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IN-PILE SWELLING OF UO_2 FUELS

ETR TESTS OF HIGHLY RATED UO_2 FUEL SAMPLES

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ABSTRACT

An experiment was designed to obtain basic irradiation data which could be used to establish fuel swelling correlations for cylindrical UO_2 fuels. The design parameters specified for the experiment are summarized. Unexpectedly high flux levels in the test positions used in these ETR tests resulted in fuel melting and produced a sufficiently large number of cladding failures to severely compromise the ability of the experiment to yield a useful fuel swelling correlation. The test program was discontinued at this point, since consideration of the effect of existing failures in combination with the likelihood of other test pins having sustained some damage led to the conclusion that the attainment of significant UO_2 fuel performance data was highly unlikely. The evaluation of the failed test pins and the failure analyses leading to the decision to terminate the test are described.

1. INTRODUCTION

Bulk UO_2 swelling at high burnups is a potential problem in fuel designs since the anticipated low ductility of sheath materials at high irradiation doses precludes extensive plastic deformation of the clad as a normal design condition. Therefore, fuel volume expansion must be either accommodated internally or restrained by elastic stresses in the sheath. The purpose of the present study was to establish general correlations which could quantitatively describe and predict the swelling of cylindrical UO_2 fuel during irradiation.

The variables chosen were fuel density, clad restraint, power level, and burn-up. All are variables which are known to be significant in influencing fuel swelling. The ranges of variables originally selected are summarized below:

TABLE 1-1

FUEL SWELLING PARAMETERS

UO_2 Density (percent theoretical)	90	94	97
Power Level (kw/ft)	12	18	24
Burnup (MWD/MTU)	20,000 - to - 50,000		
Restraint (equivalent thickness of 304 stainless steel, mils)	0	16.0	50
Sheath Material	304 Stainless Steel, Zircaloy-4		

A fractional replicate experimental design was selected to include the critical combinations of parameters required to construct swelling correlations. On the basis of the statistical design, 68 fuel pins were specified for irradiation.

The basic capsule design is a twelve-inch long, four rod cluster of fuel pins. NaK (Na-56 w/o K) bonding was used in some cases to achieve fuel and sheath temperatures which simulate those in pressurized water reactors and also to provide a protective outer can in the event of primary sheath failure. All four rods in a given capsule were designed to operate at identical heat fluxes.

All capsules were irradiated in the ETR aluminum reflector using four-hole reflector pieces. Design parameters, heat transfer data, and flux-time requirements for each of the 68 fuel pins are given in Appendix A. Figures B-1 to B-4 of Appendix B illustrate the mechanical design of the test capsules.

2. SUMMARY OF EXPERIMENTAL DESIGN

The experimental design of this study can be broken down into the areas of thermal-hydraulic design, mechanical design, and nuclear design. Each of these is discussed separately below.

2.1 Thermal-Hydraulic Design

Thermal-hydraulic calculations pertinent to the experimental design were made.^[1] Based upon these calculations, the restraints imposed upon the design were:

1. No nucleate boiling was to occur; cladding burnout safety factors were selected to ensure no burnout.
2. The fuel pins were expected to accommodate stresses imposed by center melting and thermal expansion in the event of a postulated maximum heating condition of 45 percent overpower.

To orient the reader of this topical report, a listing of the thermal characteristics of the highest rated experiment is given in Table 2-1. The thermal-hydraulic characteristics of the Engineering Testing Reactor (ETR), as originally presented, are given in Table 2-2.

2.2 Mechanical Design

The original mechanical design of this experiment is illustrated by the assembly drawings shown in Figures B-1 through B-4 of Appendix B. Figure B-1 is a general assembly drawing which shows the four-pin geometry and the supporting basket. This basket design permits utilization of the entire flow area of the irradiation facility. Figure B-2 shows the two types of fuel pins, heavy wall-single clad and double clad with a NaK annulus; Figure B-3 illustrates a fuel pin-flow blocker assembly, four of which are attached as units to the central hanger.

TABLE 2-1

THERMAL CHARACTERISTICS OF HIGHEST RATED EXPERIMENT

Specific Power Generation	25.2 kw/ft per pin
Heat Flux at Sheath-Water Interface	698,100 Btu/hr-ft ²
Heat Output of Fuel	39,450 Btu/hr per pin (157,800 Btu/hr per capsule)
Cladding Surface Temperature	240°F
Fuel Surface Temperature	1187°F
Fuel Center Temperature	4555°F
Burnup Rate	6.38×10^{18} fissions/cc per EFPD
Fuel Diameter	0.340 in.
Pin Outside Diameter	0.471 in.

TABLE 2-2

THERMAL-HYDRAULIC CHARACTERISTICS, IRRADIATION FACILITY DATA

Reactor Power	175 MW
Aluminum Reflector Facility Flow Rate	27.2 gpm
Primary Coolant Pressure (Inlet)	190 psia
Saturation Temperature Primary Coolant (170 psia)	368°F
Coolant Velocity Over the Fuel	21.9 ft/sec
Heat Transfer Film Coefficient	6456 Btu/hr-ft ² -°F
Inlet Water Temperature	120°F
Outlet Water Temperature	132°F
Pressure Drop Across Basket Assembly	43 psia

The different sheath arrangements are required to interject the fuel restraint variable into the tests and to include both Zircaloy and stainless steel as sheath materials. Dimensional data for each pin are presented in Appendix A. Following the first and second failure incidents, slight modifications to the design were adopted. These changes are discussed in detail in the discussion of the experimental results and are shown in Figure B-4.

Optimum Simulation of Actual Fuel Rods:

The present tests were designed to simulate long fuel rods by: 1) use of an inverted spacer (Figure B-2, Item 9) which provides a gas plenum without leaving free expansion space for the solid fuel should gross swelling occur. However, a gap was left at the top of the fuel column to accommodate thermal expansion of the fuel. 2) inner pin geometry and tolerances (Figure B-2) which permit both inner and outer sheaths to exert axial restraint on the fuel.

2.3 Nuclear Design

Fuel enrichments were calculated to achieve design heat ratings in various arbitrary thermal neutron fluxes between 1.00×10^{14} to 2.10×10^{14} nv (which is the range of fluxes available in the ETR aluminum reflector). This was done to provide ETR the latitude of placing each capsule in any available reflector position which satisfies the flux requirement to an original tolerance specification of ± 5 percent. Detailed enrichments, power levels, and fluxes are listed in Appendix A, and the methods by which they were calculated are described in Appendix C.

3. DISCUSSION OF RESULTS

As previously described, the original intent of this program was to obtain basic irradiation data which could be used to establish fuel swelling correlations for cylindrical UO_2 fuel shapes. It was anticipated that data would be obtained for fuel burnup values in the range of 20,000 to 50,000 MWD/MTU. It was not possible to achieve these burnup levels or the experimental fuel swelling data because of the test element failures which occurred.

A description of these failures is given below.

3.1 First Incident

Six capsules completed the first irradiation period, ETR cycle 69, without incident. Four of these were intended to operate at 12 kw/ft, one at 18 kw/ft, and one at 24 kw/ft. Included in this group of assemblies were three control capsules (WAPD-49-13, 14 and 15) which were designed to operate at 12, 18, and 24 kw/ft respectively, for one irradiation cycle only. Following completion of cycle 69, the three control capsules were removed from the test reactor and returned to the Westinghouse Post-Irradiation Facility (WPIF), as originally intended. The three remaining capsules, designed for 12 kw/ft operation, were stored at ETR. A second group of specimens, consisting of nine assemblies, was then inserted into ETR. After six days at 162 MW reactor power, iodine activity was detected in the ETR primary coolant and the reactor was shut down. All capsules were removed from the reactor pending identification and correction of the causes of the failures.

Post-irradiation examination showed that a failure was located in each of two assemblies. Two pins, numbered 32 and 60, respectively, failed by a longitudinal split, 3/8-inch long, located about 1 inch above the bottom of the six-inch fuel column. The failed pins (described in Table 3-1) were located on opposite sides of the core. Several fuel pins also showed red-brown markings over the fuel column length. This effect occurred on both Zircaloy and stainless steel sheathed pins. It was assumed that the deposit was an iron-base crud which originated from ferrous components in WAPD-49 capsules or elsewhere in the ETR core. Except for a slight effect on reducing heat transfer to the coolant, it was concluded that the crud deposit had no great significance in the analysis of the fuel pin failures.

Subsequent analysis revealed the following:

1. Discrepancies in the flux-enrichment correlation used to calculate power levels led to pin operation at twice the expected heat rating. Sheath distensions were found which were compatible with the over-power conditions. These conditions were confirmed by metallographic observations of the irradiated fuel cross sections.

TABLE 3-1

FIRST INCIDENT, FAILED FUEL PIN DESIGN DATA

Pin No.	Enrichment (w/o U-235)	Fuel Density (% Theoretical)	Design Heat Rating (kw/ft)
32	13.5	94	18
60	20.0	90	24

Note: Both pins were sheathed in 0.065-inch thick Zircaloy-4.

2. Post-irradiation examination revealed no measurable pin bowing. Therefore, thermal bowing and subsequent hot spot formation did not contribute to the cause of the failures.
3. Calculations showed that hydraulic conditions were not responsible for pin failure.
4. Gas release measurements on pellets from the same fuel lot represented in the failed pins showed less than 0.04 cc (STP)/gm; thus, volatile fuel contaminants did not contribute to the failures.

It was concluded, therefore, that the failures were due to operation at twice the intended heat rating. This, in turn, was due to discrepancies in the flux-enrichment correlations which resulted in the calculation of excessively high requested neutron fluxes.

Corrections were made in the design calculations and revised neutron fluxes were requested. Remedial action consisted of changes in ETR reflector positions to achieve the lower fluxes indicated in Table 3-2.

Capsules A, B, and C (nominally 12 kw/ft), undamaged by the overpower conditions, were reinserted in ETR prior to Cycle 73. Capsules originally designed for operation at 18 kw/ft and 24 kw/ft actually operated at 36 kw/ft and 48 kw/ft, respectively, and were regarded as potentially damaged, requiring replacement. New capsules and pins were fabricated for subsequent irradiation.

