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THE IN-PILE THERMAL CONDUCTIVITY OF SELECTED ThO₂-UO₂ FUELS AT LOW DEPLETIONS (LWBR Development Program)

D. C. Jacobs

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NOMENCLATURE

Symbol	Description
A_s	constant
A_T	constant
α	constant
B_s	constant
B_T	constant
β	constant
C_T	constant
F_a	axial gamma-heating factor
G	ratio of the gamma heating in the fuel to the gamma heating in stainless steel
H_f	total heat flow/length from the fuel
h_f	heat flow/length in fuel
h_w	heat flow/length in the capsule wall
I_0	zero-order modified Bessel function
I_1	first-order modified Bessel function
K	reciprocal neutron diffusion length
K_a	adjusted reciprocal neutron diffusion length
k_f	thermal conductivity of the fuel
k_s	thermal conductivity of stainless steel
Q_f	power density in the fuel by fission and gamma heating
Q_{ff}	power density in the fuel by fission
Q_{gf}	power density in the fuel by gamma heating
Q_{gg}	power density by gamma heating in the gamma monitor
Q_{gw}	power density by gamma heating in the capsule wall
R_f	outer radius of the fuel
r	radial position
r_o	inner radius of the capsule wall
Σ_s	scattering cross section

NOMENCLATURE (Cont)

<u>Symbol</u>	<u>Description</u>
Σ_a	absorption cross section
T	temperature
$T_f(r)$	radial distribution in temperature
$\bar{T}(r)$	mean radial temperature

The in-pile thermal conductivities of selected ThO₂-UO₂ fuel materials were determined by direct measurement using an in-pile extension of the radial heat-flow technique. The tests incorporated ThO₂ + 3.1 w/o UO₂, ThO₂ + 7.7 w/o UO₂, ThO₂ + 10 w/o UO₂, and UO₂ specimens. The microstructure, stoichiometry, density, impurity content, and unirradiated thermal conductivity were determined for each fuel material. The test results showed no statistically significant differences between the in-pile and unirradiated thermal conductivities for steady-state operation for the UO₂ specimens at $600 \leq T \leq 1000^\circ\text{C}$ at a burnup of up to 5.6×10^{18} fission/cm³ and for the ThO₂-UO₂ specimens at $400 \leq T \leq 1000^\circ\text{C}$ at a burnup of up to 1.4×10^{19} fission/cm³. The in-pile conductivities were slightly, but significantly, greater than the unirradiated conductivities at $1000 < T \leq 1300^\circ\text{C}$ for the UO₂ specimens and at $1000 < T \leq 1700^\circ\text{C}$ for the ThO₂-UO₂ specimens. However, this is considered to be an experimental anomaly and is not taken as a meaningful enhancement of in-pile conductivity. The maximum depletions achieved in these tests are one to two orders of magnitude less than typical in-service depletions. Hence, the test results do not reflect any possible changes in conductivity by the accumulation of fission products and the formation of fission gas bubbles.

THE IN-PILE THERMAL CONDUCTIVITY OF SELECTED
ThO₂-UO₂ FUELS AT LOW DEPLETIONS
(LWBR Development Program)

D. C. Jacobs

I. INTRODUCTION

The purpose of the tests reported herein was to determine, by *in-situ* measurements, the in-pile thermal conductivity of selected ThO₂-UO₂ fuels at relatively low burnups (approximately 10^{19} fission/cm³). The experimental procedure was an in-pile extension of the radial heat-flow technique whereby thermal conductivity is inferred from a measured temperature distribution and from a known heat input. The fuel specimens were contained within thick-walled stainless steel capsules, and the heat input was provided by fission and gamma heating. The power densities by both the fission and gamma-heating processes

were determined by temperature measurements in the capsule wall and in a neighboring gamma monitor. The temperature distribution within the fuel specimen was inferred from temperature measurements made at three radial locations in the specimen pellet.

The tests incorporated five groups of fuel, that is, 95-percent dense $\text{ThO}_2 + 3.1 \text{ w/o } \text{UO}_2$, 74-percent dense $\text{ThO}_2 + 7.7 \text{ w/o } \text{UO}_2$, 98-percent dense $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, 86-percent dense $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, and 98-percent dense UO_2 . For each fuel material the microstructure, stoichiometry, density, impurity content, and unirradiated thermal conductivity were determined.

The experimental in-pile temperatures ranged between 200 and 1700C. The maximum burnups achieved during the acquisition of data varied from 0.7×10^{18} fission/cm³ to 1.4×10^{19} fission/cm³, the latter being one to two orders of magnitude less than typical in-service depletions.

The details of the test hardware, experimental procedures, operating histories, and test results are given in the following sections.

II. TEST HARDWARE

A. Fuel Capsules

A typical fuel capsule is conceptually shown in Figure 1. It is a thick-walled hollow cylinder of stainless steel (type 304) containing the test pellets. Temperatures within the fuel are measured at three radial positions by thermocouples penetrating the capsule through the lower and/or upper end caps. Temperatures within the capsule wall are measured by thermocouples located at two radial positions. In the first two tests,* two pairs of wall thermocouples, 180 degrees apart, were utilized. In all subsequent tests four pairs of thermocouples, 90 degrees apart, were utilized.

The nominal outer and inner diameters of the capsules were 2.2 and 1.0 inches, respectively. The capsules were 6.75 inches high, and the assembled fuel stacks (9 or 10 pellets) were nominally 5.0 inches high. In Test 1 the as-built diametric clearance between the fuel and capsule wall was approximately 0.010 inch. In all subsequent tests the as-built clearance was maintained at 0.001 to 0.002 inch by nickel plating and/or machined bores. HfO_2 spacer pellets were placed at both ends of the fuel column. These served as thermal insulation to reduce end losses and to minimize nuclear end-peaking. The entire stack of fuel and HfO_2 pellets were maintained in contact by an Inconel spring at the upper end of the column.

*The tests in this series were numbered BETT-84-1, -2, -3, -4, and -5, but in this report they will be referred to merely as Tests 1, 2, 3, 4, and 5.

In all cases the final welding operations were performed in a dry helium atmosphere at standard temperature and pressure. The capsule integrity was assured using standard leak-detection procedures.

All data and measurements for each capsule were referenced to the fuel pellet occupying the central position in the fuel column (the prime pellet). The hot junctions of all fuel thermocouples were located in the prime pellet, and the hot junctions of the wall thermocouples were oriented along a diametric plane through the prime pellet.

B. Gamma Monitors

The gamma monitors were solid cylinders of stainless steel (type 304), which were nominally 2.2 inches in diameter and 5.0 inches high. Each monitor contained five thermocouples, one at the center and the others at two additional radial locations.

