21	32	10	21
Mn	3	<1	<1	1
N	<10	<10	11	21
V	<10 *	<10*	<10	<10
Ti	<100*	<100*	<100 *	<100*
Co	<5	<5	5	<5
Mo	<1 *	<1 *	<1*	<1 *
Ni	10	<10	11	12
Ag	<0.10	0.15	<0.10	0.15

O:U Ratio	2.07	2.07	2.09	2.05
Particle Size, μ (Fisher Sub-Sieve)	0.93	1.04	0.90	0.86
Bulk Density, gm/cc	1.14	1.19	1.20	1.17
U ²³⁵ , %	1.497	2.183	3.014	3.836

* No Determination

TABLE 2 Chemical Analysis - UO₂ Sintered PelletsImpurity Content, ppm
Enrichment, %

Element	1 5%		2 2%					3.0%					3 8%				
Al	100	20	200	200	100	100	100	30	50	50	500	10	50	20	20	200	500
Bi	1	<1	-	-	-	-	-	20	20	20	20	20	2	2	2	-	-
B	<0 2	<0 2	<0 2	<0 2	<0 2	<0 2	<0 2	0 5	<0 2	<0 2	0 2	<0 2	<0 2	0 5	<0 2	<0 2	<0 2
Cd	<1	<1	<1	<1	<1	<1	<1	1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Ca	10	10	10	10	10	10	10	5	5	10	10	5	10	10	10	10	10
Cr	25	25	20	20	20	20	20	25	25	25	50	50	50	50	50	20	20
Co	<2	<2	<2	<2	<2	<2	<2	<2	15	2	<2	<2	<2	<2	<2	<2	<2
Cu	65	65	100	100	100	200	200	10	20	20	20	30	100	50	50	100	100
Fe	500	200	1000	1000	500	1000	1000	500	500	1000	700	500	1000	1000	1000	1000	1000
Pb	<1	<1	<1	<1	<1	<1	<1	2	5	5	2	5	<1	<1	<1	<1	<1
Mg	5	5	5	10	5	10	10	5	5	10	10	5	10	10	10	5	5
Mn	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	1	<1	1	1	1	<1	<1
Mo	10	<3	<3	<3	<3	<3	10	<3	20	<3	<3	<3	<3	<3	10	<3	10
Si	30	30	200	300	60	60	100	60	30	30	20	30	60	100	60	60	60
Ag	3	10	1	3	5	0 1	5	1	1	1	1	4	3	4	2	0 5	1
Na	<35	<35	<35	<35	35	<35	<35	<35	<35	<35	<35	<35	<35	<35	<35	<35	<35
Sn	5	10	5	5	5	5	5	10	5	5	5	5	30	20	20	5	5
V	<15	<15	<15	<15	<15	<15	<15	15	<15	<15	<15	<15	<15	<15	<15	<15	<15
Zn	<10	<10	<10	<10	<10	<10	<10	<10	<10	<10	<10	<10	10	10	10	10	10
Ni	100	50	1000	200	100	100	50	50	50	50	20	100	100	100	100	100	50
Sb	2	<2	<2	2	<2	<2	<2	<2	<2	<2	2	<2	<2	<2	<2	<2	2
N	160	390	-	-	-	-	-	12	26	24	26	49	70	70	70	70	70
C	<10	<10	30	<10	10	20	<10	<10	-	-	-	-	-	-	<10	<10	<10
F	16	18	28	18	19	15	22	-	-	-	-	-	15	2	7	10	4
O U Ratio	2 03 2 01, 2 00		2 01	2 00	2 00	2 00	2 00	94 1 to 97 3**	2 01				2 00			2 01	2 01
Density, %	94 1 to 97 3*		94 0 to 96 6 TDS**										94 0 to 97 3**				
U ²³⁵ %	1 46 ± 0 04		2 26, 2 20, 2 27					3.0 ± .04 3 04	3 03		3 02 3 01		3 84	3 90	3 84		
	1 50 ± 0 04																

* Based on 15 pellets

** Based on 25 pellets

TABLE 3. Physical-Chemical Properties, Annealed Zircaloy Tubing

Ingot No. 28601, Lot No. 494

Corrosion Weight Gain - 15.8 and 15.5 mg/dm² in 3 days at 750°F, 1500 psi steam (399°C, 605 × 10⁶ kg/m².)

Chemical Analysis:

<u>Alloy Elements</u>	<u>%</u>
Sn	1.53
Fe	0.132
Cr	0.103
Ni	0.054
Fe + Cr + Ni	0.289

<u>Impurity, ppm</u>		<u>Impurity, ppm</u>		<u>Impurity, ppm</u>	
Al	29	Hf	79	V	< 20
B	< 0.3	Mg	<20	W	< 20
C	172	Mn	<20	U	< 0.2
Ca	< 20	Mo	<20		
Cd	< 20	Na	<20	H	17
Cl	< 20	Pb	<20	N	23
Co	< 5	Si	35	O	910
Cu	< 20	Ti	<20		

Longitudinal Mechanical Properties:

Ultimate Tensile Strength, ksi (kg/m ²)	84.1 (59.0 × 10 ⁶)	81.1 (57.0 × 10 ⁶)
Yield Strength (0.2% offset) ksi, (kg/m ²)	65.0 (45.6 × 10 ⁶)	63.9 (44.8 × 10 ⁶)
Elongation in 2 inches (5 cm), %	16.5	17.7
Hardness (Rockwell B)	92.0/93.0	91.5/93.0
Grain Size:	ASTM 6-7	
Flare Test:	33.6% and 34.4% o.d. expansion at 60 degrees	
Burst Test:	10,500-psi yield pressure, (7.4 × 10 ⁶ kg/m ²)	
	11,000-psi burst pressure, (7.7 × 10 ⁶ kg/m ²)

Figure 8. Fuel Rod Design

welding in the lower end plugs, they were stamped with the rod identification numbers, the cladding was cut to exact length, and the residual clad sections stored for possible analysis.

The UO_2 pellets for loading into the fuel rods were individually selected. Any visible cracks or significant corner chips were considered sufficient cause for rejecting the pellets. After loading the pellets into the cladding, the stacked fuel column height and plenum length were measured. The loaded fuel rods were then placed in a vacuum retort and heated slowly in vacuum to 900°F (480°C) to drive off residual gas in the pellets. The rods were backfilled with helium and re-evacuated six times at temperature to ensure complete outgassing by dilution. After the final evacuation, the retort containing the fuel rods was cooled to room temperature, backfilled with helium at 1 atm pressure, and transferred to the weld chamber without re-exposing the fuel rods to the atmosphere.

The plenum compression spring and upper end plug were fitted in the rods under the weld chamber inert atmosphere, and the final seal welding performed. The rods were mass spectrometer leak checked and x-rayed. Over-all x-ray pictures of the rod were taken for inspecting the fuel column condition, i.e., no fuel pellet separation or breakage. In addition, detailed x-rays were made of upper and lower end plugs at 0 and 90 degree orientations to inspect the welds. These x-rays were made through steel collimating blocks to improve resolution.

As the final step in fabrication, the rods were cleaned in acetone and loaded into the assembly hardware. The assembly was transferred to the Radioactive Material Laboratory (RML) for pre-irradiation visual inspection and dimensional examination.

In Table 4, the fabrication data are summarized for each of the fuel rods from the series of assemblies discussed. All the pellet fuel rods were fabricated at the same time before starting the first assembly irradiation.

B. EPT-COMPOSITE ASSEMBLY

The EPT-COMposite assembly was not included in the original program schedule but was a special irradiation test to determine if hollow UO_2 fuel pellets could prevent clad swelling. To expedite the test, the assembly was constructed with the four rods fabricated originally as spares for the initial four pellet fuel assemblies. Thus, each rod contained a different UO_2 enrichment and resulted in the COMposite designation for the assembly.

To refabricate the rods with hollow pellets, the rods were cut open just below the upper end plug weld and all the pellets unloaded. These pellets were then combined with the remaining pellets of the same enrichment from the original sinter. Each pellet was drilled with an 0.080-inch (0.203 cm) diamond burr to provide the central hole. To reduce breakage, the drilling was performed underwater with the pellet held in a special nylon test fixture. The hole

TABLE 4. Fuel Rod Fabrication Data

Assembly	Rod	Enrichment (%)	Cladding Diameter, in.		Pellet Diameter, in.		Average Pellet Height, in.	Fuel Column Length, in.	Fuel Weight, gm	Plenum Length in.	Maximum Heat Flux during Irradiation	Peak Heat Flux At end of Irradiation
			Outside	Inside	Outside	Inside					Btu/h-ft ²	
EPT-6	6A	1.5	0.5647 - 0.5653	0.5058 - 0.5064	0.500 - 0.501	Solid	0.519	33-3/4	1141.2	3/8	630,000	515,000
	6B	1.5	0.5642 - 0.5659	0.5060 - 0.5067	Centerless	Solid		34-5/16	1147.8	2- 1/16	665,000	545,000
	6C	1.5	0.5611 - 0.5638	0.5064 - 0.5068	Ground	Solid		34-1/4	1157.0	2- 3/16	675,000	550,000
	6D	1.5	0.5642 - 0.5654	0.5051 - 0.5056		Solid		34-3/16	1158.2	2-7/8	732,000	600,000
EPT-8	8A	2.2	0.5642 - 0.5652	0.5092 - 0.5106	0.503 + 0.001	Solid	0.518	34-1/4	1139.6	2-3/4	950,000	770,000
	8B	2.2	0.5642 - 0.5668	0.5081 - 0.5091	- 0.000	Solid		34-9/32	1134.8	2-3/4	920,000	745,000
	8C	2.2	0.5645 - 0.5663	0.5091 - 0.5100	Centerless	Solid		34-3/16	1136.2	2-13/16	780,000	631,000
	8D	2.2	0.5635 - 0.5648	0.5092 - 0.5100	Ground	Solid		34-1/4	1135.2	2-3/4	915,000	740,000
EPT-10	10A	3.0	0.5671 - 0.5688	0.5093 - 0.5101	0.504	Solid	0.500 + 0.020 - 0.000	31-1/4	1051.6	5-25/32	1,030,000	960,000
	10B	3.0	0.5675 - 0.5685	0.5094 - 0.5107		Solid		31-3/8	1060.3	5-11/16	1,040,000	965,000
	10C	3.0	0.5663 - 0.5680	0.5100 - 0.5104	Centerless	Solid		31-5/16	1059.8	5-11/16	1,140,000	1,060,000
	10D	3.0	0.5668 - 0.5690	0.5097 - 0.5107	Ground	Solid		31-1/4	1063.8	5-3/4	1,140,000	1,063,000
EPT-COM	6E	1.5	0.5647 - 0.5660	0.5046 - 0.5055	0.500 - 0.501	0.085±0.005	0.500 + 0.020 - 0.000	33-3/4	1119	2-3/4	605,000	505,000
	8E	2.2	0.5635 - 0.5657	0.5092 - 0.5100	0.503 + 0.001	0.085±0.005		34	1097	3	905,000	758,000
	10E	3.0	0.5679 - 0.5688	0.5098 - 0.5109	0.504 - 0.000	0.085±0.005		31-1/8	1018	6	1,000,000	840,000
	12E	3.8	0.5685 - 0.5690	0.5108 - 0.5110	0.5018 - 0.502	0.085±0.005		29-1/2	955	7-1/2	1,148,000	960,000
EPT-12C	12AC	3.8	Not Measured	0.5001 - 0.5006	0.495 - 0.496	0.140+	0.500 + 0.020 - 0.000	29-1/2	891	7-1/2	1,440,000	1,140,000
	12BC	3.8	Before	0.5001 - 0.5018	Centerless	0.140+		29-1/2	892	7-1/2	1,360,000	1,070,000
	12CC	3.8	Fabrication	0.5000 - 0.5009	Ground	0.140+		29-1/2	894	7-1/2	1,210,000	960,000
	12DC	3.8	See Table 11	0.5002 - 0.5009		0.140+		29-3/4	883	7-1/4	1,255,000	990,000

0.565 inch = 1.435 cm

0.500 inch = 1.27 cm

TABLE 5. Physical-Chemical Properties of the EPT-12C
Replacement Zircaloy-2 Tubing.

Ingot No. 34079, Lot No. 1321

Corrosion Weight Gain - 16.6 and 17.8 mg/dm² in 3 days at 750°F, 1500 psi steam
(399°C, 1.05×10^6 kg/m²)

Chemical Analysis:

<u>Alloy Elements</u>	<u>%</u>
Sn	1.53
Fe	0.125
Cr	0.095
Ni	0.056
Fe + Cr + Ni	0.276

Impurity, ppm

Al	46
B	0.3
C	148
Ca	<20
Cd	<0.25
Cl	<20
Co	<5
Cu	<20

Impurity, ppm

Hf	60
Mg	<10
Mn	<20
Mo	<20
Na	<10
Pb	<20
Si	35
Ti	<20

Impurity, ppm

V	<20
W	<20
U	0.28
H	16
N	31
O	1330

Longitudinal Mechanical Properties:

Ultimate Tensile Strength, ksi (kg/m ²)	107.8 (75.8×10^6)	107.2 (75.4×10^6)
Yield Strength (0.2% offset), ksi (kg/m ²)	90.8 (63.8×10^6)	98.3 (69.1×10^6)
Elongation in 2 inches, % (5 cm)	14.0	13.0
Hardness (Rockwell B)	97.5/98.0	97.0/98.0
Grain Size:	ASTM 7	
Flare Test:	27.5% and 30.5% o.d. expansion at 60 degrees	
Burst Test:	10,500 psi yield pressure (7.4×10^6 kg/m ²)	
	11,750 psi burst pressure (8.8×10^6 kg/m ²)	

was drilled halfway through from one end, the pellet inverted, and the hole completed from the other end to reduce spalling on the end faces. The measured inside diameter of the finished holes averaged about 0.085 inch (0.216 cm). After drilling, the pellets were dried and then selectively loaded into the original rod cladding. The original compression spring and disc were added and the remainder of the fabrication performed following the exact procedure and the same inspections used initially. New upper end plugs were used. The decrease in over-all length of the rods after refabrication did not exceed 1/8 inch (0.32 cm).

C. EPT-12C ASSEMBLY

The EPT-12C assembly was refabricated with hollow UO_2 fuel pellets from the fuel rods in the original EPT-12 fuel assembly. The refabrication procedure was identical to that followed for the EPT-COM assembly with two specific exceptions:

1. The hole size drilled in the EPT-12C fuel pellets was increased to a nominal 0.140-inch (0.354 cm). The diamond burrs used were 0.140-inch (0.354 cm) and produced a hole with a diameter range between 0.140 and 0.150 inch (0.354 and 0.380 cm).
2. During the high temperature outgassing step, the original EPT-12 assembly annealed cladding was damaged by overheating. This cladding was replaced with 1/4-hard Zircaloy-2 tubing to meet the required irradiation schedule. It was also necessary to centerless grind the UO_2 pellets to a slightly smaller outside diameter to maintain the same initial gap in the new cladding. The replacement cladding is from the same lot of tubing used to fabricate the vibratory compacted UO_2 fuel rods under this program. The tubing is equal in all respects to the original tubing with the exception of the cold work condition. In Table 5, the physical and chemical properties of the EPT-12C replacement tubing are listed.

VIII. ASSEMBLY IRRADIATION HISTORIES

A. ASSEMBLY EPT-6

The fuel rods in assembly EPT-6 contained 1.5% enriched UO_2 fuel pellets and were intended to operate at a nominal peak heat flux of $600,000 \text{ Btu/h-ft}^2$ (190 W/cm^2). The following actual conditions obtained at initial, 30-MW reactor power operation were:

Startup Date - 4/23/62
 GETR Cycle No. - 33
 Control Rod Bank Position - 19.3-inches (49 cm)
 Assembly Average, Peak Heat Flux - $680,000 \text{ Btu/h-ft}^2$ (214 W/cm^2)
 Rod 6A Peak Surface Heat Flux - $628,000 \text{ Btu/h-ft}^2$ (198 W/cm^2)
 Rod 6D Peak Surface Heat Flux - $734,000 \text{ Btu/h-ft}^2$ (232 W/cm^2)

The coolant conditions during the EPT-6 assembly irradiation are listed in Table 6. The assembly operated satisfactorily for the entire 25-day GETR cycle and experienced 6 full power cycles. The average thermal power of the assembly during operation was 200 kW. The GETR was refueled on May 10, 1962, which lowered the control rod bank position from 28.4 inches (72 cm) before refueling to 25.6 inches (65 cm) after refueling. Figure 9 shows the chronological variation in the average heat flux of rod 6D during the cycle. The following conditions prevailed at 30 MW reactor power immediately before shutdown:

Shutdown date - 5/19/62
 Control Rod Bank Position - 31.5 inches (80 cm)
 Assembly Average, Peak Heat Flux - $560,000 \text{ Btu/h-ft}^2$ (177 W/cm^2)
 Rod 6D Peak Heat Flux - $605,000 \text{ Btu/h-ft}^2$ (191 W/cm^2)
 Assembly Average Burnup - 1250 MWD/T_e (0.34×10^{20} fissions/ cm^3)

B. ASSEMBLY EPT-8

The fuel rods in assembly EPT-8 contained 2.2% enriched UO_2 fuel pellets and were intended to operate at a nominal peak heat flux of $800,000 \text{ Btu/h-ft}^2$ (252 W/cm^2). The actual conditions at initial, 30-MW reactor power operation were:

Startup Date - 6/1/62
 GETR Cycle No. - 34
 Control Rod Bank Position - 18.0 inches (46 cm)
 Assembly Average Peak Heat Flux - $890,000 \text{ Btu/h-ft}^2$ (281 W/cm^2)
 Rod 8A Peak Heat Flux - $950,000 \text{ Btu/h-ft}^2$ (300 W/cm^2)

TABLE 6. Specific Coolant Conditions, Pellet Fuel Assembly Irradiations

Fuel Rod Coolant Annulus Conditions					
Assembly	Flow gpm (l/sec)	Mass Velocity		Film Heat Transfer Coefficient*	Loop Pressure
		(lb/h-ft ²)	($\frac{\text{kg}}{\text{m}^2\text{h}}$)	(Btu/h-ft ² -°F) (kg-Cal/h-m ² -°C)	
EPT-6	57 (3.6)	3.22×10^6	(15.7×10^6)	7,100 (34.7×10^3)	16 (4.9) 1000 (7.03×10^5)
EPT-8	83 (5.2)	4.69×10^6	(22.9×10^6)	9,700 (47.4×10^3)	23 (7.0) 1000 (7.03×10^5)
EPT-10	92 (5.8)	5.20×10^6	(25.4×10^6)	10,500 (51.3×10^3)	26 (7.9) 1000 (7.03×10^5)
EPT-COM	120 (7.6)	6.78×10^6	(33.1×10^6)	13,500 (65.9×10^3)	34 (10.4) 1200 (8.44×10^5)
EPT-12C	120 (7.6)	6.78×10^6	(33.1×10^6)	13,500 (65.9×10^3)	34 (10.4) 1200 (8.44×10^5)

Coolant Properties at 344° F Inlet Temperature:

Density (ρ) 56 lb/ft³ (897 kg/m³)

Specific Heat (Cp) 1.05 Btu/lb-°F (1.05 gm-cal/gm-°C)

* R. P. Stein and W. Begell, "Heat Transfer to Water in Turbulent Flow in Internally Heated Annuli," A. I. Ch. E. Journal 4, 2, 127 (June 1958).

The coolant conditions during the EPT-8 assembly irradiation are listed in Table 6. Operation was uneventful for the entire 24-day reactor cycle with only 2 power cycles and no core refueling. The average thermal power of the assembly during operation was 260 kW. The chronological variation in the average heat flux of rod 8A is shown in Figure 10. The following conditions prevailed at 30 MW-reactor power, 1-1/2 hours before shutdown:

Shutdown Date - 6/24/62

Control Rod Bank Position - 27.5 inches (70 cm)

Assembly Average, Peak Heat Flux - 706,000 Btu/h-ft² (223 W/cm²)Rod 8A Peak Heat Flux - 754,000 Btu/h-ft² (238 W/cm²)Assembly Average Burnup - 1532 MWD/T_e (0.41×10^{20} fissions/cm³)

Figure 9. Thermal Performance History - Rod 6D, Assembly EPT-6

Figure 10. Thermal Performance History - Rod 8A, Assembly EPT-8

ASSEMBLY EPT-10

The fuel rods in assembly EPT-10 contained 3.0% enriched UO_2 fuel pellets and were intended to operate at a nominal peak heat flux of 1×10^6 Btu/h-ft² (316 W/cm²). The actual conditions at initial, 27-MW reactor power operation were:

Startup Date - 7/3/62
 GETR Cycle No. - 35
 Control Rod Bank Position - 18.5 inches (47 cm)
 Assembly Average, Peak Heat Flux - 1,032,000 Btu/h-ft² (326 W/cm²)
 Rod 10D Peak Heat Flux - 1,082,000 Btu/h-ft² (342 W/cm²)

The coolant conditions during the EPT-10 assembly irradiation are listed in Table 6. The assembly operated satisfactorily for 16 days and experienced eight reactor power cycles. After 16 days of operation, a slight burst of fission gas activity was observed in the loop coolant. Although the activity died out shortly after it occurred, the decision was made to remove the assembly from the loop, primarily on the basis of the results from the post-irradiation examinations of the previous two assemblies. The average thermal power of the assembly during operation was 310 kW. Figure 11 shows the chronological variation in the average heat flux of rod 10D. The conditions existing immediately before shutdown were:

Shutdown Date - 7/19/62
 Control Rod Bank Position - 24.4 inches (62 cm)
 Assembly Average, Peak Heat Flux - 1,010,000 Btu/h-ft² (320 W/cm²)
 Rod 10D Peak Heat Flux - 1,063,000 Btu/h-ft² (336 W/cm²)
 Assembly Average Burnup - 1460 MWD/T_e (0.39×10^{20} fissions/cm³)

ASSEMBLY EPT-COMPOSITE

The EPT-COMposite assembly contained four fuel rods with four different enrichments. The rotational orientation of the assembly was carefully controlled during installation so that the nominal peak heat fluxes of the 1.5, 2.2, 3.0, and 3.8% enriched fuel rods at startup would be 600,000, 900,000, 1,000,000, and 1,100,000 Btu/h-ft² (190, 284, 316, and 348 W/cm²), respectively. To prevent clad swelling, the fuel rods were filled with hollow UO_2 fuel pellets. To separate the effect of the initial cycle of melting on clad swelling from the subsequent effect of repetitive power cycling, the assembly was irradiated twice and examined between the two irradiations.