3.2 Second Incident

Sodium and iodine activity were detected in the reactor coolant during irradiation Cycle 73. Assemblies A, B, and C were removed from test and were examined. The defects were isolated to Assembly C, where two pin holes were found in Pin 13 and two small cracks were found in Pin 15. These defects were all located at the bottom end-plug welds. The design parameters for these pins are given in Table 3-3. Following these findings, the status of the program was reviewed with the ETR staff. The results of this review were as follows:

TABLE 3-2

CORRECTED OPERATING CONDITIONS FOR FUEL SWELLING CAPSULES

Capsule (WAPD-49)	Enrichment (w/o U-235)	Initial Goal Heat Rating (kw/ft)	Requested Unperturbed Neutron Flux, nv
A	18.7	12.9	6.22×10^{13}
B	18.7	12.9	6.22×10^{13}
C	18.7	12.9	6.22×10^{13}
D	16.0	19.5	9.81×10^{13}
E	13.5	19.2	10.23×10^{13}
F	13.5	19.2	10.23×10^{13}
G	16.0	19.2	9.66×10^{13}
H	17.0	25.5	12.67×10^{13}
I	18.7	25.2	12.15×10^{13}
J	18.7	25.2	12.15×10^{13}
K	20.0	25.1	11.90×10^{13}
L	20.0	25.1	11.90×10^{13}

TABLE 3-3

SECOND INCIDENT, FAILED FUEL PIN DESIGN DATA

Pin No.	Fuel Enrichment (w/o U-235)	Fuel Density (% Theoretical)	Design Heat Rating (kw/ft)	Clad Wall Thickness and Material
13	18.7	97	12	0.065" Zircaloy-4
15	18.7	90	12	0.020" Type 304 Stainless Steel

1. It was agreed that Capsules A and B would not be re-inserted into the ETR due to the uncertainties in weld integrity posed by the failures in Assembly C;
2. Newly fabricated assemblies P, Q, R, S, T, U, V, W, and X were returned to WAPD site for re-examination of welds;
3. Agreement was reached on pin and capsule design modifications necessary to meet the requirements of the ETR staff for future irradiation, principally, the rigid attachment of both ends of each fuel pin to the capsules to prevent vibration and reduce the chance of developing weld cracks in-pile.

The original experimental design was revised and the unirradiated fuel pins were redistributed in different capsule groupings to make the best use of the available fuel and components while simultaneously satisfying the statistical requirements of the experiment. A new assembly numbering system was used.^[1] Appendix A presents the revised experimental parameters.

To ensure greater reliability, the capsule design was further modified (Figure B-4, Appendix B). The essential design changes were:

1. The use of locating bosses (items 1 and 2 of Figure B-4, Appendix B);
2. The provision of a circumferential metal restraining band to prevent possible fuel pin bowing and subsequent hot spot formation.

In addition to high pressure helium leak testing and dye-penetrant inspection of all pin welds, fuel pin integrity was evaluated by inspection of weld zone radiographs by the ETR staff as well as by WAPD personnel.

Capsules AA, AB, AE, AF, AG, AL, and AM were shipped to the ETR for irradiation testing and were inserted in Cycle 76. Capsules AC, AH, and AN were inserted in Cycle 77. Assemblies AD, AJ, and AP were inserted in Cycle 78. Figure 3-1 illustrates a revised program history of irradiations, where the numbers in parentheses refer to burnup (MWD/MTU).

3.3 Third Incident

Irradiation during Cycles 76 and 77 proceeded without incident. During irradiation Cycle 78, high iodine activity was detected in the ETR reactor coolant. The source of the activity was traced to assembly AF, Fuel Pin 104 (designated in Appendix A as parameter 21), in which a longitudinal split approximately

Figure 3-1. Revised Program History of Irradiations

1/2-inch long was found in the thick-walled Zircaloy clad (Figure 3-2). The location of the failure and clad dimensions are shown in Figure 3-3. The other assemblies which contained heavy-walled Zircaloy clad fuel pins, were also discharged from the reactor. These were assemblies AB, AD, AH, AJ, AL, AM, and AN.

Assembly AF contained four Zircaloy-clad fuel pins designed to operate at a nominal linear heat rating of 18 kw/ft. Pin details are given in Table 3-4.

TABLE 3-4

ASSEMBLY AF FUEL PIN DESIGN DATA

Fuel Pin No.	Parameter No. ^a	Fuel Density (% Theoretical)	Fuel Enrichment (w/o U-235)	Clad Wall (in.)
104 (failed)	21	94	13.4	0.065
105	22	90	13.4	0.065
106	23	96	13.4	0.065
107	24	90	13.4	0.024

Since the failure occurred in a thick-walled specimen, post-irradiation examination was restricted to Pins 104, 105, and 106. Examination of the fuel cross-sections showed a large central void surrounded by a ring of unusual porosity in Fuel Pin 104 and not in the others, which indicated that center melting had occurred only in Fuel Pin 104. The other pins operated at lower power ratings, as deduced from the fuel cross-sections.

An unpolished cross-section at the failure location of Fuel Pin 104 is shown in Figure 3-4. A composite of the assembly is shown in Figure 3-5. The

a. The parameter number is included for the purpose of allowing cross-referencing with complete pin details in Appendix A.

Figure 3-2. Failed Fuel Pin in Assembly AF

Figure 3-3. Location of Failure and Cladding Dimensions - Pin 104

Figure 3-4. Fuel Cross-Section in Region of Cladding Rupture - Pin 104

Figure 3-5. Cross-Section of Assembly AF,
2-5/16" From the Base of the Fuel Column

analysis of the failure considered three possible causes: 1) localized flow restriction due to bowing, thereby leading to high clad surface temperature, 2) fabrication error involving excessive fuel enrichment, and 3) neutron flux variations and inhomogeneities.

Bowing was eliminated as a cause of the failure since the redesign of the assembly to incorporate a restraining band precludes this possibility. Also calculations show that a hot spot sufficient to cause burnout could not develop where the band impinges upon the fuel pin, and the failure was located in a region away from the restraining band.

The pins in Assembly AF were designed to operate at 18 kw/ft with fuel of 13.4 w/o enrichment. Since fuel of higher enrichment was being irradiated as part of the same experimental program, the consequences of inadvertently using the most highly enriched fuel (19.96 w/o) were examined even though SNM accountability procedures clearly obviate this as a credible overpower cause. For the reported unperturbed neutron flux of 1×10^{14} nv the linear heat rating of Pin 104, assuming 19.96 w/o enrichment, would be 20.74 kw/ft. The linear heat rating required to produce incipient center melting in Pin 104 is 29.5 kw/ft. It was concluded that the observed center melting cannot be explained by fuel enrichments in excess of design. Similarly, design uncertainties could not lead to gross center melting. In the hazards calculations, allowance was made for increased capsule heat ratings from an uncertainty of 20 percent in the flux level, 5 percent uncertainty in assembly tolerance variations and 15 percent uncertainty in hot channel conditions. The heat ratings may therefore increase by a factor of 1.45. The designed initial power output of the hottest pin in this assembly was 19.7 kw/ft (Appendix A). The total design uncertainty in the linear heat rating would therefore yield a maximum anticipated value of 28.6 kw/ft. Thus, gross center melting was not anticipated for the worst tolerance combination. These results suggested large flux uncertainties in the ETR test holes.

To aid in the failure analysis, microstructures and cesium-137 activity of the control Assemblies M, N and O were reviewed.

It was concluded, from metallographic examination of the fuel cross sections in Assemblies AF (Figure 3-5) and N (Figure 3-6), from the cesium-137 radiochemical analyses of fuel pins, and from cross sections taken at equivalent

Figure 3-6. Cross-Section of Assembly WAPD 49-N, 2-1/2"
 From the Base of the Fuel Column

locations in the previously irradiated control Assemblies M, N, and O (Table 3-5), that a significant flux gradient exists across the ETR test holes. Figures 3-5 and 3-6 show the variance of microstructures due to the flux gradient. The cesium-137 activities show ratios of the maximum to minimum values of 1.66, 2.26, and 2.80 for Assemblies M, N, and O, respectively. The average ratio is 2.24, which corresponds to an approximate variation of ± 38 percent in linear heat rating about the nominal value.

In addition to the observed variation, the mean thermal neutron flux in the assembly was reported to be 20 percent in excess of the requested value of 1×10^{14} nv. This further aggravates the conditions imposed by the flux gradient. It was concluded that increased thermal neutron flux in Pin 104, due to local flux perturbations aggravated by the reported 20 percent higher flux, was the cause of gross fuel melting. The volume expansion accompanying fuel melting led to excessive cladding strain, culminating in cladding rupture.

As a result of this incident, it was requested that in future irradiations the assemblies be oriented in such a way that the thick-walled Zircaloy pins would be placed farthest from the center of the reactor. Thus, since the flux decreases in a radial direction away from the reactor core center, in the aluminum reflector region housing these experiments, the flux in the heavy walled pins would be reduced by about 30 percent from the nominal mean value (based on the Cs-137 variations in Assemblies M, N, and O, and on the variation in microstructures noted above). It was assumed that orientation of the assemblies in this manner would prevent further failures in thick-walled pins. This request was implemented in ETR Cycle 84. The use of flux wires was requested but was not implemented. The previously discharged assemblies were reinserted into the reactor and irradiation was continued.

3.4 Fourth Incident

During ETR Cycle 80, intermittent fission product activity was detected in the reactor coolant. The source of the activity was traced to Assembly AH.

TABLE 3-5

CS-137 ACTIVITIES IN ASSEMBLIES M, N, AND O

Assembly	Fuel Pin No.	Cs-137 Activity (10^{10} dpm/gm U)	Nominal Heat Rating (kw/ft)
M	72	1.2	24
	73	1.58	24
	74	2.01	24
	75	1.48	24
N	76	3.28	36
	77	1.93	36
	78	1.45	36
	79	2.18	36
O	80	2.97	48
	81	3.72	48
	82	1.48	48
	83	1.33	48

As no sodium activity was detected, it was erroneously inferred that the failed fuel pin was Pin 114, a thick-walled Zircaloy clad fuel pin, and that failure was again due to the flux inhomogeneities since the capsules had not been oriented as described in 3.3 above. The reactor was shut down and all assemblies containing thick-walled Zircaloy fuel pins were discharged, namely AB, AD, AJ, AH, AL, AM and AN. Assemblies AA, AC, AE, AG, and AP were left in the reactor.