C. Test Assemblies

Each test assembly consisted of one or more fuel capsules and a single gamma monitor arranged in a vertical train. The capsules were separated by stainless steel spacers which served to minimize the amount of coolant water between the capsules, thereby reducing nuclear end-peaking. The assemblies were held rigid by pins extending between capsules and through the stainless steel spacers (Figure 1).

The assembled test trains were placed in aluminum reflector pieces which served to adapt the cylindrical geometry of the trains to the square reflector grids of the Experimental Testing Reactor (ETR). An assembled test train, prior to insertion in the reflector piece, is shown in Figure 2.

D. Thermocouples

1. General

In Tests 1, 3, 4, and 5 chromel-alumel thermocouples were used in the capsule walls and in the gamma monitor. Tungsten-rhenium (W - 5% Re/W - 26% Re) thermocouples were used in the fuel, since fuel temperatures in excess of 1200C were anticipated. In Test 2 chromel-alumel thermocouples were used in the fuel, capsule walls, and gamma monitor.

2. Chromel-Alumel Thermocouples

The chromel-alumel thermocouples having grounded hot junctions were insulated with Al_2O_3 and sheathed with type 348 stainless steel. The nominal wire and sheath diameters were 0.013 and 0.065 inch, respectively.

The chromel-alumel thermocouples were referenced to standard calibration tables (Reference 1). To facilitate the use of the digital computer in data reduction, EMF-temperature relationships were formulated using regression techniques. The equations for the chromel-alumel thermocouples were obtained by linearly fitting the standard calibration data within each 100C-degree increment for $0 \leq T \leq 1370\text{C}$. The regression equations and their constants are given in Table 1. The maximum deviation between true calibration and the calculated chromel-alumel temperature is $\pm 0.5\text{C}$.

3. Tungsten-Rhenium Thermocouples

The tungsten-rhenium (W - 5% Re/W - 26% Re) thermocouples were sheathed with W - 25% Re and insulated with BeO for the first 3.5 inches from the hot junction end (hot tip). The remaining length (lead section), approximately 27 feet, was sheathed with type 347 seamless stainless steel and insulated with Al_2O_3 . The sheath diameter was nominally 0.0625 inch. The transition between the 3.25-inch hot tip and the stainless steel sheathed lead section was made using a stainless steel transition sleeve which was slipped over and weld-brazed to each sheath segment.

The hot junctions were isolated (nongrounded) and insulated with a single hole BeO bead. The junctions were formed by tightly wrapping tungsten-rhenium wire around the thermocouple legs. A small welded bead was made at the end of the wire wrap to prevent unraveling. In this arrangement the effective hot junction is at the base of the wire wrap, which is sufficiently removed from the embrittled weld area. (This junction design was chosen to eliminate the brittle hot junction characteristic of welded beads.)

The lead sections of the tungsten-rhenium thermocouples were formed by swaging. No swaging was done to the 3.25-inch hot tip. This region was insulated by slipping the thermocouple legs into the BeO pellets which were then placed into the W - 25% Re sheath having the finished outside diameter. This fabrication procedure assured that the wires were loosely held in the hot tip, and therefore, free to accommodate differential expansion in an axial thermal gradient.

TABLE 1. ELECTROMOTIVE FORCE - TEMPERATURE RELATIONSHIPS FOR
CHROMEL-ALUMEL AND W - 5% Re/W - 26% Re THERMOCOUPLES

$$\text{Chromel-Alumel } T(^{\circ}\text{C}) = A_1 + A_2 \left\{ (\text{EMF}) - A_3 \right\} *$$

A_1	A_2	A_3	Electromotive Force (EMF) Range
0.0	24.39024	0.0	0.00 $\leq$ to < 4.10
100.0	24.81389	4.10	4.10 $\leq$ to < 8.13
200.0	24.50980	8.13	8.13 $\leq$ to < 12.21
300.0	23.86634	12.21	12.21 $\leq$ to < 16.40
400.0	23.52941	16.40	16.40 $\leq$ to < 20.65
500.0	23.47417	20.65	20.65 $\leq$ to < 24.91
600.0	23.64066	24.91	24.91 $\leq$ to < 29.14
700.0	24.03846	29.14	29.14 $\leq$ to < 33.30
800.0	24.63054	33.30	33.30 $\leq$ to < 37.36
900.0	25.31645	37.36	37.36 $\leq$ to < 41.31
1000.0	25.97402	41.31	41.31 $\leq$ to < 45.16
1100.0	26.80965	45.16	45.16 $\leq$ to < 48.89
1200.0	28.01120	48.89	48.89 $\leq$ to < 52.46
1300.0	28.92561	52.46	52.46 $\leq$ to < 54.88

$$\text{W - 5\% Re/W - 26\% Re } T(^{\circ}\text{C}) = B_1(\text{EMF}) + B_2(\text{EMF})^2 + \dots + B_5(\text{EMF})^5 \dagger$$

	Tests 1, 3, and 4	Test 5
B_1	73.80076	72.24462
B_2	-3.235298	-2.762305
B_3	0.2034424	0.1604651
B_4	-5.710775×10^{-3}	-4.135108×10^{-3}
B_5	6.366214×10^{-5}	4.401466×10^{-5}

*Obtained by linearly fitting the calibration data (Reference 1) within each 100C-degree increment for $0 \leq T \leq 1370\text{C}$. Maximum error $\pm 0.5\text{C}$.

†Obtained by fitting the thermocouple wire calibration data supplied by the vendor. Twenty-one points for $200 \leq T \leq 2315\text{C}$. Maximum error $\pm 3\text{C}$.

For each group of tungsten-rhenium thermocouples in the tests, the vendor supplied calibration data for the particular batches of wire used. Calibration checks were made by the vendor after fabrication and by Bettis after receipt. The wire calibration checks showed no significant shifts.

The EMF-temperature relationships for each wire batch were formulated using regression techniques. In each case the wire calibration data were fit by a five term polynomial. A total of 21 points were utilized for $200 \leq T \leq 2315\text{C}$. The regression equations and their constants are given in Table 1. The maximum deviation between true calibration and the calculated tungsten-rhenium thermocouple temperature is $\pm 3\text{C}$ for $200 \leq T \leq 2000\text{C}$.

III. DATA REDUCTION

A. Analytic Equations

In order for the capsule wall to function as a calorimeter, the heat developed in the fuel and the heat developed in the wall itself (by gamma heating) must be analytically separable. The latter is inferred from the gamma-heating rate measured directly in the gamma monitor. That is, with a known axial distribution in gamma heating along the test train, the gamma heating in the capsule wall is:

$$Q_{gw} = F_a Q_{gg} \quad (1)$$

where:

Q_{gw} = gamma heating in the capsule wall

F_a = axial gamma-heating factor

Q_{gg} = gamma-heating rate density in the gamma monitor.