Figure 11. Thermal Performance History - Rod 10D, Assembly EPT-10

The maximum actual operating conditions attained during the initial cycle of operation at 29-MW reactor power were:

Startup Date - 9/15/62
 GETR Cycle No. - 37
 Control Rod Bank Position - 17.7-inches (45 cm)
 Assembly Average, Peak Heat Flux - 985,000 Btu/h-ft² (311 W/cm²)
 Rod 12E Peak Heat Flux - 1,148,000 Btu/h-ft² (362 W/cm²)

After 1.5 hours at maximum power, and approximately 21 hours of intermediate power level operation during startup, the reactor was scrammed and the assembly removed for examination. The maximum thermal power of the assembly during this initial irradiation was 292 kW.

After being examined, the assembly was returned to the loop for one full cycle of irradiation. The conditions at initial 30-MW reactor operation were:

Startup Date - 11/25/62
 GETR Cycle No. - 39
 Control Rod Bank Position - 19.2-inches (49 cm)
 Attained 30 MW Reactor Power - 11/27/62
 Assembly Average, Peak Heat Flux - 970,000 Btu/h-ft² (306 W/cm²)
 Rod 12E Peak Heat Flux - 1,130,000 Btu/h-ft² (357 W/cm²)

The coolant conditions during both irradiations of the EPT COM assembly are listed in Table 6. The assembly operated satisfactorily for the remaining 19 days of cycle 39. There were four power cycles during this second irradiation but no core refueling was required. Figure 12 shows the chronological variation of the average heat flux for rod 12E during irradiation.

The conditions existing immediately before shutdown were:

Shutdown Date - 12/16/62
 Control Rod Bank Position - 28.5-inches (725 cm)
 Assembly Average, Peak Heat Flux - 832,000 Btu/h-ft² (263 W/cm²)
 Rod 12E Peak Heat Flux - 970,000 Btu/h-ft² (306 W/cm²)
 Assembly Average Burnup - 1725 MWD/T_e (0.46×10^{20} fissions/cm³)

Figure 12. Thermal Performance History - Rod 12E. Assembly EPT-COM

The two startups on the EPT-COM assembly were the first irradiations for which the reactor power rise was programmed. The specified procedure was less restrictive than that subsequently used with the EPT-12C assembly and required only that 2 hours of steady-state operation follow each 5-MW reactor power increment.

E. EPT-12C

The fuel rods in the EPT-12C assembly contained 3.8% enriched, hollow UO_2 fuel pellets and were intended to operate at a nominal peak heat flux of 1.2×10^6 Btu/h-ft² (379 W/cm²). The first cycle of irradiation started during GETR cycle 40 on January 29, 1963, with full power being reached on January 31, 1963. The reactor power rise rate during this startup was very carefully controlled and limited to a 1-MW power increment every 15 minutes. This same procedure was followed for all startups after reactor refueling. At the start of reactor full-power operation, the control rod bank position was 17.3-inches (44 cm) and the assembly average, peak heat flux was 1.27×10^6 Btu/h-ft² (401 W/cm²). The peak heat flux of the maximum power rod (rod 12A) was 1.38×10^6 Btu/h-ft² (436 W/cm²). The assembly operated satisfactorily for the remaining 22 days of the cycle and attained a burnup of 2240 MWD/T_e (0.6×10^{20} fissions/cm³). During the startup and the course of the cycle, the assembly experienced 20 power cycles, of which 12 were scrams from full power followed by rapid recovery.

The reactor was refueled once during the cycle necessitating an additional programmed startup. The reactor control rod bank following refueling was 17.1-inches (43.5 cm). The control rod bank before the refueling was indeterminate because of a series of scrams and recoveries immediately preceding the refueling. However, a representative value would be 20.0-inches (51cm). At the end of the cycle the control rod bank was withdrawn to 27.0-inches (68.5 cm). The average thermal power of the assembly during operation was 360 kW. Just before shutdown on February 22, 1963, the assembly average, peak heat flux was 1.00×10^6 Btu/h-ft² (316 W/cm²) and the peak heat flux of rod 12A was 1.09×10^6 Btu/h-ft² (344 W/cm²).

The EPT-12C assembly remained in the loop for another cycle of irradiation during GETR cycle 41. The startup conditions were:

Startup Date - 3/3/63

Full Power Operation - 3/4/63

Control Rod Bank at Full Power - 20.4-inches (52 cm)

Assembly Average, Peak Surface Heat Flux - 1.22×10^6 Btu/h-ft² (386 W/cm²)

Peak Heat Flux of Rod 12A - 1.33×10^6 Btu/h-ft² (420 W/cm²)

The assembly experienced one scram from full power during this cycle which brought the total power cycles for the assembly to 22. The average thermal power during operation was 350 kW and the burnup accumulated during this second cycle was 3100 MWD/ T_e (0.84×10^{20} fissions/cm³). At shutdown the following conditions prevailed:

Shutdown Date - 3/31/63

Control Rod Bank Position - 35.4-inches (90 cm)

Assembly Average, Peak Heat Flux - 882,000 Btu/h-ft² (279 W/cm²)

Rod 12A Peak Heat Flux - 964,000 Btu/h-ft² (304 W/cm²)

Accumulated Average Burnup - 5340 MWD/ T_e (1.43×10^{20} fissions/cm³)

Following this second cycle of irradiation, the assembly was removed from the loop for an interim, nondestructive examination. The assembly was returned to the loop for continued irradiation during GETR cycle 45. Startup began at 0300 on June 19, 1963. The assembly failed at 1208 during the startup at a reactor power level of 27 MW. The conditions at the time of failure were.

Control Rod Bank Position - 18.4-inches (47 cm)

Assembly Average, Peak Surface Heat Flux - 1.043×10^6 Btu/h-ft² (330 W/cm²)

Rod 12A Peak Heat Flux - 1.14×10^6 Btu/h-ft² (360 W/cm²)

Figures 13, 13a, and 13b show the chronological variation in power of rod 12D during its life. Review of the loop operating records have indicated that the requested reactor power rise schedule was not followed completely during the startup and the assembly was subjected to an approximate 4-MW step increase in power between 21 and 25-MW reactor power. This deviation is considered the probable cause for the clad swelling described in Section II-E.

On the basis of the post-irradiation examination and conclusions on the probable cause of failure, assembly EPT-12C has been rebuilt and redesignated EPT-12C-A. The new assembly contains two rods from the original EPT-12C assembly (rods 12B and 12C) and two new hollow pellet fuel rods essentially identical to those being replaced. This rebuilt assembly will be reinserted in the PWL and the irradiation continued to high burnup.

Figure 13. Thermal Performance History - Rod 12D, Assembly EPT-12C, Cycle 41

Figure 13a. Thermal Performance History - Rod 12D, Assembly EPT-12C, Cycle 42

Figure 13b. Thermal Performance History - Rod 12D, Assembly EPT-12C, Cycle 45

IX. POST-IRRADIATION EXAMINATION

A. GENERAL

Despite considerable effort, the point of fission gas release during the EPT-10 assembly irradiation has never been located. Only a single burst of noble gas activity occurred and then decayed before the reactor shutdown. No fission product activity was detected in radio-chemical analyses of the loop coolant, of water samples taken from the fuel assembly transfer can, and of solvent-soaked swabs of each fuel rod cladding. Rod 10D was mass spectrometer leak checked after it was punctured for a fission gas sample; the results were negative. Subsequent detailed examination of rod 10D, has provided some evidence that this rod had defected during irradiation and was the probable source of the fission gas burst.

A detailed description of the procedures and techniques used in the post-irradiation examination of these fuel rods is provided in the Appendices.

B. VISUAL AND DIMENSIONAL EXAMINATION

1. EPT-6

When assembly EPT-6 was initially examined in the RML, only rod 6A was easily removed from its shroud tube. After considerable physical effort, rod 6B was withdrawn from its shroud tube but the rod surface was badly scarred by the lateral spacers in the process. Neither rod 6C or 6D could be withdrawn and it was necessary to slit their shroud tubes full length to effect removal. Subsequent dimensional examination revealed severe clad swelling in rods 6C and 6D. Rod 6B was slightly swollen and 6A was just perceptibly swollen. The dimensional increase on 6A was less than the experimental measurement accuracy and may not be significant. Figure 14 illustrates the diameter changes in the EPT-6 assembly fuel rods as a function of axial position. The dark shaded areas on the plots shown in Figure 14, indicate the differences in the diameter measurements at 0 and 90 degree rotational orientations of the fuel rods. It is evident that the clad swelling is almost perfectly symmetrical and is confined to a very short, 4-inch (10 cm) axial section of the rods, centered at a position approximately 9 inches (23 cm) above the bottom end of the fuel rod. This axial position corresponds to the peak heat flux location at startup which indicates clad swelling occurred at startup.

The maximum diameter increase occurred in rod 6D and was 47 mils (0.12 cm) or an approximately 8% circumferential elongation of the cladding. Based on its integrated gamma activity determined during gamma scanning, rod 6D was also the maximum power rod in the assembly. This result coupled with the axial location of the swelling indicated strongly that the clad swelling was related to thermal performance level of the fuel rod. In Table 7, the pre- and post-irradiation measurements on assembly EPT-6 are summarized.

TABLE 7. Assembly EPT-6 Pre- and Post-Irradiation Dimensions.

Fuel Rod	Axial Distance from top (inches)	Outside Diameter in inches at 0-180 deg. Orientation			Outside Diameter in inches at 90-270 deg. Orientation		
		Irradiation			Irradiation		
		Pre-	Post-	Change, (mils)	Pre-	Post-	Change, (mils)
6-A	4	0.565*	0.565	0	0.565	0.563	-2
	16	0.564	0.564	0	0.564	0.563	-1
	20	0.564	0.564	0	0.564	0.563	-1
	24	0.564	0.564	0	0.564	0.564	0
	28	0.564	0.566	+2	0.564	0.565	+1
	32	0.565	0.565	0	0.564	0.566	+2
	36	0.565	0.565	0	0.565	0.564	-1
6-B	4	0.564	0.563	-1	0.565	0.563	-2
	16	0.564	0.563	-1	0.563	0.562	-1
	20	0.563	0.564	+1	0.564	0.563	-1
	24	0.564	0.564	0	0.564	0.564	0
	28	0.564	0.567	+3	0.564	0.568	+4
	32	0.564	0.570	+6	0.564	0.569	+5
	36	0.564	0.564	0	0.565	0.563	-2
6-C	4	0.563	0.561	-2	0.563	0.561	-2
	16	0.561	0.560	-1	0.561	0.561	0
	20	0.560	0.560	0	0.560	0.559	-1
	24	0.560	0.562	+2	0.560	0.561	+1
	28	0.561	0.564	+3	0.560	0.564	+4
	30	-	0.605	-	-	0.605	-
	32	0.561	0.578	+17	0.561	0.575	+14
	34	-	0.563	-	-	0.564	-
	36	0.562	0.562	0	0.562	0.561	-1
6-D	4	0.564	0.563	-1	0.565	0.563	-2
	16	0.563	0.562	-1	0.563	0.562	-1
	20	0.563	0.563	0	0.564	0.563	-1
	24	0.563	0.564	+1	0.564	0.565	+1
	28	0.563	0.567	+4	0.564	0.568	+4
	30	-	0.579	-	-	0.581	-
	31	-	0.610	-	-	0.611	-
	32	0.564	0.572	+8	0.564	0.573	+9
	34	-	0.563	-	-	0.564	-
	36	0.564	0.564	0	0.564	0.563	-1
Over-All Length inches **							
6-A		39.018	39.126	+0.108			
6-B		39.008	39.145	+0.137			
6-C		39.006	39.086	+0.080			
6-D		39.004	39.062	+0.058			

* 0.565 inch = 1.435 cm

** Accuracy of length ± 0.010 inch

The visual appearance of the rods 6C and 6D was generally identical to the pre-irradiation condition except at the location of the swelling. At this location, the rod surfaces were badly scored from the lateral spacers during the initial attempt to remove them from their shroud tubes. Rod 6D was selected for destructive examination because it was the maximum power rod and also the rod with the maximum diameter increase. Subsequently, rod 6A was also destructively examined for comparison.

2. EPT-8

None of the fuel rods in assembly EPT-8 could be removed without slitting their shroud tubes. The results of the dimensional examination showed that all four fuel rods had experienced significant clad swelling. Figure 15 shows the dimensional results for the EPT-8 fuel rods; this plot is similar to that given in Figure 14 for EPT-6. The dimensional data on EPT-8 are summarized in Table 8.

It is again evident that the clad swelling is almost perfectly symmetrical and occurs at the peak heat flux location at startup. The maximum diameter increase observed in the maximum power rod, rod 8A, is 53 mils (0.135 cm). This diameter increase is only slightly larger than that observed in rod 6D which operated at a significantly lower peak heat flux. However, the fuel rod length over which swelling occurred is generally longer than that in the EPT-6 rods. This observation indicated that the fuel rod volume increase from clad swelling was a more significant parameter than the maximum diameter increase and the volume increase was used for subsequent correlation of results on fuel rod swelling. Rod 8A, the maximum power rod, was selected for destructive examination.

3. EPT-10

All four shroud tubes in assembly EPT-10 had to be slit to remove the fuel rods. Dimensional examination revealed extreme clad swelling in all four rods with a significantly larger equivalent volume increases than the EPT-8 rods. A maximum diameter increase of 74 mils (0.188 cm), equivalent to a circumferential clad elongation of 13%, was observed in the maximum power rod, 10D. This elongation is within the range of the ultimate circumferential elongation usually observed for Zircaloy-2 tubing in the annealed condition.⁽²³⁾ It seems likely that the clad was very close to the fracture point. Figure 16 illustrates the diameter changes observed on the EPT-10 fuel rods. The profiles are very similar to those obtained from the previous assemblies except for the larger diameter increases over a greater length of the fuel rod. The swelling is still very symmetrical and peaked at the location corresponding to peak heat flux at startup. The one peculiarity in these dimensional data is the distinct depressions in profiles at a position 29 inches (74 cm) below the top of rods 10A, 10B, and 10C. This location corresponds to the axial position of the fourth set of lateral spacers on the shroud tube.

TABLE 8. Assembly EPT-8 Pre- and Post-Irradiation Dimensions

Fuel Rod	Axial Distance from top (inches)	Outside Diameter in inches at 0-180 deg. Orientation			Outside Diameter in inches at 90-270 deg. Orientation		
		Irradiation Pre-	Post-	Change, (mils)	Irradiation Pre-	Post-	Change, (mils)
8-A	4	0.564	0.563	-1	0.564	0.563	-1
	16	0.563	0.562	-1	0.563	0.562	-1
	20	0.564	0.565	+1	0.563	0.562	-1
	24	0.563	-	-	0.563	-	-
	26	-	0.567	-	-	0.567	-
	27	-	0.572	-	-	0.572	-
	28	0.564	-	-	0.563	-	-
	29	-	0.581	-	-	0.581	-
	30	-	0.610	-	-	0.610	-
	31	-	0.592	-	-	0.594	-
	32	0.563	0.576	+13	0.563	0.575	+12
	34	-	0.571	-	-	0.571	-
	36	0.564	-	-	0.564	-	-
8-B	4	0.565	0.565	0	0.564	0.565	+1
	16	0.564	0.563	-1	0.564	0.564	0
	20	0.563	0.563	0	0.563	0.563	0
	24	0.564	0.566	+2	0.563	0.564	+1
	25	-	0.568	-	-	0.566	-
	26	-	0.571	-	-	0.567	-
	27	-	0.579	-	-	0.578	-
	28	0.564	0.588	+24	0.563	0.591	+28
	29	-	0.581	-	-	0.581	-
	30	-	0.583	-	-	0.585	-
	31	-	0.575	-	-	0.576	-
	32	0.564	0.572	+8	0.564	0.570	+6
	33	-	0.567	-	-	0.570	-
	34	-	0.566	-	-	0.565	-
	35	-	0.566	-	-	0.565	-
	36	0.565	0.565	0	0.564	0.565	+1
8-C	4	0.565	0.563	-2	0.566	0.563	-3
	16	0.564	0.563	-1	0.565	0.563	-2
	20	0.564	0.563	-1	0.564	0.564	0
	24	0.564	0.565	+1	0.564	0.563	-1
	28	0.564	0.571	+7	0.564	0.571	+7
	29	-	0.573	-	-	-	-
	30	-	0.577	-	-	0.573	-
	31	-	0.571	-	-	0.579	-
	32	0.565	0.571	+6	0.564	0.570	+6
	33	-	0.568	-	-	0.567	-
	34	-	0.564	-	-	0.565	-
	35	-	-	-	-	0.565	-
	36	0.565	0.564	-1	0.566	0.564	-2

TABLE 8. (Continued)

Fuel Rod	Axial Distance from top (inches)	Outside Diameter in inches at 0-180 deg. Orientation			Outside Diameter in inches at 90-270 deg. Orientation		
		Irradiation		Change, (mils)	Irradiation		Change, (mils)
		Pre-	Post-		Pre-	Post-	
8-D	4	0.563	0.567	+4	0.564	0.567	+3
	16	0.563	0.562	-1	0.563	0.563	0
	20	0.563	0.564	+1	0.563	0.567	+4
	24	0.563	0.564	+1	0.563	0.563	0
	28	0.563	0.583	+20	0.563	0.581	+18
	29	-	0.580	-	-	0.578	-
	30	-	0.583	-	-	0.580	-
	31	-	0.573	-	-	0.576	-
	32	0.563	0.571	+8	0.563	0.573	+10
	33	-	0.565	-	-	0.567	-
	34	-	0.567	-	-	0.564	-
	35	-	0.565	-	-	0.563	-
	36	0.563	0.564	+1	0.564	0.563	-1

Over-All Length (inches)*

8-A	39.014	39.071	+0.057
8-B	39.020	39.082	+0.062
8-C	39.031	39.087	+0.056
8-D	39.032	39.086	+0.054

* Accuracy of length ± 0.010 inch

Figure 14. Dimensional Changes - Assembly EPT-6

Figure 15. Dimensional Changes - Assembly EPT-8

As a result, the depressions in the swelling profiles are considered to be caused by restraint imposed by the spacer pins at this point. Rod 10D was chosen for destructive examination. The dimensional data for the EPT-10 assembly are listed in Table 9.

4. EPT-COM

After the initial cycle of irradiation for 1.5 hours at full power, three of the four fuel rods in assembly EPT-COM were easily removed from their shroud tubes. Dimensional examination indicated no detectable clad swelling in rods 6E, 8E, and 12E. The fourth rod, 10E, could not be removed but was quite free to move axially within the shroud tube. Since it was planned to re-irradiate the assembly, extreme care was exercised not to exert any significant force on rod 10E to prevent damage to the clad. However, it seemed very probable that rod 10E could have been removed by applying some force and that the maximum diameter increase of this rod was only slightly greater than 10 mils. No dimensional measurements could be obtained. Since both rods 10E and 12E operated at thermal performance levels equivalent to the EPT-10 assembly, the complete absence of swelling in rod 12E and slight swelling in rod 10E indicated that the use of hollow pellets had been effective, at least for a single cycle of melting.

After the second, full reactor cycle of irradiation, the EPT-COM assembly was again examined and essentially the same results were obtained. Rods 6E, 8E, and 12E again showed no detectable clad swelling, despite four additional power cycles. This time rod 10E was forcefully removed from its shroud tube and dimensional measurements obtained. The dimensional profile of rod 10E indicates a maximum clad diameter increase of 13 mils (0.033 cm). The dimensional data for the EPT-COM assembly fuel rods are given in Table 10. Thus, the clad swelling was just slightly greater than that necessary to prevent easy removal of the fuel rod. The maximum increase in clad swelling that could have occurred from power cycling during the second irradiation was less than 3 mils (0.008 cm). Since Rods 6E, 8E, and 12E did not exhibit any change during the second irradiation, it seems probable that all the swelling in rod 10E also occurred during the first melting cycle. On this basis, power cycling does not appear to promote clad swelling, at least for the few, rapid power cycles experienced by the EPT-COM assembly.

The results of physics analyses and post-irradiation measurements indicate that rod 12E was the maximum power rod in the assembly; the swelling in rod 10E rather than 12E is unexplained. The slight swelling in rod 10E does indicate that the free volume provided by the 0.085-inch (0.216 cm) diameter central hole was slightly insufficient for the thermal performance level and associated volume of UO_2 melting.