The defect was located in Assembly AH which was designed to operate at a nominal linear heat rating of 18 kw/ft. It was irradiated for three reactor cycles to an estimated burnup of 11,250 MWD/MTU. The assembly consisted of the fuel pins listed in Table 3-6.

TABLE 3-6
ASSEMBLY AH, FUEL PIN DESIGN DATA

Fuel Pin No.	Design ^a Parameter No.	Fuel Density (% T.D.)	Fuel Enrichment (w/o U-235)	Clad Wall Thickness and Material	
				Inner	Outer
112	29	96	15.96	0.005" 304 Type Stainless Steel	0.020" 304 Type Stainless Steel
113(failed)	30	96	15.96	0.015" 304 Type Stainless Steel	0.020" 304 Type Stainless Steel
114	31	96	15.96	0.065" Zircaloy-4	---
115	32	90	15.96	0.024" Zircaloy-4	0.020" 304 Type Stainless Steel

a. The parameter number is included for the purpose of allowing cross-referencing with complete pin details in Appendix A.

Preliminary visual examination at the ETR site revealed no gross defects.

The assembly was then transferred to the Westinghouse Waltz Mill site for detailed post-irradiation examination.

Since there was no obvious point of failure, the fuel pins were removed from the assembly. Fuel Pins 112, 114, and 115 were leak tested, but no pin-holes or weld defects were found. An elliptically-shaped hole 0.079 inches long by 0.058 inches wide was observed visually in the wall of Fuel Pin 113. This hole was located 4-11/16 inches from the bottom of the pin facing the assembly central support rod. There was a deposit on the immediately adjacent sheath surface of Pin 112 in the same area. Metallographic examination revealed a fuel structure consistent with a fuel surface temperature of about 1700°C, with extensive center-melting of the fuel indicative of a linear heat rating of about 30 kw/ft. The fuel burnup, determined from the radiochemically analyzed Zr-95 content, was 8,000 ±800 MWD/MTU. The linear heat rating of Pin 113, calculated from the radiochemistry data to an accuracy of ±10 percent, was 28 kw/ft. The fuel pin therefore operated at power levels significantly greater than the nominal design value of 18 kw/ft. A metallographic cross section of the fuel pin at the point of cladding failure is shown in Figure 3-7. Resolidified fuel and clad are located in the NaK annulus in a region extending over about one-half the circumference of the inner clad.

Possible mechanisms of the fuel pin failure were examined in detail. It was concluded that the pin failed because the NaK was omitted during fabrication. This resulted in excessive fuel and cladding temperatures, causing melting of the inner clad and center melting of the fuel. Molten material came in contact with the outer clad, reacted, and formed a hole. This conclusion was based on the following observations:

1. No sodium was detected in the reactor coolant at the time of failure. In previous failures with doubly-encapsulated Pins 13 and 15, sodium activity had been observed in the coolant;
2. The internal transfer of NaK from the annulus to the fuel column, through a postulated defect in the inner clad, was rejected because of insufficient volume to accommodate the NaK in this region;
3. Fuel and clad were found in the NaK annulus.

NEG. #3105

6.6X

Figure 3-7. Cross-Section at Location of Failure - Pin 113

4. Calculations show that the observed metallographic structure and the linear heat rating are consistent with operation of this fuel pin without NaK.
5. Re-examination of the fabrication record showed no entry for NaK on the log sheet for Pin 113. The omission was originally attributed to an oversight in record keeping. However, the listed observations indicate that the NaK loading step was actually omitted.

The fabrication records showed that NaK was inserted into all other doubly encapsulated fuel pins: consequently, repetition of this type of failure was not expected.

Approval was obtained for re-insertion of the unfailed assemblies at the mid-cycle shutdown, reactor Cycle 83. Assemblies AD and AN were re-inserted, but Assemblies AB, AJ, AL and AM would not fit into the facility holes until the lower support basket was straightened. Consequently, these assemblies were re-inserted in reactor Cycle 84. Assemblies AI, AK, AO and AQ were inserted in reactor Cycle 84.

3.5 Fifth Incident

All available assemblies were irradiated during reactor Cycle 84. No incidents were observed during this cycle. However, high activity was observed in the reactor coolant during the startup of Cycle 85. This activity was traced to Assemblies AK and AQ. All assemblies were discharged. Difficulties were experienced in discharging Assembly AG. Assemblies AA, AB, AC, AD, AE, AG, AI, AJ, AL, AM, AN, AO and AP were examined in the ETR canal and were stored pending disposition of the experiment.

The restraining strap on Assembly AC was slightly displaced and the bottom support basket was missing from Assembly AG. The fuel pin cluster and the upper support basket of Assembly AG were undamaged. The pin attaching the lower basket to the fuel pin cluster was sheared during discharge as the lower basket jammed in the experimental hole. The lower support pieces in Assemblies AB, AL and AM were partially pushed into the lower support baskets. The damage to these three assemblies, as well as Assembly AG, was considered relatively minor. All other assemblies were in good condition.

Hot cell examination of Assemblies AK and AQ revealed longitudinal cladding rupture in Pins 125 (Assembly AK) and 150 (Assembly AQ).

The rupture in Pin 125 was 1/4-inch long and was located 3-1/16 inches above the pin bottom. The rupture in Pin 150 was 5/8-inch long and was located 2 inches above the pin bottom. The location of the ruptures is shown in Figure 3-8. The general appearance of the pins is shown in Figure 3-9.

Assemblies AR and AQ were sectioned through the mid-point of the failure and at a location 8 inches from the top end of the pin. These locations are shown in Figure 3-8 as short horizontal dashed lines. The as-sectioned assemblies are shown at the locations of failure in Figures 3-10 and 3-11. In both cases, the view is toward the bottom of the pins. It can be noted in the cross-section of Assembly AK, Figure 3-10, that both the inner and outer clad of Pin 125 are ruptured, and the inner clad of Pin 124 has also failed. The latter was apparently unnoticed at the time of sectioning. Subsequent examination at the WAPD Waltz Mill Facility clearly showed the rupture. A center-melted fuel structure and evidence of NaK reaction with the fuel are revealed in Figures 3-12 and 3-13, demonstrating that the rupture was produced in-pile and not as a result of the post-irradiation sectioning procedure.

In addition, a cross-section taken 8 inches from the top of each pin, away from the external failure, showed a small portion of the inner clad of Pin 124 was ruptured and although the outer clad of Pin 125 was intact, a portion of the inner clad was ruptured, showing that the fuel expansion had occurred over an extensive portion of this pin.

The location and orientation of Assemblies AK and AQ in the ETR core are shown in Figure 3-14. Design details of the pins in these assemblies are given in Tables 3-7 and 3-8.

Before Assemblies AK and AQ were shipped to the Westinghouse Waltz Mill site, possible causes of the failures were discussed with the ETR staff. The preliminary consensus at this time was that the thermal-hydraulic conditions, due to the small clearance between the assemblies and the I.D. of the experimental facility wall, are not known with certainty and a hot-spot could conceivably develop, resulting in pin failure.

Figure 3-8. Location of Rupture Areas, Pins 125 and 150.

Figure 3-9. Cross Section of Failed Fuel Pin
125 (Left), and 150 (Right)

Figure 3-10. Assembly AK - As-Sectioned

Figures 3-11. Assembly AQ - As-Sectioned

↑
CENTER OF
ETR CORE

18 KW/FT DESIGN RATING

13.5X

MACRO #3304

Figure 3-12. As-Polished Surface of Pin 124

Figure 3-13. Autoradiograph of Pin 124

Figure 3-14. ETR Core Cross-Sectional Diagram, Showing Location and Orientation of Assemblies AK and AQ.

TABLE 3-7

ASSEMBLY AK FUEL PIN DESIGN DATA

Pin No.	Fuel Density (% Theoretical)	Fuel Enrichment (w/o U-235)	Clad Wall Thickness and Material	
			Inner	Outer
124(failed)	96	15.96	0.005" 304 Type Stainless Steel	0.020" 304 Type Stainless Steel
125(failed)	96	15.96	0.024" Zircaloy-4	0.020" 304 Type Stainless Steel
126	96	15.96	0.065" Zircaloy-4	---
127	96	15.96	0.024" Zircaloy-4	0.020" 304 Type Stainless Steel

Notes: All pins were designed for operation at 18 kw/ft (nominal)

TABLE 3-8

ASSEMBLY AQ FUEL PIN DESIGN DATA

Pin No.	Parameter No.	Fuel Density (% Theoretical)	Fuel Enrichment (w/o U-235)	Clad Wall Thickness and Material	
				Inner	Outer
148	65 ^a	94	19.96	0.024" Zircaloy-4	0.020" 304 Type Stainless Steel
149	66	96	19.96	0.024" Zircaloy-4	0.020" 304 Type Stainless Steel
150(failed)	67	96	19.96	0.065" Zircaloy-4	---
151	68	94	19.96	0.065" Zircaloy-4	---

Note: All pins were designed for operation at 24 kw/ft (nominal)

a. The parameter number is included for the purpose of allowing cross-referencing with complete pin details in Appendix A.

The original fuel enrichment calculations^[1] were rechecked and found to be without error. However, since selection of the fuel enrichment is dependent on flux perturbation factors, the validity of the design equation

$$f = \frac{(0.75)(0.8)}{1 + (1.05)_{\text{gm}} U^{235}/\text{in.}},$$

where

f = flux perturbation factor, was reconsidered and found to be applicable.

Detailed microstructural examination and burnup analyses showed that the cause of failure was overpower, resulting in center melting of all eight fuel pins in Assemblies AK and AQ, with consequent volume expansion and cladding rupture.

The microstructural features which have been correlated with center melting are illustrated in Figures 3-13 to 3-33, inclusive. The central voids, together with the central ring of unusual porosity are indications of center-melting of the fuel^[2] (Figures 3-13, 3-15, 3-16, 3-18, 3-20, 3-23, 3-27, 3-30, 3-31). Either as-polished or as-ground surfaces are shown, depending on which method of surface preparation best revealed the structure. (Figures 3-15 and 3-16 for Pin 125, show the differences between these types of surface preparation.)