The gamma heating in the monitor is calculated from the relationship:

$$2\pi \int_0^r r Q_{gg} dr = -2\pi r k_s(T) \frac{dT}{dr} \quad (2)$$

where:

$k_s(T)$ = thermal conductivity of stainless steel (taken to be a function of temperature).

Integration of Equation (2) between two radial positions, r_1 and r_2 , yields:

$$Q_{gg} = \frac{-4}{(r_2^2 - r_1^2)} \int_{T(r_1)}^{T(r_2)} k_s(T) dT \quad (3)$$

The thermal conductivity of stainless steel is linear over the temperature of interest; thus,

$$k_s(T) = A_s + B_s T . \quad (4)$$

Combining Equations (3) and (4), the power density in the gamma monitor is given by:

$$Q_{gg} = \frac{4}{(r_2^2 - r_1^2)} \left\{ A_s [T(r_1) - T(r_2)] + (B_s/2) [T^2(r_1) - T^2(r_2)] \right\} . \quad (5)$$

Equation (5) is perfectly general and can be used to calculate the power density from any two thermocouple indications. In practice, it is used to calculate three values, $Q_{gg}(0, r_1)$, $Q_{gg}(0, r_2)$, and $Q_{gg}(r_1, r_2)$ where the quantities in parenthesis indicate the particular limits of integration and correspond to the radial positions of the thermocouples in the monitor. These three values are averaged to yield $\bar{Q}_{gg}$ with suitable weight given to $Q_{gg}(0, r_2)$, that is,

$$\bar{Q}_{gg} = (1/4) [Q_{gg}(0, r_1) + Q_{gg}(r_1, r_2)] + (1/2) Q_{gg}(0, r_2) . \quad (6)$$

The gamma-heating rate in the capsule wall is calculated from Equations (1) and (6). This quantity and the measured temperatures within the capsule wall are used to calculate the total heat flow/length from the fuel. The heat flow/length at any position in the wall is:

$$h_w(r) = H_f + 2\pi Q_{gw} \int_{r_0}^r r dr = -2\pi r k_s(T) \frac{dT}{dr} \quad (7)$$

where:

$h_w(r)$ = heat flow/length at any position in the wall

H_f = total heat flow/length from the fuel

r_0 = inner radius of the capsule wall.

Therefore:

$$- \int_{T(r_0)}^{T(r)} k_s(T) dT = \frac{H_f}{2\pi} \int_{r_0}^r \frac{dr}{r} + \frac{Q_{gw}}{2} \int_{r_0}^r \frac{(r^2 - r_0^2)}{r} dr . \quad (8)$$

The heat flow/length from the fuel is determined by inserting Equation (4) and integrating Equation (8) between r_4 and r_5 , the radial positions of the wall thermocouples:

$$H_f = \left[2\pi / \ln(r_5/r_4) \right] \left\{ A_s [T(r_4) - T(r_5)] + (B_s/2) [T^2(r_4) - T^2(r_5)] - (Q_{gw}/2) [(r_5^2 - r_4^2)/2 - r_o^2 \ln(r_5/r_4)] \right\} . \quad (9)$$

By definition, the mean power density in the fuel is:

$$\bar{Q}_f = H_f / \pi R_f^2 \quad (10)$$

where:

$\bar{Q}_f$ = mean power density in the fuel

R_f = outer radius of the fuel.

The mean power density in the fuel is the sum of the mean power densities by fission, $\bar{Q}_{ff}$, and gamma heating, $\bar{Q}_{gf}$. The latter is inferred from the gamma heating in the capsule wall by utilizing the ratios of gamma heating in the fuel to the gamma heating in stainless steel, G. Therefore,

$$\bar{Q}_{gf} = G \bar{Q}_{gw} \quad (11a)$$

$$\bar{Q}_{ff} = \bar{Q}_f - \bar{Q}_{gf} . \quad (11b)$$

In order to determine the thermal conductivity of the fuel, the radial distributions in heat flow/length and in temperature must be known. The former is given by:

$$h_f(r) = 2\pi \int_0^r r Q_f(r) dr \quad (12)$$

where:

$h_f(r)$ = radial distributions in heat flow/length

$Q_f(r)$ = radial distribution in power density by fission and gamma heating.

The power density by fission can be expressed in terms of the neutron diffusion approximation:

$$Q_{ff}(r) = \beta I_o (Kr) \quad (13)$$

where:

β = constant

K = reciprocal neutron diffusion length

$I_0(Kr)$ = zero-order modified Bessel function.

Taking gamma heating to be independent of position, combining Equations (12) and (13), and integrating yields:

$$h_f(r) = \alpha r I_1(Kr) + \pi r^2 Q_{gf} \quad (14)$$

where:

α = constant = $2\pi\beta/K$

$I_1(Kr)$ = first-order modified Bessel function.

The constant α can be determined by evaluating Equation (14) at the outer radius of the fuel:

$$\alpha = [h_f(R_f) - \pi R_f^2 Q_{gf}] / R_f I_1(KR_f) \quad (15)$$

where:

R_f = outer radius of the fuel

$h_f(R_f) = H_f$ as determined in Equation (9).

The radial distribution in temperature is fit to the functional form:

$$T_f(r) = A_T + B_T \left\{ 1 - [I_0(Kr)/I_0(KR_f)] \right\} + C_T [R_f^2 - r^2] \quad (16)$$

where the constants A_T , B_T , and C_T are determined using the measured temperature data from the three radial thermocouple positions. The form of Equation (16) is that of the approximate solution of the Fourier equation in which the distribution in power density is given by:

$$Q_f(r) = \beta I_0(Kr) + Q_{gf} \quad (17)$$

Equations (14) and (15) can be used to calculate the thermal conductivity integrals and the corresponding mean conductivities at two locations in the fuel, that is, between the three radial thermocouple locations.