Figure 16. Dimensional Changes - Assembly EPT-10

TABLE 9. Assembly EPT-10 Pre- and Post-Irradiation Dimensions

Fuel Rod	Axial Distance from top (inches)	Outside Diameter in inches at 0-180 deg. Orientation			Outside Diameter in inches at 90-270 deg. Orientation		
		Irradiation		Change, (mils)	Irradiation		Change, (mils)
		Pre-	Post-		Pre-	Post-	
10-A	4	0.568	0.567	-1	0.568	0.567	-1
	16	0.568	0.566	-2	0.568	0.566	-2
	20	0.567	0.566	-1	0.567	0.565	-2
	24	0.567	0.570	+3	0.567	0.571	+4
	25	-	0.573	-	-	0.573	-
	26	-	0.576	-	-	0.575	-
	27	-	0.578	-	-	0.577	-
	28	0.568	0.581	+13	0.568	0.580	+12
	29	-	0.575	-	-	0.575	-
	30	-	0.580	-	-	0.578	-
	31	-	0.609	-	-	0.609	-
	32	0.568	0.615	+47	0.568	0.615	+47
	33	-	0.589	-	-	0.596	-
	34	-	0.575	-	-	0.576	-
	35	-	0.570	-	-	0.571	-
	36	0.568	0.568	0	0.568	0.569	+1
	37	-	0.567	-	-	0.568	-
10-B	4	0.568	0.566	-2	0.568	0.565	-3
	16	0.567	0.566	-1	0.569	0.566	-2
	20	0.567	0.565	-2	0.568	0.565	-3
	24	0.568	0.572	+4	0.568	0.571	+3
	25	-	0.577	-	-	0.575	-
	26	-	0.582	-	-	0.581	-
	27	-	0.590	-	-	0.592	-
	28	0.568	0.592	+24	0.568	0.594	+26
	29	-	0.579	-	-	0.581	-
	30	-	0.586	-	-	0.587	-
	31	-	0.617	-	-	0.616	-
	32	0.567	0.627	+60	0.568	0.624	+56
	33	-	0.595	-	-	0.595	-
	34	-	0.573	-	-	0.574	-
	35	-	0.568	-	-	0.569	-
	36	0.568	0.567	-1	0.568	0.568	0
	37	-	0.568	-	-	0.566	-
	38	-	0.565	-	-	0.565	-
10-C	4	0.567	0.565	-2	0.567	0.565	-2
	16	0.567	0.565	-2	0.566	0.565	-1
	20	0.567	0.566	-1	0.566	0.566	0
	24	0.567	0.568	+1	0.566	0.567	+1
	25	-	0.571	-	-	0.571	-
	26	-	0.576	-	-	0.577	-
	27	-	0.585	-	-	0.586	-
	28	0.567	0.594	+27	0.566	0.593	+27
	29	-	0.593	-	-	0.589	-
	30	-	0.587	-	-	0.588	-
	31	-	0.611	-	-	0.604	-
	32	0.568	0.622	+54	0.567	0.623	+56
	33	-	0.618	-	-	0.621	-

TABLE 9. (Continued)

Fuel Rod	Axial Distance from top (inches)	Outside Diameter in inches at 0-180 deg. Orientation			Outside Diameter in inches at 90-270 deg. Orientation		
		Irradiation		Change, (mils)	Irradiation		Change, (mils)
		Pre-	Post-		Pre-	Post-	
10-C	34	-	0.588	-	-	0.585	-
	35	-	0.572	-	-	0.572	-
	36	0.568	0.570	+2	0.567	0.571	+4
	37	-	0.569	-	-	0.569	-
	38	-	0.566	-	-	0.565	-
10-D	4	0.569	0.566	-3	0.568	0.566	-2
	16	0.568	0.566	-2	0.567	0.565	-2
	20	0.568	0.568	0	0.567	0.566	-1
	24	0.567	0.568	+1	0.567	0.569	+2
	25	-	0.570	-	-	0.570	-
	26	-	0.570	-	-	0.571	-
	27	-	0.573	-	-	0.574	-
	28	0.567	0.574	+7	0.567	0.576	+9
	29	-	0.577	-	-	0.577	-
	30	-	0.590	-	-	0.596	-
	31	-	0.623	-	-	0.623	-
	32	0.568	0.642	+74	0.568	0.641	+73
	33	-	0.628	-	-	0.625	-
	34	-	0.595	-	-	0.592	-
	35	-	0.574	-	-	0.573	-
	36	0.568	0.571	+3	0.568	0.572	+4
	37	-	0.569	-	-	0.569	-
	38	-	0.567	-	-	0.566	-

	Over-All Length (inches)*		
10-A	39.033	39.152	+0.119
10-B	39.012	39.104	+0.092
10-C	39.026	39.120	+0.094
10-D	39.025	39.079	+0.054

* Accuracy of length ± 0.010 inch

TABLE 10. Assembly EPT-COMposite Pre- and Post-Irradiation Dimensions

Fuel Rod	Axial Distance from top (inches)	Outside Diameter in inches at 0-180 deg. Orientation					Outside Diameter in inches at 90-270 deg. Orientation				
		Interim		Final		Change, (mils)	Interim		Final		Change, (mils)
		Irradiation Pre-	Irradiation Post-	Change, (mils)	Post-		Irradiation Pre-	Irradiation Post-	Change, (mils)	Post-	
6-E	4	0.566 *	0.565	-1	0.562	-3	0.565*	0.564	-1	0.562	-2
	16	0.564	0.563	-1	0.562	-1	0.564	0.563	-1	0.563	0
	20	0.564	0.564	-	0.563	-1	0.564	0.563	-1	0.563	0
	24	0.565	0.564	-1	0.564	0	0.564	0.564	0	0.564	0
	28	0.565	0.565	0	0.564	-1	0.565	0.564	-1	0.564	0
	32	0.565	0.566	+1	0.564	-2	0.565	0.564	-1	0.563	-1
	36	0.565	0.565	0	0.564	-1	0.565	0.564	-1	0.563	-1
8-E	4		0.563	-	0.563	0	-	0.564	-	0.564	0
	16	0.5635**	0.563	-	0.564	+1	-	0.563	-	0.564	+1
	20	to	0.561	-	0.562	+1	-	0.563	-	0.563	0
	24	0.5657	0.562	-	0.562	0	0.5646**	0.562	-	0.563	+1
	28		0.562	-	0.563	+1	Avg.	0.564	-	0.563	-1
	32	0.5646	0.563	-	0.563	0	-	0.565	-	0.564	-1
	36	Avg.	0.565	-	0.563	-2	-	0.563	-	0.563	0
10-E	4		***	-	0.566	-	-	***	-	0.566	-
	16		-	-	0.565	-	-	-	-	0.565	-
	20		-	-	0.566	-	-	-	-	0.566	-
	24		-	-	0.566	-	-	-	-	0.566	-
	28		-	-	0.567	-	-	-	-	0.567	-
	29	0.5679**	-	-	0.568	-	0.5684**	-	-	0.568	-
	30	to	-	-	0.569	-	Avg.	-	-	0.568	-
	31	0.5688	-	-	0.571	-	-	-	-	0.572	-
	32		-	-	0.579	-	-	-	-	0.579	-
	33		-	-	0.577	-	-	-	-	0.576	-
	34	0.5684	-	-	0.569	-	-	-	-	0.570	-
	35	Avg.	-	-	0.568	-	-	-	-	0.568	-
	36		-	-	0.567	-	-	-	-	0.566	-

TABLE 10. (Continued)

Fuel Rod	Axial Distance from top (inches)	Outside Diameter in inches at 0-180 deg. Orientation					Outside Diameter in inches at 90-270 deg. Orientation				
		Interim			Final		Interim			Final	
		Irradiation Pre-	Post-	Change, (mils)	Post-	Change, (mils)	Irradiation Pre-	Post-	Change, (mils)	Post-	Change, (mils)
12-E	4		0.566	-	0.565	-1	-	0.567	-	0.566	-1
	16	0.5685**	0.567	-	0.566	-1	-	0.566	-	0.566	0
	20	to	0.567	-	0.566	-1	0.5688**	0.567	-	0.566	-1
	24	0.5690	0.567	-	0.566	-1	Avg.	0.567	-	0.565	-2
	28		0.567	-	0.566	-1	-	0.567	-	0.566	-1
	32	0.5688	0.567	-	0.566	-1	-	0.567	-	0.565	-2
	36	Avg.	0.568	-	0.565	-3	-	0.567	-	0.566	-1

Over-All Length in inches

6-E	38.965	38.991	0.026	39.005	0.040
8-E	38.960	38.970	0.010	38.981	0.021
10-E	38.990	***	-	39.042	0.052
12-E	38.990	39.001	0.011	39.014	0.024

* Dimensions measured prior to removing fuel and coring pellets

** Original Tube Dimensions

*** Fuel element stuck in flow tube; therefore, measurements could not be obtained.

Figure 4-1. Dimensional Profile - Rod 10-E

5. EPT-12C

When initially examined after 5300 MWD/T_e (1.4×10^{20} fissions/cm³) burnup, all four fuel rods from assembly EPT-12C were easily removed from their shrouds. No dimensional changes were detected during the measurements. The visual appearance of the rods was excellent and essentially indistinguishable from the pre-irradiation condition.⁽²⁴⁾ The single observed feature which could be attributed to the irradiation was the spacer bearing mark on rod 12A (Figure 4). Even this mark was slight and appeared to be an indentation, rather than a wear spot. The Zircaloy-2 oxide film was not fractured.

When assembly EPT-12 was returned for examination after failure at the startup of GETR cycle 45, two of the four rods, 12B and 12C, were easily removed from their shroud tubes. Dimensional examination of each rod showed very slight swelling at the peak heat flux zone. The maximum diameter increase in each rod was 2 mils (0.005 cm). Neither rod 12A nor 12D could be removed and the shroud tubes had to be slit to permit examination. Rod 12D was found to be failed with a 1-inch (2.5 cm) long split in the clad at the peak heat flux zone. The uniform diameter increase at the failure was only 15 mils (0.038 cm). The swelling in rod 12A was larger with a maximum diameter increase of 19 mils (0.048 cm). Dimensional profiles for all four rods are shown in Figure 18. In Table 11, the dimensional data for the EPT-12 assembly fuel rods are listed. Rod 12A also appeared to be kinked at the swelling zone. Based on the orientation of the assembly during irradiation, the post-irradiation gamma scans, and the clad swelling results, rod 12A was the maximum power rod in the assembly, even though rod 12D experienced the actual failure. Destructive examination of rod 12D has just been started, therefore, no further results are available on this assembly.

6. Correlation of Clad Swelling Observations

The following correlation of the results on fuel rod clad swelling is confined primarily to the first 3 fuel assemblies. In later assemblies, changes were incorporated to eliminate clad swelling; as a result, the dimensional changes are not representative of those experienced with solid UO₂ fuel pellets.

As shown, the fuel rod volume increase appears to be the most representative measure of the clad swelling phenomenon. This volume increase was calculated on the basis of radial dimensional changes only, and the observed axial expansion of the rods was not included in the calculation. The fuel rod volume increase versus the respective peak thermal performance value at startup, is shown in Figure 19. In this plot, the parameter

$$\int_0^T Q_{cl} \, k dT$$

is employed as the independent variable rather than peak surface heat flux to obtain a linear

TABLE 11. Assembly EPT-12C Pre- and Post-Irradiation Dimensions

Fuel Rod	Axial Distance from top (inches)	Outside Diameter in inches at 0-180 deg. Orientation					Outside Diameter in inches at 90-270 deg. Orientation				
		Interim			Final		Interim			Final	
		Irradiation Pre-	Post-	Change, (mils)	Post-	Change, (mils)	Irradiation Pre-	Post-	Change, (mils)	Post-	Change, (mils)
12-A	4	0.563	0.564	+1	0.563	0	0.562	0.562	0	0.564	+2
	16	0.562	0.561	-1	0.566	+4	0.562	0.561	-1	0.564	+2
	20	0.562	0.561	-1	0.562	0	0.562	0.562	0	0.562	0
	24	0.562	0.562	0	0.562	0	0.562	0.562	0	0.565	+3
	25	-	-	-	0.564	-	-	-	-	0.567	-
	26	-	-	-	0.568	-	-	-	-	0.568	-
	27	-	-	-	0.581	-	-	-	-	0.582	-
	28	0.562	0.563	+1	0.575	+13	0.562	0.563	+1	0.577	+15
	29	-	-	-	0.571	-	-	-	-	0.569	-
	30	-	-	-	0.567	-	-	-	-	0.567	-
	32	0.562	0.562	0	0.566	+4	0.562	0.562	0	0.567	+5
	35	-	-	-	0.566	-	-	-	-	0.563	-
	36	0.563	0.563	0	0.566	+3	0.562	0.563	+1	0.562	0
12-B	4	0.562	0.562	0	0.562	0	0.563	0.562	-1	0.563	0
	16	0.562	0.562	0	0.563	+1	0.562	0.562	0	0.564	+2
	20	0.562	0.562	0	0.563	+1	0.562	0.562	0	0.563	+1
	24	0.562	0.562	0	0.564	+2	0.562	0.561	-1	0.563	+1
	28	0.562	0.562	0	0.564	+2	0.562	0.563	+1	0.563	+1
	32	0.562	0.562	0	0.564	+2	0.562	0.562	0	0.563	+1
	36	0.563	0.563	0	0.563	0	0.563	0.564	+1	0.563	0
12-C	4	0.562	0.562	0	0.562	0	0.562	0.562	0	0.562	0
	16	0.561	0.561	0	0.561	0	0.561	0.562	+1	0.561	0
	20	0.561	0.561	0	0.562	+1	0.561	0.562	+1	0.562	+1
	24	0.561	0.563	+2	0.561	0	0.561	0.561	0	0.561	0
	28	0.562	0.562	0	0.562	0	0.561	0.561	0	0.562	+1
	32	0.562	0.562	0	0.563	+1	0.562	0.562	0	0.562	0
	36	0.562	0.563	+1	0.563	+1	0.562	0.562	0	0.562	0

TABLE 11. (Continued)

Fuel Rod	Axial Distance from top (inches)	Outside Diameter in inches at 0-180 deg. Orientation					Outside Diameter in inches at 90-270 deg. Orientation				
		Interim			Final		Interim			Final	
		Irradiation Pre-	Irradiation Post-	Change, (mils)	Post-	Change, (mils)	Irradiation Pre-	Irradiation Post-	Change, (mils)	Post-	Change, (mils)
12-D	4	0.563	0.562	-1	0.565	+2	0.562	0.562	0	0.563	+1
	16	0.562	0.561	-1	0.561	-1	0.562	0.562	0	0.564	+2
	20	0.562	0.562	0	0.562	0	0.562	0.562	0	0.561	-1
	24	0.561	0.561	0	0.566	+5	0.561	0.562	+1	0.567	+6
	25	-	-	-	0.565	-	-	-	-	0.565	-
	26	-	-	-	0.566	-	-	-	-	0.566	-
	27	-	-	-	0.568	-	-	-	-	0.568	-
	28	0.562	0.562	0	0.566	+4	0.561	0.562	+1	0.568	+7
	29	-	-	-	0.577	-	-	-	-	0.577	-
	30	-	-	-	0.566	-	-	-	-	0.568	-
	31	-	-	-	0.563	-	-	-	-	0.563	-
	32	0.562	0.562	0	0.566	+4	0.562	0.562	0	0.564	+2
	33	-	-	-	0.564	-	-	-	-	0.562	-
	36	0.562	0.562	0	0.563	+1	0.562	0.562	0	0.563	+1
Over-All Length in inches											
12-A		39.032	39.088	+0.056	39.062	+0.030					
12-B		39.036	39.064	+0.028	39.056	+0.020					
12-C		39.035	39.079	+0.044	39.072	+0.037					
12-D		39.045	39.103	+0.058	39.101	+0.056

Figure 18. Dimensional Changes (After Second Irradiation) Assembly EPT-12C

Figure 19. Swelling Results - Pellet Fuel

measure of the UO_2 thermal performance level. The thermal performance level of the UO_2 is not exactly in linear proportion to the heat flux because of enrichment variation in the UO_2 between successive assemblies. From the plot in Figure 19, it is apparent that the radial clad swelling is a threshold type phenomenon and starts at a clad peak surface heat flux value equivalent to a $\int_0^{T_{cl}} k dT$ value of ~ 92 W/cm. As discussed previously, this is the peak surface heat flux based on a uniform diameter rod. The peak surface heat flux may have increased as a result of swelling, but for establishing the exact threshold for clad swelling, the heat flux based on the unswelled diameter is the pertinent parameter.

The threshold thermal performance level for clad swelling is just slightly greater than that predicted for the onset of central melting, i. e., 550,000 Btu/h-ft².^{*} This similarity strongly indicates an association exists between the clad swelling and central melting. The small discrepancy can be easily reconciled on the basis that some finite increment of UO_2 melting must occur before it is manifested by detectable clad expansion.

* See Section XI.

C. DESTRUCTIVE EXAMINATION

In addition to complete a dimensional examination, all four fuel rods from each assembly were gamma scanned and punctured to take a fission gas sample. The fuel rods selected for detailed examination were then cut into 4-inch (10 cm) sections along their length and photographs were taken of each "as-fractured" face. Based on the appearance of the as-fractured sections and the location of the sections relative to areas of interest such as the zone of maximum swelling, peak heat flux position at startup, maximum burnup location, etc., samples were selected for detailed metallographic examination. Generally, two transverse cross sections were prepared from each rod, one at the zone of maximum swelling and one at the zone of maximum burnup. These two sections also correspond approximately to the peak heat flux location at startup and at shutdown, respectively. In addition, a longitudinal section of the fuel at the upper termination central void was prepared where possible. This section is of particular interest and is significant to the analyses to determine the UO_2 molten boundary and thermal conductivity discussed in Section XI.

Three different preparations were made on each metallographic sample. The first was a low magnification (10X) photograph of the sample in the rough-polished condition to show the UO_2 structural changes and yet keep the perspective of the full section. The second was an autoradiograph of the section at the same magnification to indicate the fission product activity distribution and the relationship of this distribution to the UO_2 structure zones in the rough-polished sample. The third preparation is a 100X magnification, radial composite photograph of the fine-polished and etched UO_2 structure. These three preparations have been combined and shown in one illustration for each of the metallographic samples to be discussed.

1. Gamma Scan Results

Based on previous experience with gamma scans from fuel rods without central melting, a normal gamma activity profile is a smooth, regular curve with a shape approximating the average thermal neutron flux profile during irradiation. Rods containing UO_2 pellets usually exhibit slight depressions in the profile at the pellet interfaces and one of the main values of gamma scan has been to locate the individual pellets and, thereby, show the general, post-irradiation condition of the fuel column.

The gamma scan from rod 6D (Figure 20), is a normal profile in most respects. However, the scan indicates that definite pellet separation occurred in the region of maximum clad swelling, 9 inches (23 cm) above the rod bottom. This separation was confirmed by sectioning and metallography. Axial separation of the pellets has not been observed in any other rod examined during the program and the reason for its occurrence in rod 6D is unexplained. It cannot be attributed to clad swelling alone since subsequent rods exhibited greater clad swelling without axial separation.

ROD 6D EPT-6

3

5

6

7

2

CROSS SECTION KEY

8

9

CROSS SECTIONS

Figure 20 Rod 6D Section Properties Assembly EPT 6

MACROSCOPIC SECTION

PEAK HEAT FLUX ZONE (MET. NO. 8561)—ROD 6 D
AXIAL LOCATION 8 INCHES FROM BOTTOM OF ROD (20.4 cm)
PEAK HEAT FLUX DURING LIFE 711,000 Btu h-ft² (224 W cm²)
HEAT FLUX AT END OF LIFE 406,000 Btu h-ft² (128 W cm²)

AUTORADIOGRAPH

RADIAL MICROSCOPIC SECTION, ETCHED

Figure 20a Rod 6D Peak Heat Flux Zone (Metallography No. 8561)

MAXIMUM BURNUP ZONE (MET NO. 8560) - ROD 6D
AXIAL LOCATION 18.3 INCHES FROM BOTTOM OF ROD (46.5 cm)
PEAK HEAT FLUX DURING LIFE 638,000 Btu h-ft² (202 W cm²)
HEAT FLUX AT END OF LIFE 590,000 Btu h-ft² (186 W cm²)

MACROSCOPIC SECTION

AUTORADIOGRAPH

RADIAL MICROSCOPIC SECTION, ETCHED

Figure 20b Rod 6D Maximum Burnup Zone (Metallography No. 8560)

The most unusual feature of the rod 6D gamma scan is the sharp activity spike near the axial midpoint of the rod. This type of spike had not been observed before and it is believed to be an indication of a concentration of fission products in the hottest zone of the rod; probably a concentration within the UO_2 that was molten just before shutdown. When coupled with the autoradiograph results discussed in Section IX-E, there is little doubt that the axial relocation and concentration of fission product activity actually occurs.

The single activity spike in Rod 6D progresses to multiple activity spikes in the gamma scan of rod 8A (Figure 21). Nearly one-half the length of this rod exhibits perturbations from a normal gamma scan profile. The average activity level over this perturbed zone appears to be essentially constant, which is indicative of activity concentration and equilibration over the molten zone. It has been demonstrated both analytically and experimentally that the activity spikes are not the result of radial cracks in the pellets. The gamma energy observed is greater than 0.7 MeV so that self-shielding by the fuel should be slight. Gamma scans taken at different radial orientations of the fuel rod do not differ significantly.

The gamma scan of rod 10D (Figure 22) shows some of the same characteristics as that from Rod 8A. However, the length of the perturbed region is not as large as could be expected considering the very high thermal performance level of rod 10D, probably because of the early termination of the irradiation at a relatively low reactor control rod bank position. The gamma activity level is very high in the maximum swelling region and may reflect the higher power generation in this region caused by severe clad swelling. The gamma scan from the 12E fuel rod shown in Figure 23 more nearly reflects the characteristics that could be projected from the gamma scan of rod 8A.