Beta-gamma autoradiographs of seven pins were prepared, and are shown as Figures 3-14, 3-17, 3-19, 3-21, 3-24, 3-28 and 3-32. These also show evidence of center melting. The ring structure of each figure, indicative of a low-activity band (the white rings in the positive prints of the autoradiographs), has been related to the extent of melting.^[2,3] Similarly, a high activity, dark band, rich in fission products and metallic particles, provides additional evidence of abnormally high heat ratings. These features are best seen in the autoradiographs of Pin 149, (Figure 3-28), and the photomicrographs of Pin 127 (Figure 3-22), Pin 148 (Figure 3-25), Pin 149 (Figure 3-29), and Pin 151 (Figure 3-33).

These particles, observed under the microscope as bright, etch-resistant spheroids, are assumed to be either uranium particles formed when the fuel becomes substoichiometric at high temperatures^[4,8] or a fission product alloy.^[7]

CENTER OF
ETR CORE

18KW/FT DESIGN RATING

13.7X

MACRO #3274

Figure 3-15. As-Ground Surface of Pin 125

CENTER OF
ETR CORE

18 KW/FT DESIGN RATING

13.5X

MACRO #3307

Figure 3-16. As-Polished Surface of Pin 125

AUTORADIOGRAPH
PIN #125

Figure 3-17. Autoradiograph of Pin 125

CENTER OF
ETR CORE

18 KW/FT DESIGN RATING

13.5X

MACRO #3301

Figure 3-18. As-Polished Surface of Pin 126

AUTORADIOGRAPH
PIN #126

Figure 3-19. Autoradiograph of Pin 126

CENTER OF
ETR CORE

18 KW/FT DESIGN RATING

12.7X

MACRO #3302

Figure 3-20. As-Polished Surface of Pin 127

AUTORADIOGRAPH
PIN #127

Figure 3-21. Autoradiograph of Pin 127

Figure 3-22. Fuel at R/A = 0.5, Pin 127

24 KW/FT DESIGN RATING

13.7X

MACRO #3277

Figure 3-23. Photomicrograph of As-Ground Surface of Pin 148

Figure 3-24. Autoradiograph of Pin 148

NEG. #582C1

250X

Figure 3-25. Cross Section of Fuel Near Center of Pin - Pin 148

NEG. #582D₂

100X

NEG. #582E₂

180° POSITION

NEG. #582F₂

100X

NEG. #582G₂

270° POSITION

Figure 3-26. Photomicrograph of Fuel Pin 148

CENTER OF
ETR CORE

24 KW/FT DESIGN RATING

13.5X

MACRO #3305

Figure 3-27. As-Polished Surface of Pin 149

AUTORADIOGRAPH
PIN #149

Figure 3-28. Autoradiograph of Pin 149

↑
CENTER
OF PIN

NEG. #583C1

250X

Figure 3-29. Photomicrograph of Fuel at $R/A = 0.5$ - Pin 149

CENTER OF
ETR CORE

24 KW/FT DESIGN RATING

14X

MACRO #3279

Figure 3-30. As-Ground Surface of Pin 150

CENTER OF
ETR CORE

24 KW/FT DESIGN RATING

14X

MACRO #3280

Figure 3-31. As-Ground Surface of Pin 151

AUTORADIOGRAPH
PIN #151

Figure 3-32. Autoradiograph of Pin 151

Figure 3-33. Photomicrograph of Fuel at R/A = 0.25 - Pin 151

Subgrain structures, similar to those previously observed in molten UO_2 ^[3] were also observed in the region of high activity particles. These are best illustrated in Figure 3-22, Pin 127.

The appearance of the microstructures and autoradiographs, with central void formation, areas of unusual porosity, radiation, activity rings, high activity particles, and subgrains, appear to some extent in each pin and have been correlated with UO_2 fuel center-melting.^[2,3,5,6] These observations provide conclusive evidence of center melting in each of the pins in Assemblies AK and AQ.

Fuel burnup samples were taken from the middle region of the fuel column in each pin except Pin 150, where the sample was taken in the failed region. Calculations of the burnup and heat ratings were made from the Zr-95 count. These are shown in Table 3-9. These data are time-averaged over the cycle and do not show the expected variation in reactor flux from the beginning to the end of a cycle. These data, especially from Assembly AK, show the variance of burnup and power output within the assemblies. In Pins 127 and 148 the power output data are consistent with center melting. In Pin 125, the power output is consistent with center melting when the upper range of accuracy is considered.

The burnup data for Pins 126 and 150, where center melting was observed, do not support center melting. However, as noted above, the data represent the power output averaged over the irradiation cycle and do not reveal possible short-time power variations within a cycle sufficient to cause center melting. These data also support the observations made previously (Section 3.3) of flux inhomogeneities in the ETR.

The microstructures also show a skew in flux circumferentially with respect to the core center, e.g., in Figure 3-18 (Pin 126), and Figure 3-23 (Pin 148) the hottest point in the fuel, as evidenced by the central void, is off-center at approximately a right angle to the core-center direction. (The arrows on the figures designating the orientation of the pins with respect to the core center were determined at the ETR site by comparing the assembly cross-sections Figures 3-10 and 3-11, with the in-pile orientation markings on the top of the capsules.) Figure 3-26 further demonstrates the

TABLE 3-9

BURNUP AND HEAT RATING DATA, ASSEMBLIES AK AND AQ

Pin No.	Burnup (MWD/MTU)	Heat Rating (kw/ft)	
		Actual	Design
AK-125 (failed)	3530 $\pm$ 350	26.6 $\pm$ 2.7	18
AK-126	2580 $\pm$ 260	19.4 $\pm$ 1.9	18
AK-127	3800 $\pm$ 380	28.6 $\pm$ 2.9	18
AQ-148	4290 $\pm$ 430	30.8 $\pm$ 3.1	24
AQ-150 (failed)	3060 $\pm$ 310	24.6 $\pm$ 2.5	24

asymmetry of flux within Pin 148. In this figure, showing the edge of the pellet at two angular positions in a direction clockwise from the arrow shown in Figure 3-23, it can be seen that the distance from the pellet edge of the interface between the equiaxed grains and the as-sintered structure, and the equiaxed columnar grains, representing fuel temperatures of 1400°C and 1800°C, respectively, varies significantly, showing the skew of the flux pattern within this pin. The other pins showed similar structures to varying degrees.

It was concluded that uncertainties in ETR conditions, coupled with the stringent flux requirements of this experiment, were the cause of the failures in Pins 125, 126, and 150. The remaining assemblies, AA, AB, AC, AD, AE, AG, AI, AJ, AL, AM, AN, AO, and AP were not further irradiated.

4. SUMMARY AND CONCLUSIONS

The experimental difficulties and results have been reviewed. As a result, the following conclusions have been reached:

1. Considerable value has been lost as a result of the failures. It is probable that the remaining assemblies have been compromised by the conditions responsible for the most recent failures.
2. At the time of the most recent failure incident, all experimental assemblies had been irradiated. Since most of the pins were doubly encapsulated, it was impractical to disassemble them for examination and assessment of the individual fuel pins.
3. Further modifications in experimental design or ETR test operating conditions would not unequivocally guarantee continued operation without spurious pin failure.

As a consequence of these conclusions, the irradiation was not continued.

5. REFERENCES

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2. R. F. Boyle, et al., "Autoradiography and Burr Drilling Techniques for Fission Product Migration Studies of Irradiated UO_2 ," GEAP-4452, June 1964.
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4. A. E. Martin and R. K. Edwards, "The Uranium-Uranium Dioxide Phase Diagram at High Temperatures," J. Phys. Chem., 69, 1788 (1965).
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APPENDIX A
DESIGN PARAMETERS

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin):	12	Film ΔT ($^{\circ}F$):	51
Required Unperturbed Thermal Neutron Flux (Av):	$6.078 \cdot 10^{13}$	Mean Bulk Water Temperature ($^{\circ}F$):	123
Fuel Enrichment (w/o U235 in Total U):	18.64	Fuel Length (in):	5.50
Water-Outer Sheath Surface Heat Flux (BTU/hr ft ²):	332,300	Fuel Diameter (in):	0.340 ± 0.0005
Film Coefficient (BTU/hr ft ² $^{\circ}F$):	6,456	Fuel-Sheath Diametral Gap (in):	0.002 ± 0.0002

B. SPECIFIC FUEL PINS

Parameter		1	2	3	4
Goal Burnup (MWD/MTU)		40,000	40,000	40,000	40,000
ETR Cycles (inclusive)		75-87	75-87	75-87	75-87
Initial Kw/ft		12.47	12.18	12.18	12.47
Terminal Kw/ft		11.42	11.16	11.16	11.42
Fuel Density (% Theoretical)		96.2	94	94	96.2
g U235/in. Fuel		2.63	2.52	2.52	2.63
Inner Sheath		S/S 0.005" Wall x 0.343" I.D.	Zirc. 0.024" Wall x 0.343" I.D.	S/S 0.005" Wall x 0.343" I.D.	Zirc. 0.024" Wall x 0.343" I.D.
Components (g)	U235	14.46	13.85	13.85	14.46
	U238	68.5	65.7	65.7	68.5
	304 Stainless Steel	167	133.5	167	68
	Zircaloy 4	--	50.4	--	122.5
	NaK (56 w/oK)	6.50	3.42	6.50	--
Temperature Distribution ($^{\circ}F$)	Outer Sheath O.D.	174	174	174	174
	Outer Sheath I.D.	243	243	243	243
	Inner Sheath O.D.	328	284	328	284
	Inner Sheath I.D.	350	388	350	388
	Fuel-Sheath ΔT	184	184	184	184
	Fuel Surface	534	572	534	572
	Fuel Center	1590	1642	1590	1642
Outer Sheath		S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin):	12	Film ΔT ($^{\circ}F$):	51
Required Unperturbed Thermal Neutron Flux (A_v):	$6.078 \cdot 10^{13}$	Mean Bulk Water Temperature ($^{\circ}F$):	123
Fuel Enrichment (w/o U ²³⁵ in Total U):	18.64	Fuel Length (in):	5.50
Water-Outer Sheath Surface Heat Flux (BTU/hr ft ²):	332,300	Fuel Diameter (in):	0.34C + 0.0005
Film Coefficient (BTU/hr ft ² $^{\circ}F$):	6,456	Fuel-Sheath Diametral Gap (in):	0.002 + 0.0002