Specifically, if in the general equation,

$$h_f(r) = -2\pi r k_f(r) \frac{dT}{dr}, \quad (18)$$

the thermal conductivity of the fuel, k_f , is taken to be a function only of temperature, then:

$$\int_{T(o)}^{T(r)} k_f(T) dT = -(1/2\pi) \int_0^r \frac{h_f(r) dr}{r}. \quad (19)$$

Combining Equations (14) and (19) and integrating between r_1 and r_2 , two radial positions of fuel thermocouples, the conductivity integral is:

$$\int_{T(r_2)}^{T(r_1)} k_f(T) dT = (\alpha/2\pi K) [I_o(Kr_2) - I_o(Kr_1)] + (Q_{gf}/4)(r_2^2 - r_1^2). \quad (20)$$

The corresponding mean thermal conductivity, $\bar{k}_f$, between r_1 and r_2 , is then:

$$\bar{k}_f(r_1, r_2) = \left\{ 1/[T(r_1) - T(r_2)] \right\} \int_{T(r_2)}^{T(r_1)} k_f(T) dT \quad (21a)$$

or

$$\begin{aligned} \bar{k}_f(r_1, r_2) = & \left\{ (\alpha/2\pi K) [I_o(Kr_2) - I_o(Kr_1)] \right. \\ & \left. + (Q_{gf}/4)(r_2^2 - r_1^2) \right\} / [T(r_1) - T(r_2)]. \end{aligned} \quad (21b)$$

Equation (18) is perfectly general and makes no assumption as to why the conductivity at r has its particular value. However, in the formulation of Equation (19), it is tacitly assumed that the conductivity is a function of temperature only, and that it is not influenced by other factors such as inhomogeneities, phase changes, etc.

An alternate procedure yields a more general quantity, the radial distribution in thermal conductivity, $k_f(r)$. Taking Equation (18) in its exact form yields:

$$k_f(r) = -h_f(r)/2\pi r T_f(r) \quad (22)$$

where $h_f(r)$ and $T_f(r)$ are given by Equations (14) and (16), respectively.

The distribution, $k_f(r)$, can then be related to known temperature distributions and/or to distributions in physical characteristics. Combining Equations (14), (16), and (22) yields:

$$k_f(r) = \left[\alpha I_1(Kr) + \pi r Q_{gf} \right] / 2\pi \left[\left\{ B_{TKI_1}(Kr) / I_0(KR_f) \right\} + 2C_T r \right] . \quad (23)$$

It should be noted that both Equations (21b) and (23) are independent of the fuel surface temperature. As such, they are free from any assumptions relative to the heat transfer across the fuel-capsule interface.

The equations derived above are based on one-dimensional heat transfer, and therefore, require angular symmetry in the neutron flux distribution. Clearly, lack of such symmetry would result in asymmetric distributions in heat flow, $h(r,\theta)$, temperature, $T(r,\theta)$, and thermal conductivity, $k(r,\theta)$, (if conductivity is a function of temperature). However, the employment of the one-dimensional equations with mean radial temperatures:

$$\bar{T}(r) = (1/2\pi) \int_0^{2\pi} T(r,\theta) d\theta$$

will yield mean radial heat flows:

$$\bar{h}(r) = (1/2\pi) \int_0^{2\pi} h(r,\theta) d\theta$$

and mean radial conductivities:

$$\bar{k}(r) = (1/2\pi) \int_0^{2\pi} k(r,\theta) d\theta .$$

Since all tests occupied positions in the ETR reflector (where neutron flux gradients are significant), asymmetric temperature distributions were anticipated. Accordingly, the test capsules were oriented with respect to the core such that the line of symmetry through the pellets was normal to a diagonal from the core center (Figure 12). As such, the arithmetic mean temperature of each radial pair of thermocouples corresponded approximately to the true mean radial temperature. Therefore, in implementing the one-dimensional analytic equations, the arithmetic mean temperatures of each radial pair of thermocouples was employed, and the resultant quantity is essentially the mean radial thermal conductivity.

B. Parameters

The preceding equations utilize a number of parameters which must be determined by direct measurement or by analysis. These parameters and the source of their values are discussed below.

1. Thermocouple-Induced Temperature Perturbation

Each thermocouple within the fuel is a region of relatively high thermal conductivity essentially devoid of heat generation. Consequently, the thermocouple perturbs the true temperature distribution in its locale and will indicate this perturbed temperature. This effect is illustrated in Figures 3a and 3b which show typical radial isotherm contours in hypothetical fuel with and without thermocouples.

The perturbation effect was evaluated by numerical simulation utilizing a range of values for power density and fuel thermal conductivity. The effect was found to increase with radial position but was essentially independent of thermal conductivity and power density. Accordingly, prior to their use, the indicated fuel temperatures are multiplied by appropriate perturbation factors. The factors are:

$$\begin{aligned} P_1 &= 1.000 \text{ for } r = 0.0 \text{ inch} \\ P_2 &= 1.016 \text{ for } r = 0.20 \text{ inch} \\ P_3 &= 1.060 \text{ for } r = 0.35 \text{ inch.} \end{aligned}$$

2. Axial Gamma-Heating Factor

To obtain the axial gamma-heating factor, F_a , the assumption is made that, within the test location, the relative axial distributions in gamma heating and neutron flux are equal. On this basis the axial factors were calculated from measured flux wire data obtained from cobalt-aluminum wires situated in the particular test location and/or in neighboring locations. The axial factors for each test are given in Table 8.

3. Thermal Conductivity of Stainless Steel

The thermal conductivity data applied to the capsule walls and gamma monitors were obtained from the literature (Reference 2). These data are shown in Figure 4. The data for $127 \leq T \leq 627^\circ\text{C}$ were fit by Equation (4) yielding $A_s = 0.15269$ and $B_s = 1.3781 \times 10^{-4}$ (with $T = ^\circ\text{C}$ and $k = \text{w/cm-}^\circ\text{C}$). The maximum error in a calculated value is $\pm 0.004 \text{ w/cm-}^\circ\text{C}$ (including uncertainties in the data).

The effect of irradiation on the thermal conductivity of stainless steel has been considered and was taken to be negligible. In irradiation experiments performed at Oak Ridge National Laboratory, the in-pile thermal conductivity of types 316 and 310 stainless steels did not change after several months of

exposure in the Oak Ridge Reactor and after several hundred hours in the Low Intensity Testing Reactor*. These in-pile measurements were made at $250 \leq T \leq 830^\circ\text{C}$ and subsequent post-irradiation measurements were made at 150°C .

4. Radial Positions of Thermocouples and Pellet Diameters

The pellet diameters and the radial locations of all thermocouple holes were measured to within 0.001 inch prior to capsule assembly. In the Tests 2 through 5 the hot tips of the fuel thermocouples were wrapped with 0.008-inch wire in the region of the prime pellet. The wire served as a shim, centering the 0.065-inch holes.

No shims were used in Test 1. However, the radial positions of the thermocouples were measured during post-irradiation examination. In order to assure that the thermocouples did not shift during disassembly, the capsule internals were bonded with Hysol prior to the first cutting operation.

5. Ratio of Gamma Heating in the Fuel to Gamma Heating in Stainless Steel

The gamma-heating ratios were calculated using gamma-heating rates for the elements relative to that in stainless steel†. The relative gamma-heating rate for any alloy or compound is calculated by summing the products of the weight fraction and gamma rate for each elemental constituent. The ratios of gamma heating in the fuel to gamma heating in stainless steel, G , are given in Table 2.