Figure 24 shows the gamma scan from rod 12D in the EPT-12C assembly after 5300 MWD/ T_e burnup (1.4×10^{20} fissions/cm³). This scan confirms the previous observations concerning the concentration and equilibration of fission products over the molten UO_2 zone. This scan is perfectly flat from a point below the peak heat flux point at startup to a point just slightly below the top of the fuel column. The chemical burnup samples that will be taken along this region in rod 12D should be extremely interesting, particularly a comparison of burnup values based on elements of widely different volatilities. Such a burnup examination is planned, but has not yet been performed.

2. Fission Gas Release Results

All the per cent fission gas release values determined to date are presented in Table 12. Both gas chromatography and K^{85} gamma spectrometer techniques were used to determine these values for each rod. In each case, the values are averages of multiple samples aliquoted from the original gas volume. The per cent release values of duplicated samples obtained by the same analysis technique are in good agreement. However, the values determined by the two different techniques showed some instances of disagreement, notably, the results for rods 8A and 12E. The reason for the discrepancy is unexplained. The K^{85} spectrometry is generally considered the more reliable technique, but neither method is sufficiently unreliable to account for the differences. The problem may well be caused by the method used to aliquote samples and the associated determination of the sample fraction of the original gas volume. An effort is being made to resolve this question.

TABLE 12. Fission Gas Release based on Atoms of Kr⁸⁵

Assembly	Rod	Atoms Produced Based on Calorimetry	Atoms Released		Release, %	
			γ -Spectrometry	Gas Chromatography	γ -Spectrometry	Gas Chromatography
EPT-6	A	0.906×10^{19}	1.05×10^{18}	1.13×10^{18}	11.5	12.5
	B	0.963×10^{19}	3.00×10^{18}	2.56×10^{18}	31.2	26.5
	C	0.973×10^{19}	2.99×10^{18}	2.90×10^{18}	30.7	29.8
	D	1.058×10^{19}	- -	- -	- -	- -
EPT-8	A	1.253×10^{19}	- -	- -	- -	- -
	B	1.241×10^{19}	3.97×10^{18}	5.15×10^{18}	32.0	41.5
	C	1.045×10^{19}	2.51×10^{18}	3.27×10^{18}	24.0	31.3
	D	1.225×10^{19}	3.70×10^{18}	4.68×10^{18}	30.2	38.2
EPT-10	A	0.992×10^{19}	4.75×10^{18}	4.42×10^{18}	47.8	44.5
	B	0.992×10^{19}	5.09×10^{18}	4.60×10^{18}	51.3	46.5
	C	1.088×10^{19}	5.10×10^{18}	5.85×10^{18}	46.8	53.8
	D*	1.090×10^{19}	2.87×10^{18}	4.34×10^{18}	26.3	39.8
EPT-COM	6E	0.920×10^{19}	1.12×10^{18}	1.23×10^{18}	12.2	13.3
	8E	1.425×10^{19}	5.60×10^{18}	4.95×10^{18}	39.3	34.7
	10E	1.319×10^{19}	6.16×10^{18}	7.16×10^{18}	46.8	54.3
	12E	1.384×10^{19}	- -	8.24×10^{18}	- -	59.5
EPT-12C	A	3.920×10^{19}	24.30×10^{18}	24.54×10^{18}	62.1	62.6

* Suspected Failure

Figure 21. Rod 8A Sectioning Profiles. Assembly EPT-8

MACROSCOPIC SECTION

PEAK HEAT FLUX ZONE (MET. NO. 8596) - ROD 8A
AXIAL LOCATION: 9.6 INCHES FROM BOTTOM OF ROD (24.4 cm)
PEAK HEAT FLUX DURING LIFE: 944,500 Btu h-ft² (298 W cm²)
HEAT FLUX AT END OF LIFE: 969,600 Btu h-ft² (220 W cm²)

AUTORADIOGRAPH

RADIUS OF MELTING $\alpha = .523$ →

RADIAL MICROSCOPIC SECTION, ETCHED

Figure 21a. Rod 8A Peak Heat Flux Zone (Metallography No. 8596)

MACROSCOPIC SECTION

MAXIMUM BURNUP ZONE (MET. NO. 8597) - ROD 8A
AXIAL LOCATION: 17.9 INCHES FROM BOTTOM OF ROD (45.5 cm)
PEAK HEAT FLUX DURING LIFE: 784,700 Btu h-ft² (248 W cm²)
HEAT FLUX AT END OF LIFE: 753,700 Btu h-ft² (238 W cm²)

AUTORADIOGRAPH

RADIAL MICROSCOPIC SECTION, ETCHED

Figure 21b. Rod 8A Maximum Burnup Zone (Metallography No. 8597)

MACROSCOPIC SECTION

ROD 8A EPT-8

DIAM. OF MELTING
 $\alpha = .293$

AUTORADIOGRAPH

TOP OF CENTRAL VOID (MET. NO 8600) - ROD 8A
AXIAL LOCATION: 26.8 - 28.0 INCHES FROM BOTTOM OF ROD (68-71 cm)
PEAK HEAT FLUX DURING LIFE: 479,600 Btu/h-ft² (151 W/cm²)
HEAT FLUX AT END OF LIFE: 479,600 Btu/h-ft² (151 W/cm²)

MICROSCOPIC DIAMETRAL SECTION

Figure 21c. Rod 8A Top of Central Void (Metallography No. 8600)

CROSS SECTIONS

Figure 22 Rod 10D Sectioning Profiles Assembly EPT-10

MACROSCOPIC SECTION

PEAK HEAT FLUX ZONE (MET. NO. 8661) - ROD 10D
 AXIAL LOCATION: 7.0 INCHES FROM BOTTOM OF ROD
 PEAK HEAT FLUX DURING LIFE: 1,031,500 Btu/h-ft²
 HEAT FLUX AT END OF LIFE: 930,600 Btu/h-ft²

AUTOGRAPH

RADIAL MICROSCOPIC SECTION, ETCHED

Figure 22a. Rod 10D Peak Heat Flux Zone (Metallography No. 8661)

MACROSCOPIC SECTION

MAXIMUM BURNUP ZONE (MET. No. 8662) ROD 10D
 AXIAL LOCATION: 13.2 INCHES FROM BOTTOM OF ROD (33.6 cm)
 PEAK HEAT FLUX DURING LIFE: 1,126,300 Btu/h-ft² (356 W/cm²)
 HEAT FLUX AT END OF LIFE: 1,063,500 Btu/h-ft² (336 W/cm²)

AUTORADIOGRAPH

DIAM OF MELTING
 $\alpha = .673$

RADIAL MICROSCOPIC SECTION, ETCHED

Figure 22b. Rod 10D Maximum Burnup Zone (Metallography No. 8662)

MACROSCOPIC SECTION

DIAM. OF MELTING
 $\alpha = .666$

LONGITUDINAL SECTION (MET. No. 8664) ROD 10 D
AXIAL LOCATION: 19.17 - 20.27 INCHES FROM BOTTOM OF ROD (48.5 - 51.5 cm)
PEAK HEAT FLUX DURING LIFE: 883,700 Btu/h-ft² (279 W/cm²)
HEAT FLUX AT END OF LIFE: 883,700 Btu/h-ft² (279 W/cm²)

RADIUS OF MELTING $\alpha = .641$

RADIAL MICROSCOPIC SECTION, ETCHED

Figure 22c. Rod 10D Longitudinal Section (Metallography No. 8664)

Figure 23 Rod 12E Sectioning Profiles Assembly EPT COM

MACROSCOPIC SECTION

PEAK HEAT FLUX ZONE (MET. No. 8713) ROD 12E
 AXIAL LOCATION: 5.6 INCHES FROM BOTTOM OF ROD (14.2 cm)
 PEAK HEAT FLUX DURING LIFE: 1,022,300 Btu/h-ft² (323 W/cm²)
 HEAT FLUX AT END OF LIFE: 659,000 Btu/h-ft² (208 W/cm²)

AUTORADIOGRAPH

RADIUS OF MELTING $\alpha = .339$ →

RADIAL MICROSCOPIC SECTION, ETCHED

Figure 23a. Rod 12E Peak Heat Flux Zone (Metallography No. 8713)

Figure 23b. Rod 12E Maximum Burnup Zone (Metallography No. 8714)

MACROSCOPIC SECTION

DIAM. OF MELTING
 $\alpha = .515$

AUTORADIOGRAPH

TOP OF CENTRAL VOID (MET. NO. 8748) ROD 12E
AXIAL LOCATION: 28.6 - 29.25 INCHES FROM BOTTOM OF ROD
PEAK HEAT FLUX DURING LIFE: 558,700 Btu/h-ft²
HEAT FLUX AT END OF LIFE: 558,700 Btu/h-ft²

RADIUS OF MELTING $\alpha = .451$

RADIAL MICROSCOPIC SECTION, ETCHED

Figure 23c. Rod 12E Top of Central Void (Metallography No. 8748)

Figure 24 Rod 12D Sectioning Profiles Assembly EPT-12C

MACROSCOPIC SECTION

PEAK HEAT FLUX ZONE (MET No. 9911) ROD 12D
 AXIAL LOCATION: 9.4 INCHES FROM BOTTOM OF ROD (23.9 cm)
 PEAK HEAT FLUX DURING LIFE: 1,245,000 Btu h-ft² (304 W cm²)
 HEAT FLUX AT END OF LIFE: 985,000 Btu h-ft² (244 W cm²)

AUTORADIOGRAPH

MICROSCOPIC DIAMETRAL SECTION, ETCHED

Figure 24a. Rod 12D Peak Heat Flux Zone (Metallography No. 8811)

MACROSCOPIC SECTION

MAXIMUM BURNUP ZONE MET No. 8810) ROD 12D
 AXIAL LOCATION 18.5 INCHES FROM BOTTOM OF ROD (47 cm)
 PEAK HEAT FLUX DURING LIFE 965,000 Btu h-ft² (392 W cm²)
 HEAT FLUX AT END OF LIFE 685,000 Btu h-ft² (328 W cm²)

AUTORADIOGRAPH

MICROSCOPIC DIAMETRAL SECTION ETCHED

Figure 24b Rod 12D Maximum Burnup Zone (Metallography No. 8810)

Figure 25 Fission Gas Release Results

Very little correlation analysis has been attempted on the fission gas data. Figure 25 shows the per cent release values versus the fuel rod average surface heat flux at end of life. This parameter was selected as the simplest, most convenient measure of the average UO_2 thermal performance level and internal temperature distribution at the time considered most significant in influencing fission gas release. A trend to increasing fission gas release with increased thermal performance level is evident in Figure 25, particularly if the data from rod 10D, a suspected leaker, are neglected. However, the rate of increase is low and provides a relatively flat curve over an extremely large range in thermal performance.

Considering the fission gas release data as a whole, the per cent release values are generally lower than could have been expected based on earlier measurements of fuel rods without center melting. In Figure 26 the data are replotted on the same curve with the results obtained by Spalaris and Megerth. (25) The saturation or leveling off of the per cent release values is readily apparent. (Note that the independent variable in Figure 26 is slightly different than that

Figure 26. Comparison of Fission Gas Release Data with Previous Results

in Figure 25 to make a correct comparison with the referenced data. For fuel rods with appreciable central melting, the average surface heat flux at end of life rather than the average surface heat flux over the entire irradiation period is considered the more influential parameter relative to the post-irradiation release measurement.) The maximum release value observed to date has been ~62%. As a possible explanation, it may very well be that the post-irradiation puncturing technique for measuring fission gas release is unrealistic and not applicable for fuel rods that operated with appreciable central melting. Examination of the post-irradiation UO_2 microstructure in the molten region shows considerable closed porosity. The pores could be the result of fission gas entrapment during solidification. This gas would not be measured by the post-irradiation puncture method and the consequent release value would be low, i. e., unrepresentative of the actual release at operating conditions. An out-of-pile experiment has been started to determine the extent xenon and krypton can be trapped during UO_2 solidification.

The fission gas release sample from rod 10D provided the first evidence that this rod may have failed under irradiation. The gas chromatography results indicated that the major constituent, > 90%, of this gas sample was hydrogen. No other gas sample had shown more than a trace quantity of hydrogen. It seems likely that the hydrogen was produced by disassociation of water that entered the rod through a defect.

3. Fuel rod Sectioning Results

Profiles of the "as fractured" cross sections for each fuel rod destructively examined are shown in Figures 20, 21, 22, 23, and 24. The axial position of each cross section is indicated by a number key above the fuel rod diagram on each figure. On the basis of these cross sections, the diameter and length of the central void was estimated and is shown on the fuel rod diagram. The cross sections shown on the figures represent less than one-half of the total cross section photos taken on each rod.

As the initial step in a very detailed analysis of each fuel rod, all the "as fractured" cross sections have been examined to determine the radial location of three distinguishable UO_2 microstructure boundaries, i. e., the radii for initial grain growth, for melting, and for the central void. These boundaries are plotted as function of axial position, with the fuel rod dimensional results, on Figures 27, 28, 29, and 30. The location of these boundaries coupled with a knowledge of the thermal performance conditions at each position should permit a determination of the UO_2 thermal conductivity and pellet-to-clad interface conductivity during operation. This analysis work is still in preliminary stages.

It has already been demonstrated that the radius for observable grain growth determined from a "fracture" cross section is, on the average, a very accurate measure of the exact position for the onset of grain growth.⁽¹⁰⁾ By its very nature, the determination of the central void radius from fractured sections should also be accurate. However, the ability to distinguish the precise UO_2 molten boundary in a fracture cross section is not well established and requires further work. Inter-comparison of the fracture cross sections with detailed metallography prepared at the same location has provided a general guide to the appearance of molten and non-

Figure 27. Fracture Section Boundaries - Axial Profiles. Rod 6D

Figure 28. Fracture Section Boundaries - Axial Profiles, Rod 8A

Figure 29. Fracture Section Boundaries - Axial Profiles, Rod ACP

Figure 30. Fracture Section Boundaries - Axial Profiles, Rod 12E

molten UO_2 structures in fractured sections. The difference is best illustrated by reference to the sections. The UO_2 that was molten at the time of shutdown has a very dark, glossy appearance in a fracture section such as the central region of section 5, in Figure 22. The surface is very similar to that of a piece of glass that has been fractured. On the other hand, nonmolten UO_2 is generally lighter in color, more grainy in texture, and may show radial orientation such as Section I, Figure 21. The opinion expressed earlier that central melting occurred in the X-1-e-3 failure is based on the presence of the dark, glossy, central zone in the as fractured section.⁽¹⁾

The characteristic central void distribution in these fuel rods, particularly that shown in Figures 28, 29, and 30, is quite unusual and appears to be typical of fuel rods that have operated with appreciable central melting. The central void exhibits its maximum diameter very near the upper end of the molten zone within the rod (Figure 21C). From this point, the central void tapers down until its diameter is extremely small at the axial location where the peak heat flux at startup occurred. The taper is not completely regular, and bridges of UO_2 are observed across the void at some points. However, the over-all effect is that of a long, slender, inverted cone, very similar to the shrinkage pipe in a casting. This void distribution contrasts sharply with that observed in fuel rods without central melting where the central void is usually symmetrical about the peak power position with a diameter roughly proportional to the axial power profile. The unique shape of the central void in these rods is believed to be the result of molten UO_2 flowing down to fill the lower end as the UO_2 shrinks after irradiation. Calculations based on the most recent expansion data for UO_2 ^(7, 8) indicate that the central void should be approximately filled by the melting and expanding UO_2 when the fuel rods are operated at full power.

It is important to note the following two points. First, the maximum diameter of the central void is a considerable distance above the rod peak heat flux location at the end of the irradiation. As a result, UO_2 vaporized from hotter regions of the rod would tend to deposit at the location where the central void diameter is the largest. The fact that the UO_2 did not build up at this point indicates that the inner UO_2 surface temperature of the void was above the melting point and that the condensed UO_2 simply flowed back down the inner surface. Second, the UO_2 near the upper end of the rods reached its maximum thermal performance level at the very end of the irradiation. Because of the shift in neutron flux profile (Appendices A and B), the upper end of the rods operated at relatively low thermal performance levels at startup, and then gradually and continuously increased in thermal performance as the reactor cycle progressed. For rod 8A, the startup thermal performance level at the axial position corresponding to the upper end of the central void was significantly below that required to produce grain growth. Thus, during the early part of the irradiation there was considerable time for columnar grains to form by mechanisms other than melting. Columnar grains did form as can be seen from the axial metallographic sample taken from rod 8A, Figure 21C, (region immediately above the central void termination). Based on these two points, the diameter of the central void at the upper end of the fuel rod was expected to be very definitive with regard to the extent of melting and UO_2 thermal conductivity in the presence of columnar grains. The central void diameter represents the minimum diameter

of melting at that section. The diameter of melting may actually be larger since there is the possibility that some finite thickness of molten UO_2 will adhere to the unmelted inside surface without flowing down. However, the diameter of melting is not less than the central void diameter. Based on this observation, considerable attention and weight was attached to the central void diameter at the upper end of the rod (Section XI).

D. METALLOGRAPHIC EXAMINATION

1. Observations Related to Clad Swelling

Figures 20A, 21A, and 22A show the metallographic results for sections of rods 6D, 8A, and 10D, each taken through the point of maximum clad swelling. It is apparent that the UO_2 entirely fills the cladding despite the very large increases in clad diameter. This evidence strongly indicates that internal UO_2 movement outward, generated the clad swelling. Very severe radial cracking of the UO_2 is apparent around the outer periphery in all three sections, commensurate with the large diameter increase and lack of plasticity in the cold outer zone of the pellets. However, the inward ends of the cracks appear to have been healed by sintering. This healing and the correspondence of the swelling region to the peak heat flux zone at startup, indicate that the cracking and associated clad swelling occurred early in the irradiation, probably during startup.

The association between the severe, radial pellet cracking and clad swelling is confirmed by the cross sections shown in Figures 20B, 21B, and 22B. These cross sections were taken through the maximum burnup zones of rods 6D, 8A, and 10D, respectively, and also correspond very closely to location of fuel rod peak heat flux at shutdown. There was no appreciable clad swelling at the location of any of these sections, and radial cracking of the pellets is almost negligible. Similarly, both the peak heat flux at startup and maximum burnup cross sections from rod 12E in Figures 23A and 23B show negligible radial cracking. Rod 12E did not experience clad swelling despite being operated at very high heat flux.

2. Interpretation of UO_2 Microstructures

Consider first the cross section from the peak swelling zone of rod 6D in Figure 20A. There are only three zones of distinctly different UO_2 structure in the cross section. Starting inward radially from the outer surface of the pellets, there is an annular band of UO_2 with fine, equiaxed grain structure, unchanged from the as-sintered condition. The radial cracks in the pellets terminate at the inward edge of this band as shown in Figure 31. Adjacent to this outer band is a very narrow, annular zone of equiaxed UO_2 that has undergone grain growth. Finally, there is a very large diameter zone of fine, needle-like columnar grains extending all the way in to the edge of the very small central void. This sequence of UO_2 structures is evident in either the polished cross section or the magnified radial composite (Figure 20A).

V8561-25 ETCHED 500X
AREA A. RESIDUAL POROSITY FROM INCOMPLETE SINTERING ACROSS CRACK

V8561-24 ETCHED 100X
Figure 31. Radial Cracks Healed by Sintering

This UO_2 microstructure is considered to be characteristic of UO_2 that was operating at temperatures below the melting point. It has been demonstrated in out-of-pile annealing experiments with UO_2 that the same type of fine, needle-like columnar grains can be produced by heating to temperatures below the melting point in a suitable temperature gradient.⁽²⁶⁾ Similar annealing experiments with UO_2 in this program have confirmed this result.⁽²⁷⁾ Thus, there is no evidence of molten UO_2 in the microstructure from the startup peak heat flux zone of rod 6D despite the very high, peak heat flux attained at startup. The presence of this microstructure does not preclude the presence of molten UO_2 during the early part of the irradiation. It simply means that none of the cross section was molten when the fuel rod irradiation was terminated.

Turning to the maximum burnup cross section from rod 6D (Figure 20B) the same inward progression in UO_2 structure is apparent. However, in this cross section a new structure zone appears within the zone of fine, needle-like columnar grains. The transition to the new structure occurs at a point approximately one-quarter of the pellet radius and is a change from very fine, needle-like columnar grains to very wide grains which are only slightly columnar. Examination of the UO_2 microstructure in the radial composite (Figure 20B) indicates that the porosity distribution in the UO_2 also begins to change at approximately the point of transition to the larger grains. Moving inward radially from the grain size transition point, the UO_2 pores appear to coalesce, increase in size, and become almost spherical. Further inward, the pore density appears to increase and then, finally, the pores abruptly terminate at a point some distance out from the central void boundary. The remaining, annular band of UO_2 around the central void is essentially 100 % dense, pore-free, UO_2 . This same porosity distribution has been observed in previous UO_2 High Performance Program capsule irradiation experiments,^(9, 10) whether or not a central void was present. The pore-free UO_2 surrounding the central void is considered to be UO_2 that was molten during operation and the sharp boundary defined by the disappearance of pores to be the molten boundary.

3. Identification of the UO_2 Molten Boundary

The out-of-pile annealing experiments conducted under this program and referred to earlier,⁽²⁷⁾ were performed to study the relationship of UO_2 grain size to annealing time and temperature. High-density, sintered UO_2 pellets were heated isothermally to precisely controlled temperatures and held at temperature for varying lengths of time. Post-annealing metallography on the UO_2 samples indicated that in samples heated to temperatures up to 2400°C, the UO_2 porosity remained finely distributed and appeared essentially unchanged from the as-sintered condition. However, in samples heated to 2580°C, the porosity appearance changed markedly. The pores seemed to have coalesced, expanded in size, and became almost spherical in shape. Finally, in samples heated above the UO_2 melting point to 2850°C, porosity completely disappeared and the UO_2 became essentially 100% dense. This sequence of porosity changes observed in the out-of-pile samples is shown in Figure 32. The correspondence between these porosity changes and those observed in the maximum burnup cross section metallography of rod 6D (Figure 20B) is

Figure 32. UO_2 Porosity Changes with Operating Temperature

obvious and constitutes the first part of the evidence for the previous identification of the molten boundary.