B. SPECIFIC FUEL PINS

Parameter		5	6	7	8
Goal Burnup (MWD/MTU)		40,000	40,000	40,000	40,000
ETR Cycles (inclusive)		75-87	75-87	75-87	75-87
Initial Kw/ft		12.47	12.47	12.18	12.18
Terminal Kw/ft		11.42	11.42	11.16	11.16
Fuel Density (% Theoretical)		96.2	96.2	94	94
g U ²³⁵ /in. Fuel		2.63	2.63	2.52	2.52
Inner Sheath		----	*S/S 0.015" Wall x 0.343" I.D.	*S/S 0.015" Wall x 0.343" I.D.	*S/S 0.015" Wall x 0.343" I.D.
Components (g)	U ²³⁵	14.46	14.46	13.85	13.85
	U ²³⁸	68.50	68.50	65.7	65.7
	304 Stainless Steel	68	180	180	180
	Zircaloy 4	122.5	---	---	---
	NaK (56 w/oK)	---	4.90	4.90	4.90
Temperature Distribution ($^{\circ}F$)	Outer Sheath O.D.	174	174	174	174
	Outer Sheath I.D.	441	243	243	243
	Inner Sheath O.D.	---	305	305	305
	Inner Sheath I.D.	---	361	361	361
	Fuel-Sheath ΔT	184	184	184	184
	Fuel Surface	625	545	545	545
	Fuel Center	1770	1610	1610	1610
Outer Sheath		Zirc. 0.065" Wall x 0.343" I.D.	S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	Zirc. 0.065" Wall x 0.343" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin): 12
 Required Unperturbed Thermal Neutron Flux (Av): 6.002×10^3
 Fuel Enrichment (w/o U235 in Total U): 18.64
 Water-Outer Sheath Surface Heat Flux (BTU/hr ft²): 332,300
 Film Coefficient (BTU/hr ft² °F): 6,456

Film ΔT (°F): 51
 Mean Bulk Water Temperature (°F): 123
 Fuel Length (in): 5.50
 Fuel Diameter (in): 0.340 ± 0.005
 Fuel-Sheath Diametral Gap (in): 0.002 ± 0.002

B. SPECIFIC FUEL PINS

Parameter		9	10	11	12
Goal Burnup (MWD/MTU)		28,000	28,000	28,000	28,000
ETR Cycles (inclusive)		79-87	79-87	79-87	79-87
Initial Kw/ft		12.31	12.31	12.03	12.03
Terminal Kw/ft		11.65	11.65	11.38	11.38
Fuel Density (% Theoretical)		96.2	96.2	94	94
g U235/in. Fuel		2.63	2.63	2.52	2.52
Inner Sheath		S/S 0.005" Wall x 0.343" I.D.	* S/S 0.015" Wall x 0.343" I.D.	* S/S 0.015" Wall x 0.343" I.D.	S/S 0.005" Wall x 0.343" I.D.
Components (g)	U235	14.46	14.46	13.85	13.85
	U238	68.5	68.50	65.7	65.7
	304 Stainless Steel	167	180	180	167
	Zircaloy 4	---	---	---	---
	NaK (56 w/oK)	6.50	4.90	4.90	6.50
Temperature Distribution (°F)	Outer Sheath O.D.	174	174	174	174
	Outer Sheath I.D.	243	243	243	243
	Inner Sheath O.D.	328	305	305	328
	Inner Sheath I.D.	350	361	361	350
	Fuel-Sheath ΔT	184	184	184	184
	Fuel Surface	534	545	545	534
	Fuel Center	1590	1610	1610	1590
Outer Sheath		S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin): 12
 Required Unperturbed Thermal Neutron Flux (Av): $6.002 \cdot 10^{13}$
 Fuel Enrichment (w/o U235 in Total U): 18.64
 Water-Outer Sheath Surface Heat Flux (BTU/hr ft²): 332,300
 Film Coefficient (BTU/hr ft² °F): 6,456

Film ΔT (°F): 51
 Mean Bulk Water Temperature (°F): 123
 Fuel Length (in): 3.50
 Fuel Diameter (in): 0.340 ± 0.0005
 Fuel-Sheath Diametral Gap (in): 0.004 ± 0.0002

B. SPECIFIC FUEL PINS

Parameter		13	14	15	16
Goal Burnup (MWD/MTU)		28,000	28,000	28,000	28,000
ETR Cycles (inclusive)		79-87	79-87	79-87	79-87
Initial Kw/ft		12.31	12.03	11.52	12.03
Terminal Kw/ft		11.65	11.38	10.90	11.38
Fuel Density (% Theoretical)		96.2	94	90	94
g U235/in. Fuel		2.63	2.52	2.41	2.41
Inner Sheath		-----	----	* S/S 0.015" Wall x 0.343" I.D.	*S/S 0.015" Wall x 0.343" I.D.
Components (g)	U235	14.46	13.85	13.25	13.25
	U238	68.5	65.7	62.8	62.8
	304 Stainless Steel	68	68	180	180
	Zircaloy 4	122.5	122.5	---	---
	NaK (56 w/oK)	---	---	4.90	4.90
Temperature Distribution (°F)	Outer Sheath C.D.	174	174	174	174
	Outer Sheath I.D.	441	441	243	243
	Inner Sheath C.D.	---	---	305	305
	Inner Sheath I.D.	---	---	361	361
	Fuel-Sheath ΔT	184	184	184	184
	Fuel Surface	625	625	545	545
	Fuel Center	1770	1770	1610	1610
Outer Sheath		Zirc. 0.065" Wall x 0.343" I.D.	Zirc. 0.065" Wall x 0.343" I.D.	S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin): 18
 Required Unperturbed Thermal Neutron Flux (Av): $9.796 \cdot 10^{13}$
 Fuel Enrichment (w/o U235 in Total U): 15.96
 Water-Outer Sheath Surface Heat Flux (BTU/hr ft²): 498,500
 Film Coefficient (BTU/hr ft² °F): 6,456

Film ΔT (°F): 77
 Mean Bulk Water Temperature (°F): 124
 Fuel Length (in): 5.50
 Fuel Diameter (in): 0.340 ± 0.0005
 Fuel-Sheath Diametral Gap (in): 0.004 ± 0.0002

B. SPECIFIC FUEL PINS

Parameter		17	18	19	20
Goal Burnup (MWD/MTU)		50,000	50,000	50,000	50,000
ETR Cycles (inclusive)		76-90	76-90	76-90	76-90
Initial Kw/ft		19.22	18.78	17.98	17.98
Terminal Kw/ft		16.37	16.0	15.32	15.32
Fuel Density (% Theoretical)		96.2	94	90	90
g U235/in. Fuel		2.25	2.15	2.06	2.06
Inner Sheath		S/S 0.005" Wall x 0.343" I.D.	S/S 0.005" Wall x 0.343" I.D.	S/S 0.005" Wall x 0.343" I.D.	Zirc. 0.024" Wall x 0.343" I.D.
Components (g)	U235	12.37	11.86	11.30	11.30
	U238	70.70	67.82	65	65
	304 Stainless Steel	167	167	167	133.5
	Zircaloy 4	---	---	---	50.4
	NaK (56 w/oK)	6.50	6.50	6.50	3.42
Temperature Distribution (°F)	Outer Sheath O.D.	201	201	201	201
	Outer Sheath I.D.	293	293	293	293
	Inner Sheath O.D.	420	420	420	355
	Inner Sheath I.D.	447	447	447	508
	Fuel-Sheath ΔT	276	276	276	276
	Fuel Surface	723	723	723	784
	Fuel Center	2960	2960	2960	3340
Outer Sheath		S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin): 18
 Required Unperturbed Thermal Neutron Flux (Av): $1.062 \cdot 10^{14}$
 Fuel Enrichment (w/o U²³⁵ in Total U): 13.4
 Water-Outer Sheath Surface Heat Flux (BTU/hr ft²): 498,500
 Film Coefficient (BTU/hr ft² °F): 6,456

Film ΔT (°F): 77
 Mean Bulk Water Temperature (°F): 124
 Fuel Length (in): 5.50
 Fuel Diameter (in): 0.340 ± 0.0005
 Fuel-Sheath Diametral Gap (in): 0.004 ± 0.0002

B. SPECIFIC FUEL PINS

Parameter		21	22	23	24
Goal Burnup (MWD/MTU)		50,000	50,000	50,000	50,000
ETR Cycles (inclusive)		76-90	76-90	76-90	76-90
Initial Kw/ft		19.25	18.43	19.7	18.43
Terminal Kw/ft		15.22	14.58	15.58	14.58
Fuel Density (% Theoretical)		94	90	96.2	90
g U ²³⁵ /in. Fuel		1.82	1.74	1.89	1.74
Inner Sheath		----	----	----	Zirc. 0.024" Wall x 0.343" I.D.
Components (g)	U ²³⁵	10.0	9.56	10.4	9.56
	U ²³⁸	70.8	67.7	73.8	67.7
	304 Stainless Steel	68	68	68	133.5
	Zircaloy 4	122.5	122.5	122.5	50.4
	NaK (56 w/oK)	----	----	----	3.42
Temperature Distribution (°F)	Outer Sheath O.D.	201	201	201	201
	Outer Sheath I.D.	601	601	601	293
	Inner Sheath O.D.	---	---	---	355
	Inner Sheath I.D.	---	---	---	508
	Fuel-Sheath ΔT	276	276	276	276
	Fuel Surface	877	877	877	784
	Fuel Center	3410	3410	3410	3340
Outer Sheath		Zirc. 0.065" Wall x 0.343" I.D.	Zirc. 0.065" Wall x 0.343" I.D.	Zirc. 0.065" Wall x 0.343" I.D.	S/S 0.020" Wall x 0.431" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin): 18
 Required Unperturbed Thermal Neutron Flux (Av): $8.993 \cdot 10^{13}$
 Fuel Enrichment (w/o U235 in Total U): 19.96
 Water-Outer Sheath Surface Heat Flux (BTU/hr ft²): 498,500
 Film Coefficient (BTU/hr ft² °F): 6,456