6. Reciprocal Neutron Diffusion Length

Equation (13) represents the solution of the neutron diffusion equation in solid cylindrical geometry, assuming azimuthal and axial symmetry. In its explicit formulation Equation (13) would use a reciprocal neutron diffusion length calculated from the macroscopic cross sections (Reference 3):

$$K \approx \sqrt{3\Sigma_s \Sigma_a} \quad (24)$$

where Σ_s and Σ_a are the scattering and absorption cross sections, respectively.

Although Equation (13) is derived from first principles, it is limited by the assumptions inherent to the neutron diffusion approximation. Typically,

*Information obtained from J. T. Howe, Oak Ridge National Laboratory, August 1953.

†Information obtained from R. M. Lieberman, Bettis Atomic Power Laboratory, December 1965.

TABLE 2. SUMMARY OF FUEL CHARACTERISTICS

Nominal Composition:	ThO ₂ + 3.1 w/o UO ₂	ThO ₂ + 7.7 w/o UO ₂	ThO ₂ + 10 w/o UO ₂	ThO ₂ + 10 w/o UO ₂	UO ₂
Batch Number:	6274	5953	6735	6783	6959
Test Capsule:	Test 1, Middle	Test 1, Bottom	Test 2, Bottom and Middle, Test 3	Test 4	Test 5
Total Uranium, w/o	2.74	6.69	8.85	8.87	87.79
Total Thorium, w/o	84.59	80.55	79.41	79.36	-
Hexavalent Uranium, percent total Uranium	3.30	< 1.0	1.4	< 1.0	< 1.0
Molecular Mass Uran- ium, gm-atoms	235.252	236.227	237.23	237.36	237.80
Stoichiometry	Th(0.969)U(0.031) O(2.001)	Th(0.925)U(0.075) O(2.001)	Th(0.914)U(0.087) O(2.001)	Th(0.902)U(0.099) O(2.000)	U(1.000)O(2.000)
Composition	ThO ₂ + 3.12 w/o UO ₂	ThO ₂ + 7.66 w/o UO ₂	ThO ₂ + 10.04 w/o UO ₂	ThO ₂ + 10.07 w/o UO ₂	UO ₂
Impurities, ppm					
Al	55	70	< 50	76	< 50
B	< 0.25	< 0.25	< 0.25	< 0.25	< 0.25
C	134	117	36	36	< 20
Ca	< 25	< 25	< 25	< 25	< 10
Cd	< 0.25	0.25	< 0.25	< 0.25	< 0.25
Cr	< 10	10	< 15	< 1.5	< 15
Co	< 2.5	2.5	< 2.5	< 2.5	< 2.5
Cu	< 2.5	2.8	< 2.5	< 2.5	< 2.5
Fe	< 25	< 25	200	< 25	150
Gd	< 0.5	< 1	< 0.5	< 0.5	6.7
Hf	-	-	< 0.5	< 0.5	1.1
Mg	< 5	< 5	< 10	< 10	< 5
Mn	< 2.5	< 2.5	< 2.5	< 2.5	< 2.5
Mo	< 5	< 5	< 5	< 5	< 500
N	< 5	< 5	< 10	< 10	46
Ni	< 5	< 5	< 5	< 5	26
Pb	< 5	< 5	< 5	< 5	< 5
Si	< 50	< 50	< 50	< 50	< 50
Sn	< 5	< 5	< 5	< 5	< 5
Th	-	-	-	-	62
Ti	< 2.5	< 2.5	< 2.5	< 2.5	< 2.5
Zr	100	< 100	< 100	-	< 100

TABLE 2 (Cont)

Nominal Composition:	ThO ₂ + 3.1 w/o UO ₂	ThO ₂ + 7.7 w/o UO ₂	ThO ₂ + 10 w/o UO ₂	ThO ₂ + 10 w/o UO ₂	UO ₂
Batch Number:	6274	5935	6735	6783	6959
Test Capsule:	Test 1, Middle	Test 1, Bottom	Test 2, Bottom and Middle, Test 3	Test 4	Test 5
Lattice Constant, Å	a ₀ = 5.5942	a ₀ = 5.5872	a ₀ = 5.5861	a ₀ = 5.5857	a ₀ = 5.4704
ASTM Grain Size	6.80	7.14	7.1	8.9	9.81
Macroporosity	Lenticular and uniform	Prismatic and uniform	Spherical and uniform	Prismatic and uniform	Spherical and uniform
Microporosity	Spherical and uniform	Spherical and uniform	Spherical and uniform	Spherical and uniform	Spherical and uniform
General Metallographic Structure	Microcracks		Small white Fe inclusions		Inclusions of Fe and Mo
Photomicrographs	Figure 17	Figure 17	Figure 18	Figure 19	Figure 20
Distribution U ²³⁵	Uniform	Uniform	Uniform	Uniform	Uniform
Density (Percent Theoretical Density)					
a. Prime Pellet(s)	94.9	74.1	98.5, 98.2, 98.5	86.0	98.5
b. Unirradiated Control(s)	90.1	74.2	98.2, 98.3	85.8, 86.1	98.6
c. Post-Irradiation Sample	94.8	-	-	-	-
Height, inches*	0.514	0.512	0.528	0.526	0.480
Diameter, inches*	0.9951	0.9954	0.9950	1.0902	0.9953
Weight, gram*	59.61	46.54	62.98	56.53	63.25
Grams U ²³⁵ /gm U(Total)*	0.932	0.618	0.287	0.395	0.100
Grams U ²³⁵ /pellet*	1.52	1.92	1.60	1.98	5.55
Grams U ²³⁵ /inch of pellet*	2.96	3.76	3.03	3.76	11.57
Reciprocal Neutron Diffusion Length, cm ⁻¹ †	0.919	1.015	0.932	0.997	1.669
Gamma Heating Relative to Stainless Steel‡	1.80	1.43	1.77	1.56	2.09

*Average of pellets in capsule(s)

†Determined from Monte Carlo prediction

‡(w/cm³ fuel)/(w/cm³ stainless steel)

distributions predicted by diffusion theory are significantly different than those predicted by other techniques and/or different than distributions determined in post-irradiation investigations.

One method for obtaining better results is to adjust the value of K so that the desired distribution is obtained. That is, given a distribution in fission power density, an adjusted value, K_a , is calculated so that Equation (13) is forced to fit the given distribution. No restriction is placed on the source of the given distribution; it may be obtained using other computational techniques or by direct post-irradiation measurement.

Given the distribution in power density by fission, $Q_f(r)$, and using the series approximation to $I_0(K_a r)$, then:

$$K_a = \frac{2}{r} \left[Y(r) - 0.25000 Y^2(r) + 0.09723 Y^3(r) - 0.04514 Y^4(r) + 0.02308 Y^5(r) - 0.01254 Y^6(r) + \dots \right]^{1/2} \quad (25)$$

where:

$$Y(r) = \frac{Q(r)}{Q(0)} - 1. \quad (26)$$

(A derivation of Equation (25) is given in Appendix A.)