As a result of this evidence, the metallography from all of the rods has been carefully examined to confirm the boundary identification. Of particular interest were the axial cross sections taken at the upper end of the central void. As noted previously, the central void at the upper end of these rods is believed to be the direct result of central melting (after columnar grains were formed) and not UO_2 vaporization or void migration as is possible in other locations. The most significant metallographic cross section in this respect is that taken from rod 8A (Figure 21C). It can be seen in the UO_2 microstructure that the boundary of the central void just grazes the edge of the pore termination boundary. This is the largest diameter central void, in terms of its distance out into the UO_2 microstructure, that has been observed in these rods. Other physically larger central voids have been observed, e.g., the upper end of the central void in rod 12E (Figure 23C). However, here the central void boundary is separated from the porosity boundary by a layer of pore-free UO_2 . As indicated earlier, the result obtained from these cross sections does not indicate that the molten boundary is at a smaller radius than the pore termination boundary, but only that it is not smaller than the observed central void boundary. The pore-free UO_2 undoubtedly is molten UO_2 that adhered to the unmelted surface. This adhered layer of molten UO_2 is also apparent in the cross section of rod 8A, both above and below the point where the central void boundary grazes the pore termination boundary. Based on this evidence, the pore termination boundary is believed to correspond to the UO_2 molten boundary during operation. This same molten boundary selection is also supported by the earlier results from the short-time capsule irradiations sponsored by this Program. This boundary identification is used in subsequent analysis work to determine the integral UO_2 thermal conductivity

$$\left(\int_0^{T_m} k dT \right)$$

during operation.

4. Comparison of the Molten Boundary Identification with Previous Results

Horn and Christensen (28) have recently published the results of an experiment specifically designed to determine the location of the UO_2 molten boundary. Tungsten wire markers were arranged in a spiral pattern within a sintered UO_2 pellet as shown in Figure 33. This pellet was then irradiated at a thermal performance level sufficient to produce considerable central melting. After irradiation, the pellet was sectioned transversely and a polished metallographic specimen prepared of the UO_2 cross section. The post-irradiation cross section is also shown in Figure 33. It was expected in this experiment that those wires within the molten UO_2 zone would sink to the bottom of the pellet and would no longer be observed. The radially innermost wire remaining in its original location would then indicate UO_2 that had almost, but not quite, reached the melting point.

Figure 33. Fuel Cross Sections - Tungsten Wire Experiment

Examination of the microstructure in Figure 33 indicates the same inward progression in UO_2 structure changes as previously described for fuel rods from this Program. It is evident that the innermost remaining wire lies in the zone of enlarged, almost spherical porosity, just slightly outward, radially, from the porosity termination boundary.

An annular band of essentially 100% dense, pore-free UO_2 is inside the porosity termination boundary. However, further inward in this cross section a new structure zone appears. This zone contains very large, radially elongated pores and a radially oriented, relatively fine sub-grain structure. The outer boundary of this new zone is very well defined and is perfectly circular. The pre-irradiation location of the next tungsten wire further inward along the spiral was within this new UO_2 structure zone; however, the wire itself was missing in the post-irradiation cross section. On this basis, Horn and Christensen⁽²⁸⁾ selected the outer radius of this pore-containing structure as the molten boundary during operation as indicated on Figure 46. Robertson⁽¹⁸⁾ had indicated earlier the same boundary selection in a UO_2 microstructure as did Horn and Christensen.⁽²⁸⁾ It is evident that this boundary is slightly inward radially from the pore termination boundary we have identified as the UO_2 molten boundary.

From this discussion, it should be apparent that either boundary selection lies within the limits established by the tungsten marker experiment. The innermost remaining wire lies in a UO_2 structure zone that was below the melting point defined by either boundary. The pre-irradiation location of the radially outermost wire that disappeared lies in a zone that was molten as defined by either boundary. However, the particular microstructure zone selected by the Hanford and Chalk River groups as molten has not normally been observed in the post-irradiation UO_2 metallography of the pellet fuel rods from this Program. The UO_2 within the pore termination boundary in these rods has been essentially pore-free except in the area immediately around the central void. (See the microstructures in Figures 21A, 21B, 22A, and 22B.) Some randomly oriented sub-grain structure and metallic UO_2 inclusions are present, but the well-defined zone of oriented porosity and sub-grain structure was not observed in these fuel rods with burnups, less than 2500 MWD/ T_e (0.68×10^{20} fissions/ cm^3). However, this type structure was apparent in the cross sections of low-density, compacted powder rods irradiated under this same Program. This structure has also been observed recently in the cross sections taken from rod 12D that attained 5300 MWD/ T_e (1.4×10^{20} fissions/ cm^3) (Figures 24A and 24B). As a result, the appearance of this particular structure is believed to be related to the type of UO_2 its initial density, or perhaps the fuel rod gas content, rather than to be a specific indicator of melting. Considering the evidence offered previously and the discussion of the results obtained from other investigations, it is concluded that the radially inward termination of porosity is the actual molten boundary. This boundary lies just slightly inward radially from the transition from fine, needle-like columnar grains to very large columnar grains.

5. Limitations of Post-Irradiation Metallography in Indicating UO_2 Conditions During Operation

As indicated earlier, there is distinct evidence of molten UO_2 in the maximum burnup cross section taken from rod 6D, Figure 20B. On the other hand, there is no metallographic evidence of melting at all in the peak heat flux cross section taken from this same rod, (Figure 20A). Based on calorimetry and flux profiles, the cross section at the peak heat flux zone at startup operated at least $100,000 \text{ Btu/h-ft}^2$ (32 W/cm^2) higher than that at the maximum burnup zone. (The actual difference could have been nearly twice as large because of clad swelling at the peak heat flux zone. See column 12 Table 14.) That the section at the peak heat flux zone did operate considerably higher than that at the maximum burnup zone can be verified by comparing the diameter of the grain growth zones in the two sections. The outer limit of equiaxed grain growth occurs at a smaller fractional radius in the maximum burnup section, i. e., the width of the as-sintered structure is larger. As a first approximation, the radial location for the onset of equiaxed grain growth should correspond to the same UO_2 operating temperature in each section. Consequently, the occurrence of the same temperature further inward along the radius indicates a lower thermal performance level in the maximum burnup section and confirms the results calculated from calorimetry and flux profiles. Thus, if molten UO_2 is present in the maximum burnup cross section, a considerably larger zone must have been present in the peak heat flux cross section at the start of irradiation. However, as the irradiation progressed the surface heat flux continuously decreased at this cross section until the heat flux was lower at the end of the irradiation than it was at the maximum burnup cross section. The absence of any trace of molten UO_2 in the peak heat flux cross section is unequivocal proof that continued irradiation can completely modify the microstructure of UO_2 and, particularly, eliminate evidence of previous, higher thermal performance operation. It follows that the UO_2 microstructure characteristics observed in post-irradiation metallography are only truly representative of the UO_2 thermal performance level just before shutdown rather than any earlier period. For this reason, the analyses to determine UO_2 thermal conductivity described later is based primarily on end-of-life operating conditions and the corresponding UO_2 microstructures.

The same effect of the decreasing level of thermal performance on the UO_2 microstructure is apparent in the cross sections taken from rods 8A and 10D. Here, however, the operating thermal performance level was so high that molten UO_2 is evident in both the peak heat flux zone at startup and the maximum burnup zone. The locations of the significant changes in microstructure are so far out radially that the differences in position of these locations caused by variations in thermal performance conditions between the two sections are very small. Nevertheless, careful examination of the peak heat flux zone cross sections taken from rods 8A and 10D (Figures 21A and 22A) indicates that the fractional diameters of the grain growth zones are larger than those at the maximum burnup location (Figures 21B and 22B). This result verifies the higher thermal performance level at the peak heat flux zone. The fractional diameter of the molten zone is larger in the maximum burnup cross sections than at the peak

heat flux cross sections in both cases. The larger radial distance between the grain growth boundary and the molten boundary in the peak heat flux cross sections is occupied by a wider, annular band of fine columnar grains. A part of this band is believed to have been formed by UO_2 resolidification as the cross section decreased in heat flux.

The precise interpretation of the startup, peak heat flux zone microstructures in rods 8A and 10D is complicated by the severe clad swelling that occurred at these sections. The diameter of the end-of-life molten zone may correspond to a higher peak heat flux than that calculated on the basis of the initial, uniform rod diameter. This is suggested by the equivalent location cross section from rod 12E shown in Figure 23A, which did not experience clad swelling. At startup, the peak heat flux at this cross section was about the same as that in rod 10D. However, the molten zone at end-of-life in cross section of rod 12E is much smaller than that in either rod 8A or rod 10D. The end-of-life peak heat flux at the cross section of rod 12E is also calculated to be less than either rod 8A or rod 10D, although only slightly less than that for rod 8A. Thus, the discrepancy in the size of the molten heat zones may be attributable to swelling and its effect on surface heat flux. The correct surface heat flux value for analyzing the swollen cross sections is uncertain as previously indicated.

E. AUTORADIOGRAPHIC EXAMINATION

Autoradiography of irradiated UO_2 is a relatively new technique and was developed extensively during this Program.⁽²⁹⁾ Some consideration must be made for chronological improvement in autoradiograph preparation technique when interpreting the autoradiographs in this report. The objective in employing autoradiography was to find some reliable correspondence between boundaries observed in the autoradiographs and those apparent in polished metallographic specimens. In addition, it was expected that the autoradiographs would provide qualitative data on the distribution of fission products in fuel rods operated at central melting conditions. Both of these expectations have been realized and at least a few tentative conclusions can be made on both autoradiograph boundaries and fission product distribution. However, considerably more work with different lots of UO_2 and of varied initial density, is required to make a completely reliable interpretation of an autoradiograph. The autoradiographs in this report are positive prints made from the original glass plates. Consequently, the light areas correspond to activity concentrations and the dark areas to activity depletion.

Figure 20A shows an autoradiograph of the peak heat flux cross section from rod 6D. There is an annular zone of depleted activity with an outer diameter equal to approximately one-half the cross section diameter and an inner diameter about one-quarter of the cross section diameter. Within this zone is another zone of slightly increased activity around the central void. The central void itself is quite dark. The depleted activity zone is considered to be confirmation of a radial redistribution of fission products. Based on previous discussions on this cross section, a logical conclusion would be that the outer diameter of the depleted activity zone corresponded to the outer

limit of melting at startup. Based on the results of analyses the location of this boundary does correspond to the expected molten boundary but the diffuseness of the autoradiograph boundary and the absence of any remaining evidence of the initial molten boundary in the metallography for this particular cross section leave such a correlation quite tenuous.

Figure 20B shows the autoradiograph of the maximum burnup zone in rod 6D. It should be remembered that this cross section was taken through the zone that exhibited the large activity spike in the axial gamma scan. The concentration of activity at this cross section is readily apparent. This apparent activity concentration is not simply the result of shine through cracks in the UO_2 . Small holes have been drilled into the UO_2 structure at various points to simulate the effect of the additional surface and depth of a crack.⁽³⁰⁾ In some cases, these holes appear darker than the surroundings in subsequent autoradiography, rather than lighter. The activity,, however, may be plate out on the surfaces of the cracks, but this would still imply axial movement and concentration of fission products in the hottest zone.

The maximum burnup zone autoradiograph exhibits a very thin, annular band of depleted activity at a radial location corresponding almost exactly to the radius for the onset of grain growth. A similar band at the same location is evident in the maximum burnup cross section and other cross sections above the maximum burnup zone in rod 8A. However, the band was not visible in subsequent cross sections from rod 10D or rod 12E. Within this depleted activity band is another annular fuel zone with an activity level about equal to that of the outermost UO_2 . This annular zone corresponds to the regions of equiaxed and fine columnar grains in the metallographic cross sections. This zone terminates at another depleted activity zone which then extends inward to the very high activity zone at the central void. Again the possible correspondence of this inner depleted activity zone with maximum diameter of melting is obvious and is supported by the results of analysis (Section XI).

The autoradiograph from the peak heat flux zone of rod 8A, Figure 21A, exhibits an entirely new phenomenon that is considered most significant. In this cross section, a very narrow, bright ring of activity appears at approximately the outer limit of the central depleted activity zone. The ring is discontinuous and appears to be comprised of many small discrete particles of high activity. Particles that appear to correspond to this activity ring have been observed in the microstructure but their composition has not yet been determined. This activity ring appears in all subsequent cross sections operated at higher levels of thermal performance where there was some decrease in the level of thermal performance after the maximum was attained. This correspondence is indicative of activity being deposited from the molten UO_2 where resolidification occurs. The activity ring is believed to indicate the outer limit of melting in those cross sections where it appears and was used as the molten boundary (Section XI).

Conversely, the activity ring is not present in the cross sections that reached their maximum level of thermal performance just before shutdown. The maximum burnup zone, autoradiograph from rod 8A, Figure 21B, is an illustration of this type cross section. However, the resolution of the autoradiography in Figure 21B is so sharp that it provides the illusion of an optical photograph. The demarcation between molten and nonmolten material is almost visually evident. The diameter of the central depleted activity zone in this autoradiograph appears to be the molten boundary and was used as such for cross sections without the bright activity ring. The use of these two molten boundary indicators from autoradiography provides results on UO_2 thermal conductivity which are consistent with those obtained from UO_2 microstructure observations (Section XI).

X. SUPPORTING CAPSULE EXPERIMENTS

Two capsule irradiations were performed in an effort to improve the understanding of the clad swelling phenomenon in the PWL fuel rods. The UO_2 volume expansion on melting was suspected to be responsible for the clad swelling in the PWL fuel rods but the same phenomenon had not occurred in capsules irradiated before the start of the PWL irradiations.⁽¹⁰⁾ It was finally concluded that the discrepancy in results arose from the ability of the capsules to accommodate the UO_2 volume expansion by axial fuel movement. Such axial fuel movement could not occur in the long, unsegmented PWL rods. The new capsule irradiations were undertaken to verify this explanation for the absence of clad swelling in the previous capsules and, thus, to provide further confirmation indirectly of the suspected mechanism for clad swelling. In addition, it was hoped to obtain an initial evaluation of the ability of hollow fuel pellets to prevent clad swelling.

The basic configuration of the capsules is shown in Figure 34 and is substantially the same as those previously irradiated. The cladding is 0.565-inch (1.44 cm) o.d., 0.030-inch (0.076 cm) wall Zircaloy-2. The fuel column consists of ten, 0.5-inch (1.27 cm) o.d. by 0.5-inch (1.27 cm) high 95% dense sintered UO_2 pellets. The pellets were centerless ground on the outside diameter to provide an initial 6-mil (0.015 cm) diametral gap between the pellet and the cladding. With the exception of fuel column length, the capsule geometry is identical to the PWL fuel rods. The fabrication procedures were also identical.

The single, significant change in the new capsules from those previously irradiated is in the design of the end plug. The earlier end plugs were identical to the new end plugs in external dimensions, but contained a 3/8-inch (1 cm) o.d., 1/4-inch (0.6 cm) deep blind hole in the face adjacent to the fuel column. It was this hole that provided room for axial expansion of the molten UO_2 . In addition, some axial expansion space had been allowed between the fuel column and end plugs. The new capsules were fabricated with solid Zircaloy-2 end plugs in tight contact with the fuel column.

The capsules were irradiated in the GETR Trail Cable Facility by using direct forced convection cooling with $\sim 85^\circ\text{F}$ (30°C) inlet temperature water. The vertical location of the capsule in the facility was adjusted for the maximum obtainable power output and then the capsule was irradiated for 20 minutes at this steady-state condition.

Capsule 8 was the first capsule irradiated and contained a column of solid UO_2 pellets. The capsule operated at an average heat flux of $920,000 \text{ Btu/h-ft}^2$ (290 W/cm^2) which was sufficient to generate appreciable central melting. Dimensional measurements after irradiation showed a minimum diameter increase of 10 mils (0.025 cm) over the entire active fuel length with a peak swelling of 14 mils (0.036 cm) near the upper end. The dimensional data are listed in Table 13. The dimensional profile of the capsule is shown in Figure 35.

TABLE 13. Capsule No. 8 Pre- and Post-Irradiation Dimensions

Capsule Orientation, degrees	Axial Distance 1/2 inch Increments	Outside Measurements						Diameter Change		
		8/6/62			9/21/62			Difference in Sums	One-Half Difference	Standard Correction =+0.0009 inch
		Read I	Read II	Sum	Read I	Read II	Sum			
0-180	1	0.0645	0.0646	0.1291	0.0674	0.0656	0.1330	0.0039	0.0020	0.0029
	2	0.0665	0.0661	0.1326	0.0682	0.0681	0.1363	0.0037	0.0019	0.0028
	3	0.0655	0.0656	0.1311	0.0784	0.0790	0.1574	0.0263	0.0132	0.0141
	4	0.0655	0.0658	0.1313	0.0743	0.0743	0.1486	0.0173	0.0087	0.0096
	5	0.0655	0.0656	0.1311	0.0727	0.0735	0.1462	0.0151	0.0076	0.0085
	6	0.0657	0.0655	0.1312	0.0740	0.0749	0.1489	0.0177	0.0089	0.0098
	7	0.0659	0.0659	0.1318	0.0729	0.0739	0.1468	0.0150	0.0075	0.0084
	8	0.0659	0.0659	0.1318	0.0742	0.0748	0.1490	0.0172	0.0086	0.0095
	9	0.0660	0.0659	0.1319	0.0751	0.0754	0.1505	0.0186	0.0093	0.0102
	10	0.0661	0.0660	0.1321	0.0751	0.0755	0.1506	0.0185	0.0093	0.0102
	11	0.0664	0.0664	0.1328	0.0753	0.0751	0.1504	0.0176	0.0088	0.0097
	12	0.0675	0.0673	0.1348	0.0661	0.0663	0.1324	-0.0024	-0.0012	-0.0003
90-270	1	0.0660	0.0660	0.1320	0.0649	0.0642	0.1291	-0.0029	-0.0015	-0.0006
	2	0.0653	0.0652	0.1305	0.0694	0.0687	0.1381	0.0076	0.0038	0.0047
	3	0.0653	0.0654	0.1307	0.0779	0.0773	0.1552	0.0245	0.0123	0.0132
	4	0.0652	0.0652	0.1304	0.0753	0.0745	0.1498	0.0194	0.0097	0.0106
	5	0.0654	0.0653	0.1307	0.0739	0.0733	0.1472	0.0165	0.0083	0.0092
	6	0.0655	0.0655	0.1310	0.0744	0.0737	0.1481	0.0171	0.0086	0.0095
	7	0.0654	0.0653	0.1307	0.0744	0.0733	0.1477	0.0170	0.0085	0.0094
	8	0.0655	0.0652	0.1307	0.0746	0.0743	0.1489	0.0182	0.0091	0.0100
	9	0.0654	0.0654	0.1308	0.0766	0.0756	0.1522	0.0214	0.0107	0.0116
	10	0.0656	0.0654	0.1310	0.0757	0.0750	0.1507	0.0197	0.0099	0.0108
	11	0.0657	0.0656	0.1313	0.0756	0.0753	0.1509	0.0196	0.0098	0.0107
	12	0.0676	0.0671	0.1347	0.0670	0.0668	0.1338	-0.0009	-0.0005	0.0004

Note: Measurements were taken with a dial gage instrument and are relative to a standard capsule. Dial gage measurements on capsule standard ($\cong 0.565$ inch o.d.) averaged 0.0009 inch larger on 8/6/62 than on 9/21/62 based on 48 measurements on each date. Thus, 0.0009 inch must be added to data from 9/21/62 for comparison and determination of the true dimensional change.

Capsule 7 was the second capsule irradiated, and it contained hollow UO_2 fuel pellets with an 0.085-inch (0.216 cm) central hole, similar to the EPT-COM fuel rods. The initial irradiation of this capsule was unsatisfactory because the facility instrumentation failed. There was no detectable clad swelling after irradiation, but the capsule probably operated at a heat flux less than sufficient to generate central melting. A second irradiation was planned, but became unnecessary when the results from the initial EPT-COM assembly irradiation were obtained.

The significant swelling generated in capsule 8 provided considerable confidence that the postulated explanation for the lack of swelling in the previous capsules was correct. At the same time, the favorable results obtained from this irradiation strengthened the conviction that the UO_2 volume expansion from melting is the basic clad swelling force.

Figure 34. Configuration - Capsule No. 8

Figure 35. Dimensional Changes - Capsule No. 8

XI. ANALYSIS

A. UO₂ THERMAL CONDUCTIVITY

Intensive analysis effort has been directed toward determining the UO₂ thermal conductivity indicated by these fuel rod irradiations. The major portion of this effort was that associated with establishing the precise values for the maximum surface heat flux during life and the end-of-life heat flux applicable to each fuel cross section examined metallographically. These calculations encompass the determination of the:

1. Chronological variation in the fuel assembly net power generation,
2. Determination and application of the relative power distribution among the four assembly fuel rods, and
3. Application of axial peaking factors appropriate to the section location and the reactor control rod bank position at the time of interest.

The detailed calculation methods are described in Appendices A and B. The results of the first part of these calculations are summarized in Figures 9 through 13, which indicate the chronological variation in the average heat flux for the five fuel rods examined metallographically. The calculations used for determining the maximum heat flux and end-of-life heat flux values for the individual metallographic sections are shown in Tables 14 and 15. The axial power distribution factors used in these calculations were determined from Figures 36, 37, and 38, and are based on the PWL flux measurements described in Appendix B. Figures 39 through 43 indicate the heat flux values at any point on the rods: both the maximum value during irradiation and the value existing at end-of-life.