Film ΔT (°F): 77
 Mean Bulk Water Temperature (°F): 124
 Fuel Length (in): 5.50
 Fuel Diameter (in): 0.340 ± 0.0005
 Fuel-Sheath Diametral Gap (in): 0.004 ± 0.0002

B. SPECIFIC FUEL PINS

Parameter		25	26	27	28
Goal Burnup (MWD/MTU)		50,000	50,000	50,000	50,000
ETR Cycles (inclusive)		76-90	76-90	76-90	76-90
Initial Kw/ft		17.58	18.79	18.36	17.58
Terminal Kw/ft		15.89	16.98	16.59	15.89
Fuel Density (% Theoretical)		90	96.2	94	90
g U ²³⁵ /in. Fuel		2.58	2.81	2.69	2.58
Inner Sheath		* S/S 0.015" Wall x 0.343" I.D.	* S/S 0.015" Wall x 0.343" I.D.	* S/S 0.015" Wall x 0.343" I.D.	* S/S 0.015" Wall x 0.343" I.D.
Components (g)	U235	14.20	15.45	14.80	14.20
	U238	62.0	67.4	64.60	62.0
	304 Stainless Steel	180	180	180	180
	Zircaloy 4	---	---	---	---
	NaK (56 w/oK)	4.90	4.90	4.90	4.90
Temperature Distribution (°F)	Outer Sheath G.D.	201	201	201	201
	Outer Sheath I.D.	293	293	293	293
	Inner Sheath G.D.	385	385	385	385
	Inner Sheath I.D.	466	466	466	466
	Fuel-Sheath ΔT	276	276	276	276
	Fuel Surface	742	742	742	742
	Fuel Center	3000	3000	3000	3000
	Outer Sheath	S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin): 18
Required Unperturbed Thermal Neutron Flux (Av): $9.517 \cdot 10^{13}$
Fuel Enrichment (w/c U235 in Total U): 15.96
Water-Outer Sheath Surface Heat Flux (BTU/hr ft²): 498,500
Film Coefficient (BTU/hr ft² °F): 6,456

Film ΔT (°F): 77
Mean Bulk Water Temperature (°F): 124
Fuel Length (in): 5.50
Fuel Diameter (in): 0.340 ± 0.0005
Fuel-Sheath Diametral Gap (in): 0.004 ± 0.0002

B. SPECIFIC FUEL PINS

Parameter		29	30	31	32
Goal Burnup (MWD/MTU)		34,000	34,000	34,000	34,000
ETR Cycles (inclusive)		79-91	79-91	79-91	79-91
Initial Kw/ft		18.67	18.67	18.67	17.47
Terminal Kw/ft		17.18	17.18	17.18	16.07
Fuel Density (% Theoretical)		96.2	96.2	96.2	90
g U ²³⁵ /in. Fuel		2.25	2.25	2.25	2.06
Inner Sheath		S/S 0.005" Wall x 0.343" I.D.	S/S 0.015" Wall x 0.343" I.D.	-----	Zirc. 0.024" Wall x 0.343" I.D.
Components (g)	U ²³⁵	12.37	12.37	12.37	11.30
	U ²³⁸	70.7	70.7	70.7	65.0
	304 Stainless Steel	167	180	68	133.5
	Zircaloy 4	---	---	122.5	50.4
	NaK (56 w/oK)	6.50	4.90	---	3.42
Temperature Distribution (°F)	Outer Sheath O.D.	201	201	201	201
	Outer Sheath I.D.	293	293	601	293
	Inner Sheath O.D.	420	385	---	355
	Inner Sheath I.D.	447	466	---	508
	Fuel-Sheath ΔT	276	276	276	276
	Fuel Surface	723	742	877	784
	Fuel Center	2960	3000	3410	3340
Outer Sheath		S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	Zirc. 0.065" Wall x 0.343" I.D.	S/S 0.020" Wall x 0.431" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin):	18	Film ΔT ($^{\circ}\text{F}$):	77
Required Unperturbed Thermal Neutron Flux (Av):	$9.053 \cdot 10^{13}$	Mean Bulk Water Temperature ($^{\circ}\text{F}$):	124
Fuel Enrichment (w/o U235 in Total U):	18.64	Fuel Length (in):	5.50
Water-Outer Sheath Surface Heat Flux (BTU/hr ft ²):	498,500	Fuel Diameter (in):	0.340 ± 0.0005
Film Coefficient (BTU/hr ft ² $^{\circ}\text{F}$):	6,456	Fuel-Sheath Diametral Gap (in):	0.004 ± 0.0002

B. SPECIFIC FUEL PINS

Parameter		33	34	35	36
Goal Burnup (MWD/MTU)		34,000	34,000	34,000	34,000
ETR Cycles (inclusive)		79-91	79-91	79-91	79-91
Initial Kw/ft		18.14	18.14	18.14	17.37
Terminal Kw/ft		16.90	16.90	16.90	16.19
Fuel Density (% Theoretical)		94	94	94	90
g U235/in. Fuel		2.52	2.52	2.52	2.41
Inner Sheath		S/S 0.005" Wall x 0.343" I.D.	* S/S 0.015" Wall x 0.343" I.D.	-----	Zirc. 0.024" Wall x 0.343" I.D.
Components (g)	U235	13.85	13.85	13.85	13.25
	U238	65.7	65.7	65.7	62.8
	304 Stainless Steel	167	180	68	133.5
	Zircaloy 4	---	---	122.5	50.4
	NaK (56 w/oK)	6.50	4.90	----	3.42
Temperature Distribution ($^{\circ}\text{F}$)	Outer Sheath O.D.	201	201	201	201
	Outer Sheath I.D.	293	293	601	293
	Inner Sheath O.D.	420	385	---	355
	Inner Sheath I.D.	447	466	---	508
	Fuel-Sheath ΔT	276	276	276	276
	Fuel Surface	723	742	877	784
	Fuel Center	2960	3000	3410	3340
	Outer Sheath	S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	Zirc. 0.065" Wall x 0.343" I.D.	S/S 0.020" Wall x 0.431" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin):	18	Film ΔT ($^{\circ}F$):	77
Required Unperturbed Thermal Neutron Flux (Av):	$9.965 \cdot 10^{13}$	Mean Bulk Water Temperature ($^{\circ}F$):	124
Fuel Enrichment (w/o U ²³⁵ in Total U):	13.4	Fuel Length (in):	5.50
Water-Outer Sheath Surface Heat Flux (BTU/hr ft ²):	498,500	Fuel Diameter (in):	0.340 ± 0.0005
Film Coefficient (BTU/hr ft ² $^{\circ}F$):	6,456	Fuel-Sheath Diametral Gap (in):	0.004 ± 0.0002

B. SPECIFIC FUEL PINS

Parameter		37	38	39	40
Goal Burnup (MWD/MTU)		17,000	17,000	17,000	17,000
ETR Cycles (inclusive)		80-85	80-85	80-85	80-85
Initial Kw/ft		17.30	17.30	18.07	18.07
Terminal Kw/ft		16.33	16.33	17.05	17.05
Fuel Density (% Theoretical)		90	90	94	94
g U ²³⁵ /in. Fuel		1.74	1.74	1.82	1.82
Inner Sheath		S/S 0.005" Wall x 0.343" I.D.	Zirc. 0.024" Wall x 0.343" I.D.	S/S 0.005" Wall x 0.343" I.D.	----
Components (g)	U ²³⁵	9.56	9.56	10.0	10.0
	U ²³⁸	67.7	67.7	70.8	70.8
	304 Stainless Steel	167	133.5	167	68
	Zircaloy 4	---	50.4	---	122.5
	NaK (56 w/oK)	6.50	3.42	6.50	----
Temperature Distribution ($^{\circ}F$)	Outer Sheath O.D.	201	201	201	201
	Outer Sheath I.D.	293	293	293	601
	Inner Sheath O.D.	420	355	420	---
	Inner Sheath I.D.	447	508	447	---
	Fuel-Sheath ΔT	276	276	276	276
	Fuel Surface	723	784	723	877
	Fuel Center	2960	3340	2960	3410
Outer Sheath		S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	Zirc. 0.065" Wall x 0.343" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin):	18	Film ΔT ($^{\circ}\text{F}$):	77
Required Unperturbed Thermal Neutron Flux (Av):	9.355×10^{13}	Mean Bulk Water Temperature ($^{\circ}\text{F}$):	124
Fuel Enrichment (w/o U235 in Total U):	15.96	Fuel Length (in):	5.50
Water-Outer Sheath Surface Heat Flux (ETU/hr ft ²):	498,500	Fuel Diameter (in):	0.340 ± 0.0005
Film Coefficient (BTU/hr ft ² $^{\circ}\text{F}$):	6,456	Fuel-Sheath Diametral Gap (in):	0.002 ± 0.0002

B. SPECIFIC FUEL PINS

Parameter		41	42	43	44
Goal Burnup (MWD/MTU)		17,000	17,000	17,000	17,000
ETR Cycles (inclusive)		80-85	80-85	80-85	80-85
Initial Kw/ft		18.35	18.35	18.35	18.35
Terminal Kw/ft		17.78	17.78	17.78	17.78
Fuel Density (% Theoretical)		96.2	96.2	96.2	96.2
g U ²³⁵ /in. Fuel		2.25	2.25	2.25	2.25
Inner Sheath		S/S 0.005" Wall x 0.343" I.D.	Zirc. 0.024" Wall x 0.343" I.D.	-----	Zirc. 0.024" Wall x 0.343" I.D.
Components (g)	U ²³⁵	12.37	12.37	12.37	12.37
	U ²³⁸	70.7	70.7	70.7	70.7
	304 Stainless Steel	167	133.5	68	133.5
	Zircaloy 4	---	50.4	122.5	50.4
	NaK (56 w/oK)	6.50	3.42	---	3.42
Temperature Distribution ($^{\circ}\text{F}$)	Outer Sheath O.D.	201	201	201	201
	Outer Sheath I.D.	293	293	601	293
	Inner Sheath O.D.	420	355	---	355
	Inner Sheath I.D.	447	508	---	508
	Fuel-Sheath ΔT	276	276	276	276
	Fuel Surface	723	784	877	784
	Fuel Center	2960	3340	3410	3340
Outer Sheath		S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	Zirc. 0.065" Wall x 0.343" I.D.	S/S 0.020" Wall x 0.431" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin):	24	Film ΔT ($^{\circ}F$):	103
Required Unperturbed Thermal Neutron Flux (Av):	$1.229 \cdot 10^{14}$	Mean Bulk Water Temperature ($^{\circ}F$):	126
Fuel Enrichment (w/o U235 in Total U):	18.64	Fuel Length (in):	5.50
Water-Outer Sheath Surface Heat Flux (BTU/hr ft ²):	664,700	Fuel Diameter (in):	0.340 ± 0.0005
Film Coefficient (BTU/hr ft ² $^{\circ}F$):	6,456	Fuel-Sheath Diametral Gap (in):	0.006 ± 0.0002