The values for the reciprocal neutron diffusion length used in the analytic equations were obtained by solving Equation (25) using the fission power distributions (by thermal and fast neutrons) predicted by Monte Carlo calculations. The validity of this technique is illustrated in Figure 5 where a measured distribution in fission power for the 94.9-percent dense $\text{ThO}_2 + 3.1 \text{ w/o } \text{UO}_2^*$ is compared to: (1) the Monte Carlo prediction, (2) the distribution fit to the Monte Carlo prediction, and (3) the distribution using $K = \sqrt{\Sigma_s \Sigma_a}$. The measured values were obtained by micro-drilling the prime pellet at selected radial positions. Radiochemical analyses of the drillings were then used to reconstruct the relative radial distribution. The large scatter in the data is the result of angular asymmetry in the neutron flux. However, the average of these data agree satisfactorily with the Monte Carlo prediction and the corresponding forced distribution.

IV. FUEL PELLETS

The fuel pellets were fabricated according to the processes outlined in Figures 6 through 11. The nominal design of the finished pellet is given in Figure 12.

*From middle capsule of Test 1.

For all fuel pellet batches the following determinations were made:

1. Isotopic composition
2. Chemical composition
3. Impurity content
4. Stoichiometry
5. Lattice constant
6. Metallographic structure
7. Uniformity in U^{235} distribution
8. Unirradiated thermal conductivity.

These determinations were accomplished using one or more representative samples from each batch. In addition the dimensions, weight, density, and location of thermocouple holes for each fuel pellet was recorded. The fuel characteristics are summarized in Table 2. Included are the values for the two fuel related analytic parameters, the gamma-heating ratio, G , and the adjusted reciprocal neutron-diffusion length, K_a . These were calculated according to the procedures outlined in Sections III.B.5 and III.B.6. The Monte Carlo predictions used in obtaining the values for K_a are shown in Figures 5 and 13 through 16.

The typical microstructure of each fuel is shown in Figures 17 through 20. Note the white inclusions in the 98.5-percent dense $ThO_2 + 10$ w/o UO_2 and the UO_2 specimens (Figures 18 and 20, respectively). These have been identified by electron-microprobe analysis as inclusions of iron in the $ThO_2 + 10$ w/o UO_2 specimen and iron and molybdenum in the UO_2 specimen.

The unirradiated thermal conductivity data for representative samples of fuel are given in Tables 3 through 7. With the exception of the $ThO_2 + 3.1$ w/o UO_2 specimen, these samples had densities comparable to those of their corresponding prime in-pile samples. This arrangement obviated the need for density corrections when the in-pile and unirradiated data were compared.

The unirradiated thermal conductivities were determined at Battelle Memorial Institute (BMI) at their Columbus Laboratories using the pulse-diffusivity technique (Reference 4). Briefly, this technique involves the independent determination of thermal diffusivity, specific heat, and density, each as a function of temperature. These quantities are then used to calculate thermal conductivity. With the exception of UO_2 , the values for specific heat were calculated from enthalpy measurements made in a Bunsen ice calorimeter. The specific heat data for the UO_2 specimens were taken from the literature

TABLE 3. UNIRRADIATED THERMAL CONDUCTIVITY, $\text{ThO}_2 + 3.1 \text{ w/o } \text{UO}_2$

Batch 6274, 90.1-Percent Dense

<u>Temperature, °C</u>	<u>Thermal Conductivity, w/cm-°C</u>
300	.0496
430	.0474
488	.0387
548	.0409
656	.0319
912	.0284
934	.0267
1049	.0246
1160	.0252
1336	.0224
1432	.0216
1605	.0200
1750	.0184
1880	.0178

TABLE 4. UNIRRADIATED THERMAL CONDUCTIVITY, $\text{ThO}_2 + 7.7 \text{ w/o } \text{UO}_2$

Batch 5953

<u>Sample 5953-4, 71.2-Percent Dense</u>		<u>Sample 5953-7, 74.2-Percent Dense</u>	
<u>Temperature, °C</u>	<u>Thermal Conductivity, w/cm-°C</u>	<u>Temperature, °C</u>	<u>Thermal Conductivity, w/cm-°C</u>
280	.0373	304	.0260
283	.0339	312	.0299
295	.0390	553	.0221
496	.0283	806	.0180
522	.0282	860	.0158
611	.0260	1094	.0151
705	.0225	1177	.0128
757	.0235	1435	.0123
858	.0198	1454	.0128
940	.0187	1654	.0110
1007	.0185	1849	.0109
1018	.0171		
1020	.0182		
1230	.0161		
1255	.0148		
1285	.0152		
1332	.0152		
1490	.0137		
1520	.0147		
1548	.0139		
1675	.0143		
1699	.0132		
1805	.0125		
1890	.0123		

TABLE 5. UNIRRADIATED THERMAL CONDUCTIVITY, $\text{ThO}_2 + 10 \text{ w/o UO}_2$

Batch 6735

Sample 6735-4, 98.2-Percent Dense		Sample 6735-5, 98.3-Percent Dense	
Temperature, °C	Thermal Conductivity, w/cm-°C	Temperature, °C	Thermal Conductivity, w/cm-°C
310	.0704	101	.0837
432	.0548	305	.0641
527	.0541	407	.0551
601	.0471	563	.0479
711	.0445	807	.0365
804	.0382	903	.0343
954	.0344	1037	.0296
1048	.0291	1300	.0261
1095	.0317	1490	.0240
1222	.0253	1565	.0220
1309	.0299	1798	.0195
1417	.0224		
1523	.0253		
1614	.0199		
1687	.0202		
1794	.0170		
1846	.0175		

TABLE 6. UNIRRADIATED THERMAL CONDUCTIVITY, $\text{ThO}_2 + 10 \text{ w/o UO}_2$

Batch 6783

Sample 6783-29, 85.8-Percent Dense		Sample 6783-30, 86.1-Percent Dense	
Temperature, °C	Thermal Conductivity, w/cm-°C	Temperature, °C	Thermal Conductivity, w/cm-°C
292	.0559	225	.0569
384	.0426	242	.0537
549	.0374	423	.0429
680	.0324	424	.0438
924	.0259	622	.0359
1106	.0244	831	.0296
1123	.0227	844	.0286
1222	.0197	1013	.0263
1287	.0226	1052	.0244
1340	.0204	1227	.0220
1414	.0186	1435	.0194
1462	.0190	1642	.0173
1540	.0178	1666	.0174
1610	.0171	1827	.0151
1702	.0151		
1729	.0166		
1781	.0156		
1864	.0148		

TABLE 7. UNIRRADIATED THERMAL CONDUCTIVITY, UO_2

Batch 6959, 98.6-Percent Dense

<u>Temperature, °C</u>	<u>Thermal Conductivity, w/cm-°C</u>
147	.0596
161	.0694
225	.0635
286	.0568
368	.0611
526	.0470
592	.0436
648	.0448
780	.0367
798	.0384
882	.0349
889	.0353
1030	.0305
1206	.0274
1209	.0271
1353	.0237
1391	.0234
1463	.0232
1511	.0213
1614	.0188
1715	.0194
1796	.0193
1879	.0172

(Reference 5). The densities were calculated from linear thermal expansion data measured with a recording dilatometer. The diffusivities were measured parallel to the specimen axis using a laser flash apparatus. The values for thermal conductivity are estimated by Battelle to be accurate to within ± 6 percent.