The determination of the UO₂ thermal conductivity is based on the observed radius for melting in the metallographic sections and the use of the parameter ⁽³¹⁾

$$\int_0^{T_m} k dT$$

The value of the conductivity integral from the pellet surface to the observed molten boundary.

TABLE 14. Metallographic Section Maximum Heat Flux Values

1	2	3	4	5	6	7	8	9	10	11	12
Rod	Metallographic Section	Description	Post Irradiation Diameter, in. Fuel Clad	Axial Distance from Bottom of Rod, in.	Rod Average Heat Flux Btu/h-ft ² (W cm ²)	Date	Control Rod Bank Position, in.	Axial Peaking Factor at Section	Maximum Heat Flux at Section During Life, Btu/h-ft ² (W cm ²)	Fractional Diameter Increase at Section	Estimated Heat Flux after Swelling, Btu/h-ft ² (W cm ²)
6D	8560	MBU*, PHF** End of Life	- 0.564	18.3	483,000 (152)	5/14/62	28.1	1.32	638,000 (201)	-	-
	8561	PHF Startup, MS***0.544	- 0.606	8.0	453,000 (143)	4/24/62	19.3	1.57	711,000 (224)	1.072	763,000 (240)
	8576	Longitudinal, MS	- 0.574	9.3-10.3	453,000 (143)	4/24/62	19.3	1.62	734,000 (231)	1.019	746,000 (235)
	8577	Longitudinal, MS	- 0.589	8.2-9.3	453,000 (143)	4/24/62	19.3	1.59	720,000 (226)	1.043	751,000 (237)
8A	8596	PHF Startup, MS	0.535 0.598	9.6	583,000 (184)	6/1/62	18.0	1.62	944,500 (298)	1.061	1,001,000 (315)
	8597	MBU PHF End of Life	- 0.564	17.9	590,000 (186)	6/20/62	26.8	1.33	784,700 (247)	-	-
	8598	Longitudinal, MS	- 0.606	8.4-9.4	583,000 (184)	6/1/62	18.0	1.62	944,500 (298)	1.074	1,014,000 (319)
	8599	Longitudinal, Bottom of Rod	- 0.564	1.0-2.3	583,000 (184)	6/1/62	18.0	0.92	536,400 (169)	-	-
	8600	Longitudinal Top of Central Void	- 0.563	26.8-28.0	571,000 (180)	6/24/62	27.5	0.84	479,600 (151)	-	-
10D	8661	PHF Startup, MS	0.585 0.645	7.0	716,300 (226)	7/4/62	18.5	1.44	1,031,500 (325)	1.138	1,173,000 (369)
	8662	MBU, PHF End of Life	0.516 0.572	13.2	810,300 (255)	7/14/62	22.8	1.39	1,126,300 (355)	1.009	1,136,000 (358)
	8663	Longitudinal	- 0.632	5.7-6.75	716,300 (226)	7/4/62	18.5	1.36	974,200 (307)	1.115	1,084,000 (341)
	8664	Longitudinal	- 0.567	19.17-20.27	782,000 (246)	7/19/62	24.4	1.13	883,700 (278)	-	-
12E	8713	PHF Startup, MS	- 0.567	5.6	780,400 (246)	9/15/62	17.7	1.31	1,022,300 (322)	-	-
	8714	MBU, PHF End of Life	- 0.567	12.2	780,400 (246)	9/15/62	17.7	1.42	1,108,200 (349)	-	-
	8748	Longitudinal	- 0.567	28.6-29.25	776,000 (244)	12/16/62	28.5	0.72	558,700 (176)	-	-
12D	8811	PHF Startup 1/4 in below Failure	- 0.567	9.4	865,000 (272)	2/3/62	19.2	1.44	1,245,000 (392)	-	1,255,000 (395)
	8810	MBU	- 0.562	18.5	818,000 (258)	3/14/62	25.8	1.18	965,000 (304)	-	965,000 (304)

* MBU - Maximum Burnup Region
 ** PHF - Peak Heat Flux Region
 *** MS - Maximum Swelling Region

0.565 in = 1.435 cm

TABLE 15. Metallographic Section End of Life Heat Flux Values

<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>		<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>	<u>9</u>	<u>10</u>		<u>11</u>	<u>12</u>	
Rod	Metallo- graphic Section	Description	Post Irradiation Diameter, in Fuel	Clad	Axial Distance from Bottom of Rod, in.	Rod Average Heat Flux, Btu/h-ft ² (W/cm ²)	Date	Control Rod Bank Position, in.	Axial Peaking Factor at Section	End of Life Heat Flux at Section, Btu/h-ft ² (W/cm ²)		Fractional Diameter Increase at Section	Estimated Heat Flux after Swelling, Btu/h-ft ² (W/cm ²)	
6D	8560	MBU*, PHF ** End of Life	-	0.564	18.3	462,000 (146)	5/19/62	31.5	1.29	596,000 (188)		-	-	
	8561	PHF Startup;MS***	0.544	0.606	8.0	462,000 (146)	5/19/62	31.5	0.88	406,000 (128)		1.072	435,000 (137)	
	8576	Longitu- dinal; MS	-	0.574	9.3-10.3	462,000 (146)	5/19/62	31.5	0.97	448,000 (141)		1.019	456,000 (144)	
	8577	Longitu- dinal; MS	-	0.589	8.2-9.3	462,000 (146)	5/19/62	31.5	0.92	425,000 (134)		1.043	444,000 (140)	
8A	8596	PHF Startup;MS	0.535	0.598	9.6	571,000 (180)	6/24/62	27.5	1.22	696,600 (219)		1.061	738,000 (232)	
	8597	MBU PHF End of Life	-	0.564	17.9	571,000 (180)	6/24/62	27.5	1.32	753,700 (237)		-	-	
	8598	Longitu- dinal; MS	-	0.606	8.4-9.4	571,000 (180)	6/24/62	27.5	1.19	679,500 (214)		1.074	730,000 (230)	
	8599	Longitudinal Bottom of Rod	-	0.564	1.0-2.3	571,000 (180)	6/24/62	27.5	0.64	365,400 (115)		-	-	
	8600	Longitudinal, Top of Central Void	-	0.563	26.8-28.0	571,000 (180)	6/24/62	27.5	0.84	479,600 (151)		-	-	
10D	8661	PHF Startup; MS	0.585	0.645	7.0	782,000 (246)	7/19/62	24.4	1.19	930,600 (293)		1.138	1,058,000 (333)	
	8662	MBU PHF End of Life	0.516	0.572	13.2	782,000 (246)	7/19/62	24.4	1.36	1,063,500 (335)		1.009	1,072,000 (338)	
	8663	Longitudinal	-	0.632	5.7-6.75	782,000 (246)	7/19/62	24.4	1.12	875,800 (276)		1.115	976,000 (307)	
	8664	Longitudinal	-	0.567	19.17-20.27	782,000 (246)	7/19/62	24.4	1.13	883,700 (278)		-	-	
12E	8713	PHF Startup; MS	-	0.567	5.6	776,000 (244)	12/16/62	28.5	0.85	659,600 (208)		-	-	
	8714	MBU PHF End of Life	-	0.567	12.2	776,000 (244)	12/16/62	28.5	1.18	915,700 (288)		-	-	
	8748	Longitudinal; Top of Rod	-	0.567	28.6-29.25	776,000 (244)	12/16/62	28.5	0.72	558,700 (176)		-	-	
12D	8811	PHF Startup; 1/4 in Below Failure	-	0.567	9.4	679,000 (214)	6/17/62	18.6	1.45	985,000 (310)		-	993,000 (313)	
	8810	MBU	-	0.562	18.5	679,000 (214)	6/17/62	18.6	1.01	685,000 (216)		-	685,000 (216)	

* MBU - Maximum Burnup Region
 ** PHF - Peak Heat Flux Region
 *** MS - Maximum Swelling Region

($T = 2800^{\circ}\text{C}$) can be determined from the relationship:

$$\int_{T_s}^{T_m} k dT = Q/A D M_i$$

where

- k = Thermal conductivity;
- T = Temperature (pellet);
- T_s = Pellet Surface temperature;
- T_m = UO_2 melting point (2800°C);
- Q/A = Surface heat flux at D ;
- D = Clad or pellet outside diameter;
- M_i = Factor dependent on the pellet diameter, enrichment, and the observed radius for melting.

The values for M_i were calculated by using the flux depression versus enrichment curve shown in Figure 44 and by assuming a parabolic power distribution within the pellets.

To obtain the remaining portion of the integral, i. e. ,

$$\int_{0^{\circ}}^{T_s} k dT$$

the UO_2 thermal conductivity as a function of temperature and the pellet surface temperature must be known and the integral determined directly.

The UO_2 thermal conductivity relation used in the calculations is that derived from earlier short-time capsule irradiations. ⁽¹⁴⁾ In the temperature range of interest, i. e. , from 0°C to T_s , the conductivity curve is simply a least squares fit to all available out-of-pile conductivity data.

The main uncertainty in the

$$\int_0^{T_s} k dT$$

calculation is the value for the pellet surface temperature T_s , as a result of the uncertainty in

Figure 36. Axial Power Distribution - 34-1/4-inch Fuel Column

Figure 37. Axial Power Distribution - 31-1/4-inch Fuel Column

Figure 38. Axial Power Distribution - 29-1/2-inch Fuel Column

Figure 39. Axial Heat Flux Distribution - Rod 6D

Figure 40. Axial Heat Flux Distribution - Rod 8A

Figure 41. Axial Heat Flux Distribution - Rod 10D

Figure 42. Axial Heat Flux Distribution - Rod 12E

Figure 43. Axial Heat Flux Distribution - Rod 12D

Figure 44 UO_2 Flux Depression as a Function of Enrichment

the pellet-to-clad interface conductivity. No analyses have yet been made to evaluate the pellet-to-clad interface conductivity in these fuel rods. These analyses will be performed later; however, for this report a constant value of $2000 \text{ Btu/h-ft}^2\text{-}^\circ\text{F}$ ($\text{W/cm}^2\text{-}^\circ\text{C}$) was assumed for the pellet-to-clad interface conductivity. This value is somewhat lower than that observed ⁽¹⁰⁾ in earlier short-time capsule irradiations [$h_{if} = 2000 \text{ to } 4000 \text{ Btu/h-ft}^2\text{-}^\circ\text{F}$ ($\text{W/cm}^2\text{-}^\circ\text{C}$)] and was chosen to reflect an expected decrease in the interface conductivity as a result of fission gas contamination of the original helium atmosphere in the fuel rods. However, it should also be noted that the interface conductivity may well be much higher than the value assumed, particularly for those sections that exhibited significant clad swelling. The swelling implies a contact pressure occurred between the clad and UO_2 which was considerably higher than the clad yield strength. Very high interface conductivity values have been observed at high contact pressures in out-of-pile experiments, ⁽³²⁾ i.e., values exceeding $4000 \text{ Btu/h-ft}^2\text{-}^\circ\text{F}$ ($\text{W/cm}^2\text{-}^\circ\text{C}$). Further, the earlier capsule irradiation results indicated a precipitous rise [above $3000 \text{ Btu/h-ft}^2\text{-}^\circ\text{F}$ ($\text{W/cm}^2\text{-}^\circ\text{C}$)] in the interface conductivity, accompanying the onset of central melting. Thus, the value of $2,000 \text{ Btu/h-ft}^2\text{-}^\circ\text{F}$ ($\text{W/cm}^2\text{-}^\circ\text{C}$) is believed to be conservative and probably overpredicts both the pellet surface temperature and the value of the integral

$$\int_0^{T_s} k_s dT$$

In turn, the total integral value

$$\int_0^{T_m} k dT = \int_0^{T_s} k dT \int_{T_s}^{T_m} k dT$$

should also be conservative and represent the maximum value that can be inferred from the cross sections.

The calculations to determine the

$$\int_0^{T_m} k dT$$

values are summarized in Tables 16 and 17. In Table 16, the porosity termination boundary was used as the molten boundary indicator and the end-of-life heat flux as the thermal performance value that determined the location of the molten boundary. The location of the porosity termination boundaries was found from the detailed metallographic composite photographs. In Table 17, the bright ring from the autoradiographs was used as the indicator of the maximum diameter of melting in each section with the maximum surface heat flux experienced at the cross section. In those instances where the bright ring does not appear, the outer radius of the central zone of depleted activity was taken as the boundary as discussed in Section IX-E. The exact location of the molten boundary is indicated on the photographs taken of the various metallographic cross sections.

B. UO₂ VOLUME EXPANSION ON MELTING

It should be possible to obtain an estimate of the UO₂ volume expansion during melting from the slope of the curve shown in Figure 19. However, the determination is complicated by:

1. The gross fuel rod volume changes shown in Figure 19 include both the isothermal expansion associated with the UO₂ experiencing the phase change and the thermal expansion of all of the UO₂ as the average temperature level of the fuel rod increases. To calculate the phase change volume expansion, data for the volumetric thermal expansion must be assumed. For the analysis described here, the thermal expansion data of Christiansen⁽⁸⁾ was used.
2. The UO₂ pellets are only 95% dense. Heated above its original sintering temperature, UO₂ continues to densify and at some temperature below the melting point, the density will reach 100%. Thus, an unknown fraction of the UO₂ between the sintering temperature and melting point will densify and provide additional free volume to compensate partially for the phase change expansion. For analysis, the UO₂ volume at temperatures between

TABLE 16. $\int_0^{T_m} k dT$ Values Based on Metallography (End of Life Heat Flux Values)

Rod	Metallographic Section	End of Life Heat Flux		Normalized	Enrichment (%)	Flux Depression Factor	Based on Initial Rod Diameter				After Swelling			
		Btu/h-ft ²	(W/cm ²)	Radius For Melting (α M)			T _s (h _{gap} = 2000) °F	°C	∫ _{T_s} ^{T_{m,kdT}}	∫ ₀ ^{T_{m,kdT}}	T _s (h _{gap} = 2000) °F	°C	∫ _{T_s} ^{T_{m,kdT}}	∫ ₀ ^{T_{m,kdT}}
6D	8560	596,000	(188)	0.133	1.5	0.241	964 °F	(518)	65.0	95.4	-	-	95.4	
	8561	406,000	(128)	-	1.5	-			w/cm	w/cm				
	8576	448,000	(141)	-	1.5	-								
	8577	425,000	(134)	-	1.5	-								
8A	8596	696,600	(219)	0.523	2.2	0.179	1046 °F	(563)	56.4	88.3	1087	(587)	59.9	92.7
	8597	753,700	(237)	0.573	2.2	0.165	1102	(594)	56.4	89.8	-	-	89.8	
	8598	679,500	(214)	0.430	2.2	0.198	1028	(553)	60.8	92.3	1078	(581)	65.4	98.2
	8599	365,400	(115)	-	2.2	-	-	-	-	-	-	-	-	-
	8600	479,600	(151)	0.195	2.2	0.233	831	(444)	50.6	78.1 *	-	-	-	78.1 *
10D	8661	930,600	(293)	0.648	3.0	0.140	1268 °F	(687)	59.0	95.5	1394	(757)	67.0	106.8
	8662	1,063,500	(335)	0.673	3.0	0.132	1398	(759)	63.5	102.4	1408	(764)	64.0	103.8
	8663	875,800	(276)	0.588	3.0	0.159	1214	(657)	63.0	98.1	1312	(711)	70.2	107.6
	8664	883,700	(278)	0.641	3.0	0.142	1222	(661)	56.6	92.2	-	-	-	92.2
12E	8713	659,600	(208)	0.339	3.8	0.210	996	(535)	62.6	93.8	-	-	-	93.8
	8714	915,700	(288)	0.637	3.8	0.143	1244	(673)	59.4	95.4 *	-	-	-	95.4
	8748	558,700	(176)	0.451	3.8	0.191	898	(481)	48.3	77.4 *	-	-	-	77.4 *
										94.3 ± 2.9 w/cm				97.6 ± 4.5
12D	8811	985,000	(310)		3.8									
	8810	685,000	(216)		3.8									
* Not included in Summation														

* Not included in Summation

TABLE 17. $\int_0^T T_m k dT$ Values Based on Autoradiography

Rod	Metallographic Section	Maximum Heat Flux Btu/h-ft ² (W/cm ²)	Enrichment (%)	Normalized Radius For Melting (αM)	Flux Depression Factor	Based on Initial Rod Diameter				After Swelling			
						T_s ($h_{gap} - 2000$)		$\int_{T_s} T_m k dT$	$\int_0^T T_m k dT$	T_s ($h_{gap} = 2000$)		$\int_{T_s} T_m k dT$	$\int_0^T T_m k dT$
						°F	°C	(W/cm)	(W/cm)	°F	°C	(W/cm)	(W/cm)
6D	8560	638,000 (201)	1.5	0.410	0.205	1007	(542)	59.4	90.7	-	-	-	91.1
	8561	711,000 (224)	1.5	0.496	0.185	1056	(569)	59.5	91.7	1108	(598)	64.0	97.3
	8576	734,000 (231)	1.5	*	-	1075	(579)	-	-	-	-	-	-
	8577	720,000 (226)	1.5	*	-	1070	(577)	-	-	-	-	-	-
8A	8596	944,500 (298)	2.2	0.620	0.151	1291	(700)	64.5	101.5	1348	(731)	68.4	106.4
	8597	784,700 (247)	2.2	0.642	0.144	1133	(612)	51.1	84.8	-	-	-	84.8
	8598	944,500 (298)	2.2	0.643	0.144	1291	(700)	61.5	98.5	1361	(739)	66.0	104.2
	8599	536,400 (169)	2.2	-	-	887	(475)	-	-	-	-	-	-
	8600	479,600 (151)	2.2	0.293	0.224	831	(444)	48.5	76.0**	-	-	-	76.0**
10D	8661	1,031,500 (325)	3.0	0.720	0.116	1367	(742)	54.1	92.4	1506	(819)	61.5	102.5
	8662	1,126,300 (355)	3.0	0.715	0.118	1460	(793)	60.1	100.3	1444	(785)	60.7	100.7
	8663	974,200 (307)	3.0	*	-	1311	(710)	-	-	-	-	-	-
	8664	877,500 (278)	3.0	0.666	0.135	1216	(658)	53.7	88.9	-	-	-	88.9
12E	8713	1,022,300 (322)	3.8	0.624	0.147	1348	(731)	68.2	106.3	-	-	-	106.3
	8714	1,108,200 (349)	3.8	0.688	0.127	1431	(777)	63.5	103.1	-	-	-	103.1
	8748	558,700 (176)	3.8	0.515	0.177	898	(481)	44.7	73.5**	-	-	-	73.5**
									95.6 ± 5.1				

* No Autoradiograph

** Not included in Summation

1600°C, i. e., the original sintering temperature, and the melting point, was assumed to provide 2.5% additional free volume. Above the melting point, UO_2 was assumed to provide 5% additional free volume.

3. In the lower average thermal performance fuel rods, the extreme ends of the fuel column operate at very low power and the fuel pellets probably do not expand to fill the radial gap. As a result, the available free volume in this region is of little consequence to the clad swelling phenomenon. Only that portion of the fuel column bounded by pellets locked tightly in the cladding is effective in the volume expansion which produces clad swelling. That portion of the fuel column with an average fuel temperature of less than 1300°C was arbitrarily neglected in the volume expansion analysis in an effort to account for this effect.
4. The thermal conductivity of UO_2 above the melting point is unknown. Consequently, the temperature distribution in the liquid zone of the fuel rods is unknown. If the temperature rise were appreciable, then the thermal expansion of molten UO_2 should also be considered in the analysis. However, based on evidence for convection in the molten UO_2 already cited, the average temperature in the molten zone probably is not much greater than the melting temperature. On this basis, the volume expansion of molten UO_2 was neglected.
5. The experimentally observed volume expansions of the fuel rods represent the volume after the rods have cooled thermally and after relief of the expansion generating forces. Presumably, the volume increases were somewhat larger at the actual fuel rod operating conditions, before the thermal and elastic contractions occurred. Analyses were made to evaluate the corrections to the volume expansion data for each of the effects. The results indicate that the corrections are essentially constant for all of the rods and, thus, do not affect the slope of the volume expansion versus thermal performance curve shown in Figure 19. Therefore, the corrections can be neglected in the analysis that follows to determine the UO_2 phase change volume expansion.

The first step in the volume expansion analysis was to determine the volume of fuel between fixed temperature limits over the total length of the fuel columns. For this purpose, the fuel was divided in ten temperature zones and analyzed to determine the fuel volume in each zone for each 1-inch axial segment of the fuel column. The volumes in each zone were then added

TABLE 18. UO_2 Volume Expansion Analysis Results

Rod	6A	6B	6C	6D	8A	8B	8C	8D	10A	10B	10C	10D
Peak Heat Flux at Startup, Btu/h-ft ²	630,000	665,000	675,000	732,000	950,000	920,000	780,000	915,000	1,030,000	1,040,000	1,140,000	1,140,000
Calculated Volume, in. ³												
Molten Zone	0.235	0.364	0.389	0.596	1.439	1.375	0.850	1.342	1.501	1.518	1.816	1.821
Zone Between 1600 and 2800°C	1.125	1.191	1.199	1.270	1.349	1.349	1.329	1.347	1.352	1.356	1.334	1.334
Free Volume Densification, in. ³												
From Molten Zone	0.012	0.018	0.019	0.030	0.072	0.069	0.043	0.067	0.075	0.076	0.091	0.091
From 1600 to 2800°C	0.028	0.030	0.030	0.032	0.034	0.034	0.033	0.034	0.034	0.034	0.033	0.033
Pellet-Clad Gap Volume, in. ³	0.103	0.113	0.118	0.0997	0.1478	0.1207	0.1289	0.1415	0.1186	0.1270	0.1361	0.1361
Total Available Free Volume, in. ³	0.143	0.161	0.167	0.162	0.254	0.224	0.205	0.243	0.228	0.237	0.260	0.260
Calculated UO_2 Thermal Expansion in. ³	0.148	0.172	0.176	0.210	0.333	0.323	0.251	0.318	0.342	0.344	0.384	0.384
Fuel Rod Volume Increase, in. ³												
Calculated Net	0.007	0.011	0.009	0.048	0.079	0.099	0.046	0.075	0.114	0.107	0.124	0.124
Experimental	0.0099	0.0320	0.1320	0.0911	0.1660	0.0974	0.0479	0.0958	0.2121	0.2660	0.3052	0.2902

$$1 \text{ inch}^3 = 16.4 \text{ cm}^3$$

over the total length of the fuel column to determine the total fuel rod volume in each zone. The volume was then multiplied by an expansion coefficient appropriate to the mean temperature in the interval. The following assumptions were made in the analysis:

1. UO_2 thermal conductivity from Lyons, et al.,⁽¹⁴⁾
2. Pellet-clad interface conductivity - 2000 Btu/h-ft²-°F (0.6 W/cm²-°C)
3. The fuel rod axial power distribution appropriate to the reactor control rod position at startup, full power operation, and
4. The actual fuel rod average heat flux for startup, full-power operation.