B. SPECIFIC FUEL PINS

Parameter		45	46	47	48
Goal Burnup (MWD/MTU)		50,000	50,000	50,000	50,000
ETR Cycles (inclusive)		76-88	76-88	76-88	76-88
Initial Kw/ft		25.21	25.21	25.21	23.59
Terminal Kw/ft		22.44	22.44	22.44	20.99
Fuel Density (% Theoretical)		96.2	96.2	96.2	90
g U235/in. Fuel		2.63	2.63	2.63	2.41
Inner Sheath		S/S 0.005" Wall x 0.343" I.D.	Zirc. 0.024" Wall x 0.343" I.D.	-----	S/S 0.005" Wall x 0.343" I.D.
Components (g)	U235	14.46	14.46	14.46	13.25
	U238	68.5	68.5	68.5	62.8
	304 Stainless Steel	167	133.5	68	167
	Zircaloy 4	---	50.4	122.5	---
	NaK (56 w/oK)	6.50	3.42	----	6.50
Temperature Distribution ($^{\circ}F$)	Outer Sheath O.D.	229	229	229	229
	Outer Sheath I.D.	349	349	763	349
	Inner Sheath O.D.	518	432	---	518
	Inner Sheath I.D.	553	624	---	553
	Fuel-Sheath ΔT	368	368	368	368
	Fuel Surface	921	992	1131	921
	Fuel Center	4090	4200	4360	4090
Outer Sheath		S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	Zirc. 0.065" Wall x 0.343" I.D.	S/S 0.020" Wall x 0.431" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin):	24	Film ΔT ($^{\circ}F$):	103
Required Unperturbed Thermal Neutron Flux (A_v):	1.229×10^{14}	Mean Bulk Water Temperature ($^{\circ}F$):	126
Fuel Enrichment, (w/o U235 in Total U):	18.64	Fuel Length (in):	5.50
Water-Outer Sheath Surface Heat Flux (BTU/hr ft ²):	664,700	Fuel Diameter (in):	0.340 $\pm$ 0.0005
Film Coefficient (BTU/hr ft ² $^{\circ}F$):	6,456	Fuel-Sheath Diametral Gap (in):	0.006 $\pm$ 0.0002

B. SPECIFIC FUEL PINS

Parameter		49	50	51	52
Goal Burnup (MWD/MTU)		50,000	50,000	50,000	50,000
ETR Cycles (inclusive)		76-88	76-88	76-88	76-88
Initial Kw/ft		24.63	24.63	23.59	24.63
Terminal Kw/ft		21.93	21.93	20.99	21.93
Fuel Density (% Theoretical)		94	94	90	94
g U235/in. Fuel		2.52	2.52	2.41	2.52
Inner Sheath		S/S 0.005" Wall x 0.343" I.D.	S/S 0.015" Wall x 0.343" I.D.	-----	-----
Components (g)	U235	13.85	13.85	13.25	13.85
	U238	65.7	65.7	62.8	65.7
	304 Stainless Steel	167	180	68	68
	Zircaloy 4	---	---	122.5	122.5
	NaK (56 w/oK)	6.50	4.90	----	----
Temperature Distribution ($^{\circ}F$)	Outer Sheath O.D.	229	229	229	229
	Outer Sheath I.D.	349	349	763	763
	Inner Sheath O.D.	518	472	---	---
	Inner Sheath I.D.	553	576	---	---
	Fuel-Sheath ΔT	368	368	368	368
	Fuel Surface	921	944	1131	1131
	Fuel Center	4090	4170	4360	4360
Outer Sheath		S/S 0.020" Wall x 0.431" I.D.	S/S 0.020" Wall x 0.431" I.D.	Zirc. 0.065" Wall x 0.343" I.D.	Zirc. 0.065" Wall x 0.343" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin):	24	Film ΔT ($^{\circ}\text{F}$):	103
Required Unperturbed Thermal Neutron Flux (Av):	1.207×10^{13}	Mean Bulk Water Temperature ($^{\circ}\text{F}$):	126
Fuel Enrichment (w/o U235 in Total U):	18.64	Fuel Length (in):	5.50
Water-Outer Sheath Surface Heat Flux (BTU/hr ft ²):	664,700	Fuel Diameter (in):	0.340 + 0.000
Film Coefficient (BTU/hr ft ² $^{\circ}\text{F}$):	6,456	Fuel-Sheath Diametral Gap (in):	0.006 $\pm$ 0.000

B. SPECIFIC FUEL PINS

Parameter		53	54	55	56
Goal Burnup (MWD/MTU)		34,000	34,000	34,000	34,000
ETR Cycles (inclusive)		79-87	79-87	79-87	79-87
Initial Kw/ft		24.19	24.76	24.19	24.76
Terminal Kw/ft		22.54	23.07	22.54	23.07
Fuel Density (% Theoretical)		94	96.2	94	96.2
g U235/in. Fuel		2.52	2.63	2.52	2.63
Inner Sheath		*S/S 0.015" Wall x 0.343" I.D.	---	---	Zirc. 0.024" Wall x 0.343" I.D.
Components (g)	U235	13.85	14.46	13.85	14.46
	U238	65.7	68.5	65.7	68.5
	304 Stainless Steel	180	68	68	133.5
	Zircaloy 4	---	122.5	122.5	50.4
	NaK (56 w/oK)	4.90	---	---	3.42
Temperature Distribution ($^{\circ}\text{F}$)	Outer Sheath O.D.	229	229	229	229
	Outer Sheath I.D.	349	763	763	349
	Inner Sheath O.D.	472	---	---	432
	Inner Sheath I.D.	576	---	---	624
	Fuel-Sheath ΔT	368	368	368	368
	Fuel Surface	944	1131	1131	992
	Fuel Center	4170	4360	4360	4200
Outer Sheath		S/S 0.020" Wall x 0.431" I.D.	Zirc. 0.065" Wall x 0.343" I.D.	Zirc. 0.065" Wall x 0.343" I.D.	S/S 0.020" Wall x 0.431" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin):	24	Film ΔT ($^{\circ}F$):	103
Required Unperturbed Thermal Neutron Flux (Δv):	1.207×10^{13}	Mean Bulk Water Temperature ($^{\circ}F$):	126
Fuel Enrichment (w/o U235 in Total U):	18.64	Fuel Length (in):	5.50
Water-Outer Sheath Surface Heat Flux (BTU/hr ft ²):	664,700	Fuel Diameter (in):	0.340 ± 0.0005
Film Coefficient (BTU/hr ft ² $^{\circ}F$):	6456	Fuel-Sheath Diametral Gap (in):	0.006 ± 0.0002

B. SPECIFIC FUEL PINS

Parameter		57	58	59	60
Goal Burnup (MWD/MTU)		34,000	34,000	34,000	34,000
ETR Cycles (inclusive)		79-87	79-87	79-87	79-87
Initial Kw/ft		24.76	24.76	24.19	24.19
Terminal Kw/ft		23.07	23.07	22.54	22.54
Fuel Density (% Theoretical)		96.2	96.2	94	94
g U235/in. Fuel		2.63	2.63	2.52	2.52
Inner Sheath		-	S/S 0.005" wall x 0.343" I.D.	-	S/S 0.005" wall x 0.343" I.D.
Components (g)	U235	14.46	14.46	13.85	13.85
	U238	68.5	68.5	65.7	65.7
	304 Stainless Steel	68	167	68	167
	Zircaloy 4	122.5	--	122.5	--
	NaK (56 w/oK)	--	6.50	--	6.50
Temperature Distribution ($^{\circ}F$)	Outer Sheath O.D.	229	229	229	229
	Outer Sheath I.D.	763	349	763	349
	Inner Sheath O.D.	-	518	-	518
	Inner Sheath I.D.	-	553	-	553
	Fuel-Sheath ΔT	368	368	368	368
	Fuel Surface	1131	921	1131	921
	Fuel Center	4360	4090	4360	4090
Outer Sheath		Zirc. 0.065" wall x 0.343" I.D.	S/S 0.020" wall x 0.431" I.D.	Zirc. 0.065" wall x 0.343" I.D.	S/S 0.020" wall x 0.431" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin): 24
 Required Unperturbed Thermal Neutron Flux (A_v): 1.221×10^{14}
 Fuel Enrichment (w/o ^{235}U in Total U): 17.02
 Water-Outer Sheath Surface Heat Flux (BTU/hr ft²): 664,700
 Film Coefficient (BTU/hr ft² °F): 6,456

Film ΔT (°F): 103
 Mean Bulk Water Temperature (°F): 126
 Fuel Length (in): 5.50
 Fuel Diameter (in): 0.340 ± 0.005
 Fuel-Sheath Diametral Gap (in): 0.006 ± 0.0002

B. SPECIFIC FUEL PINS

Parameter		61	62	63	64
Goal Burnup (MWD/MTU)		17,000	17,000	17,000	17,000
ETR Cycles (inclusive)		80-85	80-85	80-85	80-85
Initial Kw/ft		23.86	24.42	23.86	24.42
Terminal Kw/ft		22.99	23.53	22.99	23.53
Fuel Density (% Theoretical)		94	96.2	94	96.2
g ^{235}U /in. Fuel		2.29	2.39	2.29	2.39
Inner Sheath		S/S 0.005" wall x 0.343" I.D.	S/S 0.005" wall x 0.343" I.D.	Zirc 0.024" wall x 0.343" I.D.	S/S 0.015" wall x 0.343" I.D.
Components (g)	^{235}U	12.60	13.14	12.60	13.14
	^{238}U	67.0	69.9	67.0	69.9
	304 Stainless Steel	167	167	133.5	180
	Zircaloy 4	--	--	50.4	--
	NaK (56 w/oK)	6.50	6.50	3.42	4.90
Temperature Distribution (°F)	Outer Sheath O.D.	229	229	229	229
	Outer Sheath I.D.	349	349	349	349
	Inner Sheath O.D.	518	518	432	472
	Inner Sheath I.D.	553	553	624	576
	Fuel-Sheath ΔT	368	368	368	368
	Fuel Surface	921	921	992	944
	Fuel Center	4090	4090	4200	4170
Outer Sheath		S/S 0.020" wall x 0.431" I.D.	S/S 0.020" wall x 0.431" I.D.	S/S 0.020" wall x 0.431" I.D.	S/S 0.020" wall x 0.431" I.D.