V. TEST OPERATION

A. Test Facility and Operational Procedures

All irradiations were conducted in the ETR. The location of each test within the facility is shown in Figure 21.

The capsules were cooled by approximately 45°C water flowing at approximately 35 gpm. Gross temperature control was achieved with the use of hafnium shrouds placed in adjacent reflector positions.

During steady-state operation, with the reactor at full power, data were collected on a once-daily basis. In addition a number of specially programmed

reactor startups and power reductions were utilized to obtain data at low temperatures. Specifically, the reactor was held at reduced-power levels for sufficient time to allow thermal equilibration and data collection; the holds typically lasted for 1/2 to 1 hour. For each test cycle these special holds normally occurred during the initial startup, 24 and 48 hours thereafter, and at the final shutdown. It is important to note that since these special holds were the only source of low-temperature data, these data were, of necessity, always obtained after irradiation at higher temperatures.

The thermocouple data from each test were collected using digital voltmeters and/or multi-point recorders. With the exception of Test 1 the thermocouples terminated at a constant temperature reference junction held at 150F.

B. Operational Histories

The operational histories of the tests are summarized in Table 8. In general, all tests operated successfully and according to expectation. However, there were a few notable exceptions:

1. Five of the six fuel thermocouples in the bottom capsule of Test 1 (containing 74-percent dense $\text{ThO}_2 + 7.7 \text{ w/o } \text{UO}_2$) were found to be nonoperative at the time of initial startup. Consequently, there were no in-pile conductivity data from this capsule.
2. A peripheral fuel thermocouple in the middle capsule of Test 1 (containing 95-percent dense $\text{ThO}_2 + 3.1 \text{ w/o } \text{UO}_2$) showed an open circuit at the time of the initial startup. Therefore, the data from this capsule were reduced to yield mean thermal conductivities, averaged between the thermocouples at the center and at the intermediate radius, (see Equations (21a) and (21b)).
3. The fuel thermocouples in Test 4 (containing 86-percent dense $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$) failed prematurely due to excessive temperatures ($>1800\text{C}$) experienced as the test approached full power during ETR cycle 90B. However, data were obtained during cycle 90A and during the low-power holds in the 90B startup.

The acquisition of data relative to thermal conductivity ceased when there was any indication of an in-service thermocouple failure. The termination of each test, defined as the time of thermocouple failure, did not necessarily correspond to the time the test was withdrawn from the facility. In fact, the irradiations typically continued for several cycles beyond the instance of thermocouple failure. The exception to this was Test 1 which was withdrawn immediately after termination.

VI. TEST RESULTS

A. Post-Irradiation Thermal Conductivity

The post-irradiation thermal conductivity was determined for the $\text{ThO}_2 + 3.1 \text{ w/o } \text{UO}_2$ fuel material. The measurements were made at BMI using the pulse-diffusivity technique. The sample used for these measurements was a disk approximately 0.83 centimeter in diameter, cut from a fuel pellet which was irradiated in the middle capsule of Test 1. This pellet was from the same fabrication batch as the prime in-pile pellet and was of comparable density (94.8-percent dense versus 94.9-percent dense). The sample disk was ultrasonically cut from a crack-free region of the fuel pellet. The center of the disk was approximately 0.73 centimeter from the center of the pellet. The arithmetic mean in-pile temperature of the disk was about 900C while the entire pellet operated at peak conditions during ETR cycle 88A.

Test 1 was withdrawn from the facility immediately after thermocouple failures precluding any further acquisition of data. Therefore, the depletions corresponding to the post-irradiation and in-pile thermal conductivities are essentially the same.

The values for the post-irradiation thermal conductivity are given in Table 9. During the measurements the heating and cooling rates were approximately 10C/min; the sample was held at each measuring temperature for about 1 minute. It should be noted that the post-test specific heat was not measured, and unirradiated values were used in calculating the post-irradiation thermal conductivity.

B. In-Pile Thermal Conductivity

Figures 22 through 24 show the in-pile and unirradiated data points for the 98-percent dense $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, 86-percent dense $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, and 98-percent dense UO_2 , respectively. In each of these three sets of data the prime in-pile sample and the unirradiated sample were of comparable density, and therefore, the data points do not include any corrections for density.

The in-pile and post-irradiation data points for the 95-percent dense $\text{ThO}_2 + 3.1 \text{ w/o } \text{UO}_2$ are given in Figure 25. The in-pile data are mean conductivity values averaged between the center and intermediate thermocouples according to Equations (21a) and (21b). Figure 25 does not include the unirradiated data for the $\text{ThO}_2 + 3.1 \text{ w/o } \text{UO}_2$ specimens. These data were excluded because of the significant disparity in the densities of the irradiated and unirradiated samples (90.1- versus 94.8- and 94.9-percent dense).