The analysis was performed on the Philco 2000 computer with the META code. The calculated UO_2 thermal expansion values for each fuel rod are listed in Table 18.

The internal free volume in each fuel rod was then calculated. This free volume consisted of that provided by the initial gap, plus that provided by densification of the fuel above 1600°C as already discussed. This available free volume was then subtracted from the calculated thermal expansion to determine the net swelling. A plot of this net calculated swelling versus the calculated molten volume for the fuel rods is shown in Figure 45. A least squares fit to the data provided the curve down through the data. The slope of this curve represents the incremental change in clad swelling attributable to thermal expansion minus densification. On this same figure the experimental data on clad swelling have been plotted and also fitted with a least square

Figure 45. UO_2 Volume Expansion Analysis

line. The slope of this line through the experimental data represents the incremental change in clad swelling from the melting phase change plus thermal expansion minus densification. Thus, the difference in the slopes of the two curves should represent the incremental change in clad swelling from the phase change alone. Since the data have been plotted versus the fuel rod molten volume, the difference of the slopes of the two curves can be interpreted directly as a percentage volume change. This method of analysis involving a comparison of the slopes of the two curves was selected to reduce the dependency on the absolute accuracy of the calculated thermal expansion data.

The slope of the experimental swelling curve is 0.145 ± 0.03 , i. e., the observed clad swelling increased at 14.5% of the molten volume increase. The calculated curve has a slope of 0.075 ± 0.005 i. e., the predicted increase from thermal expansion of the fuel and allowing for densification, is 7.5%. Thus, the volume increase attributable to the UO_2 phase change is 14.5 minus 7.5 or 7%. Considering the uncertainties in the analysis and the variance in the experimental data, this derived value is in agreement with the directly measured value of 9.6%.⁽⁸⁾

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APPENDIX A

FUEL ASSEMBLY POWER DETERMINATION

The GETR-PWL is unusual in that it is equipped with dual, completely independent instrumentation for determining the assembly thermal power output. The loop flow is measured by Gentile tube sensors, a modified form of a Venturi tube. Both the Gentile tubes and their associated differential pressure instruments were calibrated before starting the irradiation program. The calibration of each differential pressure instrument is checked routinely before every reactor cycle. Both differential pressure instrument outputs are continuously recorded in terms of flow rate: one on the main loop control panel, the other on the auxiliary loop panel in the reactor control room. Both flowmeter readings have always agreed within 3%; this same 3% value is the magnitude of the flow pulsation under steady operating conditions.

The water temperature rise through the test assembly is determined by a dual, differential temperature sensing system in which four precision resistance thermometers are the primary sensing elements. The thermometers are installed in wells in the facility tube inlet and outlet lines. The thermometers have been rotated periodically among the four positions; two of the thermometer bulbs are interchanged with two spare bulbs that are recalibrated when not in use. The associated resistance thermometer instrumentation is recalibrated routinely each shutdown by using an Electro-Scientific Instrument Company Deka Box, which is accurate to 0.001 ohm. The resistance box is used to simulate the differential resistance of the thermometers at normal coolant temperature rise conditions. The final zero output adjustment of the pairs of resistance bulbs is performed with the loop operating at temperature and rated flow just before reactor startup. The differential output of each pair of resistance bulbs is recorded continuously on the loop control panel. The agreement between the two measurements of the facility tube temperature rise has always been within 1°F (0.6°C) or 5% of the normal 25°F (14°C) temperature rise. The usual value is 0.4°F (0.2°C) or less than 2% difference.

The average value of the two water temperature rise measurements is combined with the flow measurement to calculate the gross heat input into the loop at the test section. This gross heat input value must be corrected for gamma heating contributions from the facility tube wall, test element hardware, and the coolant, and for heat loss from the facility tube between the inlet and outlet measurement points. Both the gamma heating contribution and the facility tube heat loss were determined several times in successive tests before operating the loop with fuel. The measured values were 26 and 12 kW, respectively, for gamma heating and heat loss or a net, negative correction of 14 kW to the gross heat input value. The resulting net heat input value represents the total heat generation of the four assembly fuel rods.

To determine the portion of the total heat generation attributable to each fuel rod, two different methods are used. In the first method, based on physics analyses, the total heat generation is proportioned according to the calculated neutron flux at each rod location. The assembly orientation in the facility tube is controlled so that the relative position of each fuel rod with respect to the GETR core centerline is known. In the second method, based on the post-irradiation gamma scans, the heat generation is proportioned according to the areas under the gamma activity profiles, i.e., according to the integrated fuel rod gamma activity. For this purpose, all four rods are gamma scanned in quick succession at identical instrument settings. The individual fuel rod heat generations predicted by each of the above methods agree within 3%, as shown in Table A-1. The data reported here are based on heat distribution indicated by the post-irradiation gamma scans. The resulting values represent the integrated heat output of each fuel rod over its total fuel length. Dividing the integrated heat output by the clad surface area over the active fuel length yields the average surface heat flux during irradiation.

TABLE A-1. Fuel Rod Power Split, Comparison of Physics and Gamma Scan Data

Assembly	Rod	Fuel Rod Power Output, %	
		Physics Analysis	Gamma Scan
EPT-6	6-A	22.7	23.25
	6-B	25.2	24.70
	6-C	25.5	24.95
	6-D	26.6	27.10
EPT-8	8-A	26.6	26.52
	8-B	25.5	25.96
	8-C	22.7	21.85
	8-D	25.2	25.66
EPT-10	10-A	22.6	23.68
	10-B	25.2	23.90
	10-C	25.06	26.19
	10-D	26.6	26.23
EPT-COM	6-E	17.0	17.53
	8-E	24.2	26.47
	10-E	28.9	26.84
	12-E	29.9	29.16
EPT-12C	12-A	26.6	27.3
	12-B	25.6	25.8
	12-C	22.6	23.0
	12-D	25.2	23.8

The post-irradiation gamma activity profiles were also used for correcting the chemical burnup analyses. In general, two burnup samples were taken from each rod destructively examined, one from the region of maximum burnup near the axial center of the rod and the other from a location very close to the upper end of the fuel column. Neither of these locations is representative of the fuel rod average burnup as determined by calorimetry or physics analyses. The chemical burnup data were corrected to average values for the rod by multiplying the burnup values by the ratio of the average ordinate under the gamma activity profile to the ordinate at the burnup sample location. The fuel rod burnups determined by chemical analyses, calorimetry, and physics are all listed in Table A-2. The chemical and calorimetric burnups are in excellent agreement. The burnup predicted by physics analyses is slightly lower in all cases.

TABLE A-2. Fuel Rod Burnup. Comparison of Physics, Chemical, and Calorimetric Data

$\frac{\text{MWD}}{\text{Te}}$, (fissions/cc $\times 10^{-19}$)				
<u>Assembly</u>	<u>Chemical Average</u>		<u>Physics</u>	<u>Calorimetry</u>
1. EPT-6			1180 (3.79)	1249 (4.01)
6A	1076	(3.54)		1162 (3.73)
6D	1334	(4.28)		1354 (4.35)
2. EPT-8	1555	(4.99)	1442 (4.63)	1532 (4.92)
8A	1660	(5.33)		1630 (5.23)
3. EPT-10	1390	(4.46)	1182 (3.79)	1459 (4.68)
10D	1460	(4.69)		1524 (4.89)
4. EPT-COM			1564 (5.02)	1725 (5.54)
12E	Zr 95	Ce		
	1785	2460		2014 (6.46)
	(5.73)	(7.90)		(11.46)
5. EPT-12C			5438 (2 cycle)	5225.4 (2 cycle)
			(17.17)	(16.77)

Following the calculation procedure outlined, the average surface heat flux for each fuel rod has been determined throughout its irradiation period. The calculations have been performed at sufficiently small intervals (every 2 hours) to reflect accurately the extent of the power variation and peaks during irradiation. These average heat flux values are plotted chronologically for each of the fuel rods destructively examined in Figures 9 through 13. These plots are the basic reference points for subsequent determinations of the fuel rod heat flux variation as a function of time and axial position.

To determine the fuel rod peak heat flux value at any time during the irradiation, reference must be made to the PWL axial power distribution and its variation during the reactor cycle (Appendix B). Figures 36, 37, and 38 are plots of the relative axial flux profile and thus, the

fuel rod power profile, as function of the reactor control rod bank position and fuel column length. Each of the profiles is normalized to an average value of 1.0 so that the fuel rod heat flux at any point can be determined simply by multiplying the ordinate on the flux profile by the average fuel rod surface heat flux. (The fuel rod surface heat flux values in this report were calculated with the assumption that the fuel rods did not undergo changes in diameter. For those fuel rods that experienced severe clad swelling, this assumption may result in an underestimate of the peak surface heat flux in the zone of swelling. The surface heat flux during operation could have been increased by the ratio of the final to initial diameter since the rod cross section probably remained completely filled with UO_2 . An estimate of this effect has been made in the last column of Tables 14 and 15. It is also assumed in the calculations for heat fluxes that the UO_2 is uniformly distributed along the entire length of the fuel rod, i. e., there is no central void while operating at temperature. This assumption is supported by UO_2 expansion analyses which indicate the rods are completely filled during operation. The axial fuel relocation and nonuniform central void observed in post-irradiation examinations are believed to form as a result of UO_2 contraction and shrinkage during solidification and cooldown.) During an actual reactor irradiation cycle, the lowest control rod bank position occurs at reactor startup, followed by gradual and continuous withdrawal of the rods during the cycle. As a result, the peak fuel rod heat flux occurs at startup, at an axial position approximately 9 inches (23 cm) above the bottom end of the fuel rod.

As the reactor cycle progresses, the fuel rod peak heat flux decreases in magnitude and shifts upward along the rod to a final position approximately 18-inches (46 cm) above the bottom of the rod. This shifting power profile is of extreme significance to the post-irradiation examination results on the individual fuel rods. It is particularly important for a correct interpretation of the post-irradiation UO_2 microstructure at various axial positions. To assist in the microstructure interpretation, heat flux curves have been prepared for each of the rods destructively examined as shown in Figures 39 through 43. The upper curve on each of the figures is the peak heat flux attained at each axial position during the fuel rod irradiation. These peak heat flux values did not, of course, occur simultaneously. The lower curve is the axial surface heat flux distribution at the time the fuel rod irradiation terminated. Only where these two curves coincide can the UO_2 microstructure be considered indicative of the peak conditions during operation.

It should also be noted that the peaking curves of Figures 36, 37, and 38 overlap to some extent. The flux profile at a 16-inch (41 cm) rod bank produces a higher peak heat flux at a point 12 inches (30 cm) up the rod than the subsequent curve for a 21-inch (53 cm) rod bank where the maximum in the curve is exactly at 12 inches (30 cm). This effect is considered when calculating the curves in Figures 39 through 43.

APPENDIX B

AXIAL POWER DISTRIBUTION IN THE PRESSURIZED WATER LOOP

Perhaps the single, most important requirement in the analyses and interpretation of experiments in the Pressurized Water Loop (PWL) is an accurate knowledge of the axial power distribution in the fuel rods. As constituted, it is impossible to obtain instantaneous flux profile measurements in the PWL while it is in operation. In the past, flux profiles determined in the Trail Cable Facility (outside the reactor pressure vessel) have been used to indicate the relative shift in the flux as a function of the control rod bank position. The Trail Cable flux profiles were corrected to the PWL on the basis of a comparison of flux profiles determined in both facilities at reactor startup conditions, i.e., at a very low control rod bank.

Because of the limitations in the above procedure, very extensive flux profile measurements were performed in the PWL before startup for General Electric Test Reactor (GETR) cycle 44 and in conjunction with the control rod calibrations accompanying the installation of new control rods. In these tests, the reactor core loading was varied by combinations of fresh and depleted fuel elements to provide four different control rod bank positions at criticality. The reactor was brought to critical at each of these four control rod banks and operated for approximately 2 hours at 50 kW. During each run, the reactor core was extensively instrumented with flux wires and a special flux wire irradiation assembly was installed in the PWL. After each run, the flux wires were removed for counting and were replaced with new wires. In this manner axial flux profiles in the PWL were obtained corresponding to control rod bank positions of 18.6, 23.0, 25.8, and 27.8 inches (47, 58, 65 and 71 cm).

While these flux profiles are a considerable improvement over the previously available data, the conditions of measurement still do not precisely represent actual operation, particularly near the end of life. Each of the measurements corresponds essentially to startup conditions without xenon buildup. In a high flux reactor such as the GETR, xenon poisoning is appreciable and may significantly perturb the flux profile. In addition, the measurements do not reflect the nonuniform burnup of the PWL element. Nonuniform burnup particularly influences the axial power distribution near the end of a cycle. Both these effects have been investigated analytically to determine whether any significant corrections should be applied to the experimental flux profiles to obtain truly representative axial power distribution curves. In the following paragraphs, the experimental flux profile measurements and results and, then, the supporting analytical investigation that was performed to confirm the axial power distribution curves used are described.

A PWL Flux Profile Measurements

The PWL flux measurements were performed with pure cobalt flux wires mounted in special aluminum assemblies designed to simulate the irradiation program hardware. This simulation permitted exact determination of the flux wire axial locations relative to corresponding positions on the fuel rods when installed in the loop. Each flux wire assembly contained four flux wires in positions corresponding to the axial centerline of the normal fuel rod locations. The radial orientation of the assembly was carefully controlled and recorded during installation for each measurement. Dimensional measurements were also made during the flux wire irradiations to determine accurately the flux wire assembly axial location relative to the GETR core.

After irradiation, the flux wires were removed from the assemblies and mounted on counting bars. The counting bars are designed to pass through a slot in a scintillation counter "pig." Using this arrangement, the count rate was determined for each 1/2-inch increment of the wire, corrected for background, and then plotted as a function of axial position. The total count at each position was sufficient to reduce the standard deviation to less than 1% of the count. A smooth curve was then fitted to the data and the location of the top and bottom of the GETR core determined on the plots. Figure B-1 illustrates the data for one wire from the assembly installed during operation at an 18.6-inch (47.3 cm) control rod bank. The area under each flux curve between the top and bottom of the GETR core was determined with a planimeter. The ratio of the area under an individual curve to the sum of the areas for all four wires from the same assembly was taken as a measure of the relative power of the particular radial location where the wire was irradiated. The radial power variation indicated from all four flux wire assemblies is given in Table B-1.

TABLE B-1 Fuel Rod Power Split Indicated
by PWL Flux Measurements

Wire Location Adjacent to Core Position	Flux Wire Relative Power Control Rod Bank Position			
	18.6 in.(47.3 cm)	23.0 in.(58.5 cm)	25.8 in.(65.5 cm)	27.8 in.(70.6 cm)
E-4	1.02	1.04	1.04	0.99
D-3	1.05	1.07	1.03	1.05
E-2	0.97	0.93	0.89	1.02
F-3	0.95	0.96	1.04	0.95

It is evident from Table B-1 that the relative power for a particular wire orientation was not constant for the four sets of measurements. This variation could result from misorientation of the flux wire assembly during installation or may be a real variation attributable to differences in the reactor fuel loading to obtain the four control rod positions. The true cause is uncertain and no further effort was directed to using the flux wire data to determine the radial power.

Figure B-1. Pressurized Water Loop Flux Wire Data - 18 5-inch Control Rod Position

variation between rods. (The relative power of the irradiated fuel rods was determined from the post-irradiation gamma scans as discussed in Appendix A.)

By using the areas under each flux profile, the average ordinate was determined and used to normalize the profiles to an average value of 1.0. On a normalized basis, the axial flux profiles from all four wires in an assembly were identical. The normalization was actually performed for four different fuel lengths, i. e., the full 36-inch (91 cm) core length and then for the three fuel column lengths used in this irradiation program, 34-1/4, 31-1/4, and 29-1/2 inches (87, 79, and 75 cm). The axial peak-to-average values become smaller as the fuel column length decreases because fuel at the very top of the rod operates at less than average power. The resulting normalized flux profiles for the three fuel column lengths and all four rod bank positions are shown in Figures 36, 37, and 38. The influence of fuel column length of the axial peaking factor is best illustrated by the plot in Figure B-2. Here the maximum axial peak-to-average value is plotted versus control rod bank location for all three fuel column lengths. In addition, a curve is included on this figure for a full 36-inch (91 cm) fuel column length in the PWL and also a curve for the peaking factor versus rod bank in the Trail Cable Facility. This latter curve is also for a full 36-inch (91 cm) long fuel column and is based on data obtained at the same time the PWL measurements were made. A fifth datum point was obtained for the Trail Cable curve at a 31.5-inch (80 cm) rod bank location near the end of the subsequent reactor cycle. This datum point is included on the Trail Cable curve, and the PWL curves have been extended essentially parallel to determine the axial peaking factor at rod bank locations above those actually measured.

The curves in Figures 36 through 38, and B-2 represent the final results of the experimental flux profile measurements and are the reference data for subsequent analytical investigations.

B. Analytical Corrections to the Flux Profile Measurements

To account for the effect of xenon poisoning on the flux shape, two 2-dimensional GETR flux calculations were made. In the first of these calculations, a core loading and control rod bank were assumed to simulate the highest control rod bank for which experimental flux measurements were made. In the initial calculation no xenon poisoning was assumed and the resulting axial flux profile was compared to that actually measured to ensure reasonably close correspondence. The same calculation was then repeated with the addition of distributed xenon poison in the core. The two calculated flux profiles were then compared to determine the incremental effect of the distributed poison. The changes between the two flux profiles were negligible.

To determine the influence of burnup on power distribution, the point-to-point change in enrichment along a fuel rod was calculated by using an assumed average burnup of 2000 MWD/T_e (0.54×10^{20} fissions/cm³) and the normalized, integrated average flux profile for a single reactor cycle, shown in Figure B-3. The resulting enrichment distribution was again normalized and then multiplied by the end-of-life PWL flux profile to obtain a best estimate of the actual end-of-life axial power distribution. The effect of the calculated enrichment variation was a reduction of the predicted peak heat flux values at the midpoint of the rod by 3% and an increase in the predicted values at the extreme ends of the rod by 3 to 4%.

Because of the small magnitude of this correction and the very good probability that it is reduced still further by axial redistribution and equilibration of the molten fuel, it was again decided to neglect the correction and use the experimentally determined axial flux profiles as the actual fuel rod axial power profile.

Figure B-2. Maximum Axial Peaking Factor as a Function of Control Rod Bank Position

Figure B-3. Integrated Average Flux Profile for a Single GETR Cycle

APPENDIX C

IRRADIATED FUEL ASSEMBLY TRANSFER AND DISASSEMBLY PROCEDURE

To preserve the cladding surface condition and microstructure resulting from the irradiation in the Pressurized Water Loop, the surface temperature of the cladding after irradiation should not exceed 400°F (204°C). Consequently, dissipation of the residual gamma heating in the fuel after the irradiation is of primary concern in handling the fuel assembly after reactor shutdown.

To stay below the 400°F temperature limit during the reactor shutdown, the assembly remained in the loop and the coolant circulation was continued. By allowing sufficient decay time, the residual decay heat decreased enough to transfer the assembly from the loop to the GETR pool. Here, with the pool used for a heat sink, additional cooling time was accumulated until the gamma heating decreased to produce no greater than 500 Btu/h-ft² (0.16 W/cm²) peak heat flux. This ensures limiting the cladding surface temperature to less than 400°F (204°C) when in air, the cooling medium during the post-irradiation examination in the Radioactive Materials Laboratory.

For shipment to the Radioactive Materials Laboratory the irradiated fuel assembly is placed in a transfer can which is sealed to contain any radioactive gases which might be released. This can is placed in a cask for transfer from the reactor to the Radioactive Materials Laboratory approximately 1/2 mile away. The cask is placed in the interlock of a high-level, 1-MCi cell and the assembly removed for examination.

Disassembly and removal of the fuel elements is accomplished by loosening the lock nut and washer on the "Tee" handle to free the support pin at the top of the fuel element. Then the element is rotated until the support clears the washer and it can be withdrawn from the flow tube. This procedure is used successfully to remove fuel elements which remained dimensionally stable. For fuel elements where sufficient swelling occurred to produce interference between the cladding and the spacer pins in the flow tube it is necessary to cut the flow tube with an air-driven abrasive wheel. This relieves the stresses enough to extract the fuel element from the flow tube for examination.

APPENDIX D

VISUAL AND DIMENSIONAL EXAMINATION

A. Pre-Irradiation

The pre-irradiation examinations were made in the Radioactive Materials Laboratory and included detailed visual inspections, and length and diameter measurements of the fuel rods.