A. GENERAL PARAMETERS

Nominal Linear Heat Rating (Kw/ft per fuel pin):	24	Film ΔT ($^{\circ}F$):	103
Required Unperturbed Thermal Neutron Flux (A_v):	1.164×10^{14}	Mean Bulk Water Temperature ($^{\circ}F$):	126
Fuel Enrichment (w/o U ²³⁵ in Total U):	19.96	Fuel Length (in):	5.50
Water-Outer Sheath Surface Heat Flux (BTU/hr ft ²):	664,700	Fuel Diameter (in):	0.340 ± 0.0005
Film Coefficient (BTU/hr ft ² $^{\circ}F$):	6,456	Fuel-Sheath Diametral Gap (in):	0.006 ± 0.0002

B. SPECIFIC FUEL PINS

Parameter		65	66	67	68
Goal Burnup (MWD/MTU)		17,000	17,000	17,000	17,000
ETR Cycles (inclusive)		80-85	80-85	80-85	80-85
Initial Kw/ft		23.86	24.42	24.42	23.86
Terminal Kw/ft		22.99	23.64	23.64	22.99
Fuel Density (% Theoretical)		94	96.2	96.2	94.0
g U ²³⁵ /in. Fuel		2.69	2.81	2.81	2.69
Inner Sheath		Zirc 0.024" wall x 0.343" I.D.	Zirc. 0.024" wall x 0.343" I.D.	-	-
Components (g)	U ²³⁵	14.80	15.45	15.45	14.80
	U ²³⁸	64.6	67.4	67.4	64.6
	304 Stainless Steel	133.5	133.5	68	68
	Zircaloy 4	50.4	50.4	122.5	122.5
	NaK 156 w/oK)	3.42	3.42	--	--
Temperature Distribution ($^{\circ}F$)	Outer Sheath O.D.	229	229	229	229
	Outer Sheath I.D.	349	349	763	763
	Inner Sheath O.D.	432	432	-	-
	Inner Sheath I.D.	624	624	-	-
	Fuel-Sheath ΔT	368	368	368	368
	Fuel Surface	992	992	1131	1131
	Fuel Center	1200	1200	1360	1360
Outer Sheath		S/S 0.020" wall x 0.431" I.D.	S/S 0.020" wall x 0.431" I.D.	Zirc 0.065" wall x 0.343" I.D.	Zirc 0.065" wall x 0.343" I.D.

APPENDIX B
DESIGN DRAWINGS

Note: Figures B-1 to B-3 in this Appendix represent the original design. As discussed in the text, the following modifications were incorporated in the original design following the first failure incident.

1. The Cu-Be snap rings were replaced with stainless steel wire.
2. The top and bottom ends of the fuel pins were fastened rigidly to the capsule.

Figure B-4 shows the change in design incorporated after the second failure incident. These changes are:

1. The use of locating bosses (items 1 and 2),
2. The provision of a circumferential metal restraining band.

NO.	DESCRIPTION
1	FUEL PIN ASSEMBLY
2	FUEL PIN ASSEMBLY
3	FUEL PIN ASSEMBLY
4	FUEL PIN ASSEMBLY
5	HANDLING
6	CAPSULE HOLDER
7	CAPSULE HOLDER
8	RETAINING RING
9	2000164

Figure B-1

CAPSULE - BASKET GENERAL
ASSEMBLY - UO_2 SWELLING TESTS

B-3

NO.	DESCRIPTION
1	END PLUG
2	INNER PLUG
3	INNER PLUG
4	INNER PLUG
5	OUTER TUBE
6	INNER TUBE
7	INNER TUBE
8	INNER TUBE
9	SPACER
10	DEPLETED PELLET
11	PELLET
12	END PLUG
13	OUTER TUBE

Figure B-2

FUEL PIN ASSEMBLIES -
UO₂ SWELLING TESTS.

Figure B-3

REPLACABLE FUEL PIN -
FLOW BLOCKER ASSEMBLIES -
UO₂ SWELLING TESTS.

Figure B-4. Test Capsule Revised Design: Cross-Section with Spacer Pins

APPENDIX C
POWER GENERATION CALCULATIONS

POWER GENERATION

Power generation was calculated from:

$$Q/L = \frac{\phi_p \Sigma_f}{K} = \frac{f \phi_u \Sigma_f}{K}$$

Q/L = heat output per unit length of fuel, kw/ft

ϕ_p = perturbed thermal flux, nv

Σ_f = macroscopic fission cross section, cm^{-1}

K = conversion factor, fission/sec $\text{cm}^3/\text{kw/ft}$

f = flux perturbation factor

ϕ_u = unperturbed thermal flux, nv

$$\Sigma_f = (G_f) (N_{235})$$

G_f = microscopic fission cross section, cm^2

N_{235} = density of U^{235} atoms, atoms/cm^3

$$G_f = 555 \text{ barns} = 555 \times 10^{-24} \text{ cm}^2$$

$$N_{235} = N_w \rho_F \left(\frac{235N_w + 238(1 - N_w)}{235N_w + 238(1 - N_w) + 32} \right) \left(\frac{6.023 \times 10^{23}}{235} \right)$$

$$\approx 2.255 \times 10^{21} \rho_F N_w$$

ρ_F = fuel density, g/cm^3

N_w = weight fraction of U^{235}

Fuel densities range from 10.15 to 10.54 g/cm³; 10.54 will be used in subsequent calculations.

$$N_{235} \approx 2.40 \times 10^{22} N_w$$

$$\Sigma_f = (5.55 \times 10^{-22}) (2.40 \times 10^{22}) N_w = 13.3 N_w$$

Calculation of K

$$\text{Volume per unit fuel length} = \frac{\pi D^2 L}{4}$$

$$V = \frac{\pi (0.340)^2 (12)}{4} = 1.09 \text{ in}^3/\text{ft} = 17.8 \text{ cm}^3/\text{ft}$$

$$\text{kw/ft} = \frac{2.247 \times 10^{19} \text{ mev/kw hr}}{3.6 \times 10^3 \text{ sec/hr } 17.8 \text{ cm}^3/\text{ft}} = 3.507 \times 10^{14} \frac{\text{mev}}{\text{sec cm}^3}$$

$$1 \text{ fission} = 180 \text{ mev}$$

$$K = \frac{3.5 \times 10^{14} \text{ mev/sec cm}^3}{(1 \text{ kw/ft}) (180 \text{ mev/fission})} = 1.951 \times 10^{12} \frac{\text{fissions/sec cm}^3}{\text{kw/ft}}$$

Calculation of f

$$f = \frac{(0.75) (0.8)}{1 + (1.05) (\text{gU}^{235}/\text{in})}$$

From NAA-SR-6039, Aug. 1959, the factor 0.8 is an experience factor and the 0.05 added to the second term in the denominator accounts for the poison effect of equilibrium xenon.

$$\text{gU}^{235}/\text{in} = \frac{(17.8 \text{ cm}^3 \text{UO}_2/\text{ft}) (\rho \text{ g UO}_2/\text{cm}^3 \text{UO}_2) (0.881 \text{ gU/gUO}_2)}{(12 \text{ in/ft})} \left(\frac{E}{100} \right)$$

$$E = \text{percent enrichment (w/o U}^{235} \text{ in Total U)} = 100 N_w$$

$$\rho = 10.6$$

$$\text{gU}^{235}/\text{in} = 0.138E$$

$$f = \frac{0.6}{1 + 0.145E} = \frac{0.6}{1 + 14.5N_w}$$

$$Q = \frac{(f\phi_u)(13.3N_w)}{1.95 \times 10^{12}} = 6.85 \times 10^{-12} f\phi_u N_w$$

$$\phi_u = \frac{1.46 \times 10^{11} Q}{f N_w}$$

Specific Flux Calculations

<u>Initial Goal Rating (kw/ft)</u>	<u>N_w</u>	<u>f</u>	<u>φ_u</u>
12.47	0.1864	0.1625	6.078 x 10 ¹³
12.47	0.1864	0.1625	6.078 x 10 ¹³
12.31	0.1864	0.1625	6.002 x 10 ¹³
12.31	0.1864	0.1625	6.002 x 10 ¹³
19.22	0.1596	0.1815	9.796 x 10 ¹³
19.70	0.1340	0.2044	1.062 x 10 ¹⁴
18.79	0.1996	0.1546	8.993 x 10 ¹³
18.67	0.1596	0.1815	9.517 x 10 ¹³
18.57	0.1864	0.1625	9.053 x 10 ¹³
18.49	0.1340	0.2044	9.965 x 10 ¹³
18.35	0.1596	0.1815	9.355 x 10 ¹³
25.21	0.1864	0.1625	1.229 x 10 ¹⁴
25.21	0.1864	0.1625	1.229 x 10 ¹⁴
24.76	0.1864	0.1625	1.207 x 10 ¹⁴
24.76	0.1864	0.1625	1.207 x 10 ¹⁴
24.42	0.1702	0.1735	1.221 x 10 ¹⁴
24.42	0.1996	0.1546	1.164 x 10 ¹⁴