During the initial, visual examination, any surface imperfections scratches, discolorations, etc., were noted and their location recorded. The elements were checked for straightness on a surface plate, and any bowing in excess of 1/32-inch (0.08 cm) was measured and recorded. The over-all length of each fuel rod was measured in a fixture which included a length standard and a dial gage. Length measurements were accurate to ± 0.001 -inch (0.0025 cm). A micrometer was used to obtain outside diameter (o.d.) measurements at 4, 16, 20, 24, 28, 32, and 36 inches (10, 41, 51, 61, 71, 81, and 91 cm) from the top of the fuel rod. To detect ovality in the rod, these measurements were made at two orientations: (a) the micrometer axis placed parallel (0 to 180 degrees) and (b) placed perpendicular (90 to 270 degrees) to the support pin through the top end plug. These measurements of the element outside diameter are precise to ± 0.0002 -inch (± 0.0005 cm).

B. Post-Irradiation

Visual examinations were made in the hot cell with a Kollmorgen periscope at magnifications up to 6X. The following items were specifically noted and recorded:

1. Crud deposition, if any,
2. Zircaloy-2 oxide film condition,
3. Clad appearance at spacer bearing areas and end plug welds,
4. Apparent straightness of fuel rods, and
5. Clad damage, type, and axial location.

The diameter and over-all length measurements were made on the irradiated fuel elements with equipment identical to that used for the pre-irradiation measurements. However, the equipment is operated remotely in the cell with master-slave manipulators. Under these conditions, the outside diameter measurements are accurate to ± 0.0005 -inch (± 0.0013 cm) and the over-all length can be measured to ± 0.010 -inch (0.025 cm). The axial and radial positions for post-irradiation measurements were taken as close as possible to the pre-irradiation locations. For fuel rods with significant clad swelling, the outside diameter measurements were taken at smaller axial intervals to determine the shape and extent of the swollen zone.

The basic dimensional data obtained on each of the fuel rods irradiated during the program are recorded in Tables 7 through 11 with measurements rounded to the nearest mil. The dimensional changes observed are shown in Figures 14 through 18, to indicate the axial distribution of clad swelling in the fuel rods.

APPENDIX E

GAMMA ACTIVITY SCANNING

In the Radioactive Materials Laboratory the fuel rod gamma activity profile or scan is obtained by drawing the fuel rod past a narrow, lead-shielded slit in the 3-foot-thick cell wall. The size of the usual slit is 0.010 by 0.750-inch by 3 feet (0.025 by 1.9 by 91 cm) with the 3/4-inch (1.9 cm) dimension of the slit perpendicular to the fuel rod axis. The slit width and the drive speed of the rod past the slit are adjustable to permit optimum scan resolution over a wide range of specific activities of fuel rods. The collimated gamma beam from the slit impinges on a NaI crystal which feeds a discriminator and count rate meter. The output from the count rate meter is converted to millivolts and printed on an adjustable chart speed, strip chart recorder. The fuel rod speed past the slit and the chart speed can be synchronized to provide a 1:1 correspondence of the gamma trace to the rod axial position. A diagram of the gamma scanning equipment is shown in Figure E-1.

The discriminator level is set to accept pulses corresponding to gamma ray energies above 0.7 MeV. Thus, the trace is predominately a measure of Zr-95 and La-140 activities. By limiting the measured gamma energy to that above 0.7 MeV the scan is made relatively independent of the rotational orientation of the fuel rod about its axis. Calculations indicate a maximum self-absorption factor of 0.6 for the rod. Actual measurements on a fuel rod at different rotational orientations have shown no detectable change in the trace.

In the past, gamma scans have been used primarily to assist in sectioning the rod to indicate the top and bottom of the fuel column, pellet interfaces, and unusual pellet separation. In addition, the gamma scan is normally used to indicate the peak-to-average power value at the location of the chemical burnup sample. For this program, the gamma scans have been used also to indicate the power split among the four assembly fuel rods. To do this, the most active fuel rod in each assembly was selected, and the equipment settings adjusted while scanning this rod, to provide a maximum reading of 2/3 of full scale on the recorder chart. All four assembly fuel rods were then scanned full length, consecutively, at these same settings. The resulting gamma scans are directly proportional to the integrated gamma activity of the fuel rods, and, hence, to the fuel rod power outputs during irradiation. Since the proportionality factor is the same for all four rods, the gamma scans indicate the relative power output of each rod to the others. The ratio of the area under an individual gamma scan to the total area under all four scans was used as the measure of the fraction of the net assembly power produced by the rod.

Figure E-1. Radioactive Materials Laboratory Gamma Scan Equipment

APPENDIX F

FISSION GAS RELEASE

The amounts of fission gases released from the fuel as the result of irradiation are determined by puncturing the fuel element cladding, collecting aliquots of the released gases, and analyzing them for krypton and xenon. In addition, analyses are made for helium originally present as a filler gas, and also oxygen and nitrogen from the residual air present after fuel element fabrication.

The gas sampling system for puncturing and collecting radioactive gases from irradiated fuel elements is shown in Figure F-1. The system consists of in-cell puncture equipment which includes a puncture mechanism, a particulate filter, and an iodine trap. The out-of-cell equipment consists of a vacuum system with a sampling section, pressure measuring devices, vacuum pumps, and an inert gas purging system. Auxiliary equipment, such as the refrigeration unit for the iodine trap and the hydraulic pumps and lines for the puncturing mechanism, is also located outside the cell for ease of operation and maintenance.

The following procedure was used to obtain the fission gas sample from the fuel element. The element was placed in the puncture mechanism in position suitable for a side puncture near the top end plug, and a vacuum seal was made against the cladding. The entire collection system (up to the cladding) was then evacuated to less than 10μ of mercury and the leak rate determined by isolating the vacuum pumps from the system and measuring the pressure rise with time. Leak rates less than $0.5\ \mu/\text{min}$ are considered satisfactory and the cladding was then punctured with a hydraulically driven pin to release the gases in the specimen. Approximately 30 min is required for the gases from pellet fuel rods to diffuse into the calibrated volume of the collection system and reach an equilibrium pressure condition as indicated by the Zimmerli pressure gage. The total gas collected in the system was determined from these pressure-volume measurements.

Two aliquots of the gas in the system were taken for Kr-85 analysis on a 256-channel gamma spectrometer. The 0.513-MeV gamma photons associated with Kr-85 decay were counted and the number of Kr-85 atoms in the sample determined by comparing the peaks of the sample with those of a calibrated standard. The total number of Kr-85 atoms released from the fuel element was then calculated from the aliquot fractions (Table 12). The over-all precision of the Kr-85 measurement, which includes sampling and Kr-85 counting, has been determined by a statistical analysis of data obtained from previous measurements and reported to be $\pm 5\%$.

Two aliquots of the gas were also taken for chromatographic analyses to determine the hydrogen, nitrogen, carbon monoxide, krypton, xenon, and oxygen-argon content. Oxygen

Figure F-1. Fission Gas Collection System

and argon cannot be distinguished by using this analytical technique; therefore, they are reported together. With helium as the carrier gas, each sample is passed through a molecular sieve to separate the H_2 , O_2 -A, N_2 , Kr, and Xe. By this technique, gas volumes of 1.0×10^{-3} cc are detected accurately to $\pm 5\%$. By using a silica gel column and a helium carrier, CO_2 and CO can be detected. These analytical procedures have been reported previously. *

The minimum volumes of various gases detectable by this method are:

<u>Component</u>	<u>Volume, cm^3 (STP)</u>
H_2	1.0×10^{-3}
$O_2(A)$	1.0×10^{-4}
N_2	1.0×10^{-4}
CH_4	1.0×10^{-3}
CO	1.0×10^{-3}
Kr	1.0×10^{-4}
Xe	1.0×10^{-3}

The total fission gases released from each fuel element is determined by using the Kr-85 content obtained by gamma spectrometer analysis, and the Xe-to-Kr ratio obtained from the gas chromatograph. These results are shown in Table 12.

* F. A. Brandt, P. W. Mathay, and D. L. Zimmerman, "A Compilation of Techniques used by Vallecitos Radioactive Materials Laboratory," GEAP-3683, February 1963.

APPENDIX G

BURNUP ANALYSIS

In general, two burnup samples were taken from each fuel rod destructively examined: one from the expected maximum burnup zone near the axial midpoint of the rod and the other from the coldest zone of the rod near the top of the fuel column. The two burnup samples were used to detect any influence of axial fission product migration on the burnup analyses. The sample from the maximum burnup region is more likely to be affected by fission product migration because of its higher internal temperature level.

To obtain representative samples for the volume average burnup, complete transverse sections [about 1/2 inch (1.27 cm) long which included the fuel and cladding] were cut from the fuel elements. Each section was dissolved individually in a 10 N nitric acid solution containing 10 ppm Ce, Cs, and Zr carriers. The dissolution proceeded at room temperature until the fuel was completely removed from the cladding and in solution in the acid. The fission products are also leached from the cladding surface by following this procedure. This technique precludes possible sampling problems which could result from radial redistribution of fission products from the hot fuel to the cooler cladding surface.

The acidified fuel solutions were agitated and aliquots taken for fluorometric analysis for total uranium content and the determination of the Zirconium-95, Cesium-137, and Cerium-144 with a gamma spectrometer. Based on this radiochemical data, the fuel burnups in atom per cent were calculated and are summarized in Table G-1. The fission product parameters^{*} used in these calculations were for Zirconium-95, a 65-day half-life and a fission yield of 0.060, for Cesium-137, a 29.2-year half-life and a fission yield of 0.0615, and for Cerium-144, a 285-day half-life and a fission yield of 0.056. To convert the burnup values for atom per cent to fissions per gram of U, an average value of 238.00 was assumed for the atomic weight of this slightly enriched fuel. To convert the burnup to megawatt days per tonne, the effective energy release per fission was assumed to be 205 MeV.

Because of their axial location, neither of the burnup samples represents the average burnup in the fuel rod. To correct the values to the average burnup, the area under the fuel rod gamma scan was determined with a planimeter and then divided by the base length to determine the average ordinate height in millivolts. The ratio of this average ordinate height to the ordinate height at the burnup sample location was taken as the ratio of the rod average burnup to that determined by the sample analysis. Multiplying the measured burnups by the ratio, provided the rod average burnup values listed in Table A-2.

^{*}B. F. Rider, C. P. Ruiz, and H. W. Alter, "Measurement Methods of Burnup and Heavy Element Isotopic Composition in Irradiated Power Reactor Fuel," TID-7672, February 1964 pp 275-287

The burnup calculation method is shown in the following paragraphs.

The atom percent burnup is defined as the number of fissions per 100 fissionable and fertile atoms. This can also be expressed as:

$$\text{Atom \% Burnup} = \frac{\text{fissions/milliliter solution}}{(\text{initial fissionable} + \text{fertile atoms}) / \text{ml solution}} \times 100 \quad (1)$$

The fissions per milliliter are determined by analyzing for Cs-137, Zr-95, and Ce-144 as burnup indications. The initial fissionable and fertile atoms per milliliter are determined by post-irradiation analysis for uranium atoms per milliliter to which are added the fissions per milliliter. Since each fission removes a uranium atom from the system, this sum is the initial concentration of uranium per milliliter of solution when corrected back to the start of the irradiation before any fissions have occurred.

Sample calculations for the peak burnup region of element 6-D were based on the following radiochemical analysis results:

<u>Burnup Indicator</u>	<u>Fissions per milliliter</u>
Zr-95	6.14×10^{13}
Cs-137	5.58×10^{13}
Ce-144	5.62×10^{13}
Average fissions per milliliter	$= 5.78 \times 10^{13}$
Final uranium content	$= 2.60 \times 10^{16}$ atoms per milliliter

By substituting these results into Equation (1) we obtain

$$\begin{aligned} \text{Atom \% burnup} &= \frac{5.78 \times 10^{13} \text{ fissions per milliliter} \times 10^2}{(5.78 \times 10^{13} \text{ fissions per milliliter} + 2.60 \times 10^{16} \text{ atoms per milliliter})} \\ &= 0.222 \text{ atom \% burnup of the uranium.} \end{aligned}$$

The burnup determinations are frequently normalized and reported as fissions per gram of uranium initially present. This is accomplished by first determining the number of uranium atoms in a gram of uranium. The atomic weight of uranium in slightly enriched fuel ranges from 238.07 for 0.72% U-235 to 237.94 for 6% U-235. Over this range, an average value of 238.00 is used for the atomic weight of uranium which introduces negligible error (0.03%). On this basis the number of uranium atoms per gram is:

$$\frac{6.0248 \times 10^{23} \text{ atoms}}{\text{gram atomic weight}} \times \frac{\text{gram atomic weight}}{238.00 \text{ gm of U}} = 2.53 \times 10^{21} \text{ U atoms/gm}$$

TABLE G-1. Fuel Burnup Analysis EPT-6 Assembly

Element	Location, Inches From Top Of Element	Burnup Indicator	Fissions/Gram	Burnup	
				at. %	MWD/T
6-A	9.4	Cs-137	2.22×10^{18}	0.081	783
		Ce-144	1.90×10^{18}		
		Av	2.06×10^{18}		
6-A	21.7	Cs-137	3.59×10^{18}	0.145	1399
		Ce-144	3.77×10^{18}		
		Av	3.68×10^{18}		
6-D	7.1	Zr-95	2.45×10^{18}	0.090	867
		Cs-137	2.00×10^{18}		
		Ce-144	2.38×10^{18}		
		Av	2.28×10^{18}		
6-D	20.1	Zr-95	7.00×10^{18}	0.222	2133
		Cs-137	5.40×10^{18}		
		Ce-144	5.46×10^{18}		
		Av	5.61×10^{18}		

The number of fissions per gram is determined from the follow relationship

$$2.53 \times 10^{21} \frac{\text{U atoms}}{\text{gram}} \times \frac{\text{milliliter}}{2.6058 \times 10^{16} \text{ U atoms initially}} \times 5.78 \times 10^{13} \text{ fissions/ml} \\ = 5.61 \times 10^{18} \text{ fissions/gram.}$$

Burnup is frequently expressed as megawatt days per tonne of uranium (MWD/T_e). * To convert fissions gram-U to MWD/T_e it is first necessary to compute the energy release per fission.

* Metric Tonne - 2200 lb.

According to Weinberg and Wigner * the distribution of fission energy is

Kinetic energy from fission fragments, MeV	167
Prompt gamma rays, MeV	6
Kinetic energy of fission neutrons, MeV	5
Fission product decay	
gamma, MeV	6
beta, MeV	8
Neutrinos, MeV	<u>12</u>
Total energy per average fission, MeV	204

In a nuclear reactor, the maximum available energy per fission is total energy minus the neutrino energy or 192 MeV. However, the capture of neutrons in the reactor, particularly in the (n,γ) process, produces additional energy which increases the total energy available to 205 MeV. Based on this value, the number of fissions per megawatt day can be expressed as:

$$\frac{1 \text{ fission}}{205 \text{ MeV}} \times \frac{6.2422 \times 10^{18} \text{ MeV}}{\text{MW-sec}} \times \frac{86400 \text{ sec}}{\text{day}} = 2.63 \times 10^{21} \text{ fissions/MWD}$$

The peak burnup on element 6-D can be expressed in MWD/ T_e by:

$$\frac{\text{MWD}}{T_e} = \frac{5.61 \times 10^{18} \text{ fissions}}{\text{gram-U}} \times \frac{1.0 \times 10^6 \text{ grams U}}{T_e} \times \frac{\text{MWD}}{2.63 \times 10^{21} \text{ fissions}} = 2133 \text{ MWD}/T_e$$

The average burnup in the fuel element was obtained from the radiochemical analysis and the gamma activity curve by the following procedure. A planimeter was used to measure the area under the curve, which was then divided by its length to obtain the average ordinate, 19.7 mV for fuel element 6-D. This ordinate is calibrated by dividing the burnup at a known location on the element by the ordinate of the gamma activity curve corresponding to this location. For element 6-D, the calibration is 0.090 at %/15 mV = 0.006 at %/mV and the average burnup is 19.7 mV $\times$ 0.006 at %/mV = 0.12 at %.

* A. M. Weinberg, and E. P. Wigner, "The Physical Theory of Neutron Chain Reactors," University of Chicago Press, 1958, pp. 137-138.

APPENDIX H

METALLOGRAPHIC EXAMINATION

Metallographic examinations of the fuel elements are used to follow the changes occurring in the Zircaloy-2 cladding and UO_2 fuel resulting from various irradiation testing conditions. To meet this program objective the metallographic examination of each individual specimen must provide a record of the macrostructure of the element and the microstructure of the materials in a manner amenable to comparison with the other fuel specimens in the program. With this in mind, the following systematic approach was devised for these metallographic examinations.

Each fuel element specimen is placed in a pre-drilled Bakelite metallography mount and held in position by Epoxy resin which is applied by vacuum impregnation to improve fuel retention and preserve cladding edges during grinding and polishing. The Epoxy resin formulation is Hysol 2038 resin with 3410 hardener at a 9:1 ratio by weight. This hardens at room temperature in approximately 24 hours.

After mounting, the metallography specimen is processed through the final grinding operation off silicon carbide 600 grit paper with a kerosene lubricant. At this stage of surface preparation, a photomicrograph is taken at 6 to 8X. Oblique illumination is used to delineate the UO_2 grains, cracks, voids, and gross porosity. The element is observed as an entity and examined for evidence of correlation between the fuel macrostructure and any swelling, distortion, or cracking that may be present in the cladding.

After processing through the final polish with 0.5μ diamond abrasive in kerosene, an autoradiograph was made by contacting the polished surface of the specimen on a Kodak high-resolution plate film (Appendix I). After suitable development, the resulting image shows the distribution of the radioactivity in the sample. A photographic enlargement of the autoradiograph was made to correspond with the photomicrograph at 6 to 8X for direct comparison and evaluation of activity distribution and macrostructure. The following sequence of examinations was performed on a Bausch and Lomb research metallograph with a shielded stage for the radioactive specimen.

The fuel is examined in the as-polished condition to detect any metallic phases present in the UO_2 . No water lubricant is used in the preparation of these fuel specimens to preserve any metal phase in the structure. The size, shape, and distribution of porosity in the fuel are examined at this time.

The UO_2 fuel is etched by 10% HNO_3 , 20% H_2O_2 and 70% H_2O solution for approximately 1 minute. This delineates the grain boundaries and any second phases present. In the etched condition the fuel is examined for grain size, shape, and distribution. The presence of porosity,

inclusions, and second phases in the microstructure and their distribution with respect to the grain structure is observed. A radius or a diameter across the fuel section is photographed at either 100 or 50X to obtain a composite record of representative microstructures. The location of this composite is also established on the macrograph of the section. The as-sintered grain size near the cladding and the zone showing equiaxed grain growth are photographed at 250X, the reference grain size magnification.

The microstructure of the Zircaloy-2 cladding is revealed by the following two etchants: The first etchant is 30 lactic acid, 30 HNO_3 , and 0.4 HF (parts by volume) applied for 10 seconds. After rinsing with distilled water, the second etchant 12 HNO_3 , 12 H_2O and 1 HF (parts by volume) is applied for 10 seconds. This etching procedure removes the smeared metal from the surface of the cladding and reveals the zirconium hydride phase. These etchants also attack the UO_2 fuel hence the microstructure of the fuel is examined before the cladding metallographic examination.

Metallographic examination under polarized light reveals the Zircaloy-2 grain size and shape as well as evidence of deformation, such as distortion of the grains and mechanical twinning. A reference grain size photomicrograph is made at 250X. Examination under bright field illumination discloses the presence of the zirconium hydride phase and its distribution in the cladding.

Unique characteristics in cladding and fuel microstructure such as metallic inclusions, porosity, second phases, and interaction between cladding and fuel and their environment were examined and photographed at magnifications up to 750X. All photomicrographs were correlated with the macrograph. In special cases, microhardness measurements were used to detect interstitial atoms in the matrix of the Zircaloy-2 cladding.

APPENDIX I AUTORADIOGRAPHY

After metallographic sample preparation is complete (Appendix H), the gross beta-gamma activity is measured by using a Victoreen rate meter with its plastic-bagged probe in direct contact with the sample. An exposure chart gives an estimate of exposure time. Preparations for the exposure are made using a Kodak safe light with a 60-watt bulb and a Wratten Series 1A filter as a source of illumination. The sample is placed, prepared face down, on the emulsion side of the Kodak high-resolution photographic plate which is positioned on a sheet of Masonite or similar material to prevent backscattering into the emulsion. After exposure, the plates are stored in a light-proof box and removed from the cell for processing in a radioactive controlled access area.

The plates are developed for about 5 min in Kodak D-19 or D-50 developer at 68°F (20°C) with continuous agitation, then rinsed in an acid stop bath, and fixed in Kodak Rapid Fixer for 4 min. The plates are next washed in running water for about 30 min, and then bleached in Kodak Farmers R4 to remove a thin surface layer of the emulsion which is generally fogged by scattered radiation. The plates are then ultrasonically cleaned to remove any residual contamination. After this operation, plates may be handled safely in clean area. All processing solutions are discarded as hot liquid waste.

Autoradiographs prepared by this process are suitable for printing and enlarging by using standard darkroom techniques. In the finished print the light areas correspond to areas of high radioactivity.

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publication as specified below.*

ITEM	REFERENCES (SECTION, PAGE, PARAGRAPH, LINE)	INSTRUCTIONS (CORRECTIONS AND ADDITIONS)
1	Figure 21a Page 89/90	Change "Heat Flux at End of Life" value to 696,600 Btu/h-ft ² (220 W/cm ²)
2	Figure 24a Page 113/114	Change "Peak Heat Flux Zone" to (MET No. 8811)