TABLE 8. SUMMARY OF TEST OPERATION

Test Train Capsule	TEST 1						TEST 2						TEST 3	TEST 4		TEST 5
	Middle			Bottom			Middle	Bottom				TEST 4				
Fuel composition	ThO ₂ + 3.1 w/o UO ₂			ThO ₂ + 7.7 w/o UO ₂			ThO ₂ + 10 w/o UO ₂	ThO ₂ + 10 w/o UO ₂				ThO ₂ + 10 w/o UO ₂	ThO ₂ + 10 w/o UO ₂		UO ₂	
Fuel density, percent T.D.	94.9			74.1			98.5	98.2				98.5	86.0		98.5	
Type fuel thermocouples	Tungsten-Rhenium			Tungsten-Rhenium			Chromel-Alumel	Chromel-Alumel				Tungsten-Rhenium	Tungsten-Rhenium		Tungsten-Rhenium	
ETR cycle	87A	87B	88A	87A	87B	88A	89	89	90	91	92	89	90	90	91	
ETR reflector position	A-10	A-10	B-10	A-10	A-10	B-10	B-10	B-10	B-10	B-10	B-10	E-16	C-12	C-12	I-2	
Axial gamma-heat factor, F _a	1.03	1.03	1.03	0.66	0.66	0.66	1.03	0.66	0.66	0.66	0.66	1.52	1.49	1.49	1.21	
Power density by fission heating, w/cm ³ *	15	34	54	17	36	55	66	55	54	50	55	76	50	73	75	
Power density by gamma heating, w/cm ³ *	5	6	11	3	4	8	11	7	6	6	7	19	17	19	12	
Center temperature, °C*	400	800	1200	400	800	1200	1000	600	600	500	600	1600	1400	--	1200	
Date of initial startup	2/20/67			2/20/67			5/8/67	5/8/67				5/19/67	6/24/67		8/20/67	
Date of termination†	4/2/67			2/20/67			6/4/67	11/26/67				5/24/67	6/30/67		10/1/67	
Duration (EFPH at termination)	380.5			--			525	2454				117	96		658	
Burnup at termination, fission/cm ³	2.3 x 10 ¹⁸			--			3.7 x 10 ¹⁸	13.9 x 10 ¹⁸				1.0 x 10 ¹⁸	0.7 x 10 ¹⁸		5.6 x 10 ¹⁸	
Mean internal neutron flux, thermal, n/cm ²	6.9 x 10 ¹⁸			--			10.8 x 10 ¹⁸	40.4 x 10 ¹⁸				2.9 x 10 ¹⁸	1.7 x 10 ¹⁸		4.7 x 10 ¹⁸	
Mean internal neutron flux, > 1 Mev, n/cm ²	3.7 x 10 ¹⁸			--			5.8 x 10 ¹⁸	21.7 x 10 ¹⁸				1.6 x 10 ¹⁸	0.9 x 10 ¹⁸		2.5 x 10 ¹⁸	
Type of fuel thermocouple failure	Mechanical (thermocouples were severely bent. One was inoperative at installation; others failed during service. These showed open circuits or indicated anomalously low temperatures).			Mechanical (thermocouples were severely bent. Inoperative at installation showing open circuits or indicating anomalously low temperatures).			Excessive temperature (thermocouples drifted and then opened after 1100 to 1200C occurred due to control rod motion at end of cycle).	Failure of epoxy plug which seals thermocouples from water. Thermocouples were still operative at time of plug failure.				Uncertain (signal from center thermocouple began to drift steadily downward).	Excessive temperature (thermocouples failed during cycle 90B increase in reactor power when each thermocouple temperature exceeded 1600 to 1800C).		Uncertain (signal from center thermocouple began to drift steadily downward during initial phase of cycle 92). Thermocouple was operative at cycle 91 shutdown.	

*Normal order of magnitude values at full reactor power.

†The date of termination refers to the date at which thermocouple failures precluded any further analysis of data relative to thermal conductivity.

TABLE 9. POST-IRRADIATION THERMAL CONDUCTIVITY, $\text{ThO}_2 + 3.1 \text{ w/o } \text{UO}_2$ Batch 6274, 94.8-Percent Dense, Burnup about 2.3×10^{18} fission/cm³

<u>Temperature, °C.</u>	<u>Thermal Conductivity,</u> <u>w/cm-°C</u>
Heating	
320	.0488
410	.0440
525	.0406
600	.0379
700	.0375
845	.0344
895	.0316
1135	.0273
Cooling	
380	.0465
450	.0415
660	.0367
1010	.0293
1135	.0286

The accuracy of each set of in-pile conductivity values from a given test capsule is estimated to be within ± 19 percent. This includes the uncertainties related to the analytic parameters and assumptive conditions discussed in Section III, that is, the uncertainties due to (a) gamma heating rate ($\pm 3\%$), (b) thermal conductivity of stainless steel ($\pm 5\%$), (c) positions of the thermocouples ($\pm 11\%$), (d) the reciprocal neutron diffusion length ($\pm 6\%$), and (e) the asymmetry in neutron flux ($\pm 14\%$). Thus, with the exception of the 98-percent dense $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, the accuracy of the in-pile data for each material taken by itself is within ± 19 percent. In the case of the 98-percent $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, which was irradiated in three test capsules, the accuracy is estimated to be within ± 11.5 percent.

The in-pile data points shown in Figures 22 through 25 were obtained either during a low-power hold or during steady-state operation. The low-power holds were the only source of data at the lower temperatures. Since these holds occurred intermittently during the irradiation of each test, the lower tempera-

ture data were obtained after irradiation at the higher temperatures*. Consequently, there is, for each fuel material, a temperature below which all in-pile conductivity values were acquired on a transient basis. These are as follows:

- i) 95-percent dense $\text{ThO}_2 + 3.1 \text{ w/o } \text{UO}_2$, 440C
- ii) 98-percent dense $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, 400C
- iii) 86-percent dense $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, 600C
- iv) 98-percent dense UO_2 , 600C.

VII. ANALYSIS OF DATA

A. General

In the analyses to follow all data are fit to the linear model:

$$1/k(T) = A + BT. \quad (28)$$

Theoretically, this formulation describes the lattice (phonon) component to the thermal resistivity for materials above the Debye temperature. The constant term corresponds to the contribution made by lattice imperfections, and the second term defines the contribution made by phonon-phonon interactions (Umklapp process). However, the linear model excludes, *a priori*, any contributions made by the radiative and electronic components. Therefore, no theoretical significance should be placed on the values taken by either A or B.

B. Comparison of In-Pile and Unirradiated Thermal Conductivities

Figures 26, 27, and 28 show the regression curves through the in-pile and unirradiated data for the 98-percent dense $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, the 86-percent dense $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, and the 98-percent dense UO_2 , respectively. These curves were obtained by fitting the linear model to all data taken over the entire range of in-pile experimental temperatures. These figures have the following two features in common:

1. From the lowest experimental temperatures to about 1000C there appears, by inspection, to be no significant difference between the in-pile and unirradiated thermal conductivity data.
2. At experimental temperatures greater than about 1000C, the in-pile conductivity data are significantly greater than the unirradiated data, and the disparity increases with increasing temperature.

The low-temperature result given above was verified by statistical analysis. Only the data for $T \leq 1000\text{C}$ were fit to the linear model; the unirradiated and

*The only exception to this is the data collected at low-power holds during the initial startup.

in-pile regression constants (the intercept A and slope B) were then compared using procedures outlined by Wald (Reference 6). The analyses for the three fuels are summarized in Table 10. The analyses show that, with 99-percent confidence, there are no statistically significant differences between the in-pile and unirradiated thermal conductivity data from the lowest experimental temperatures to 1000C. This result for $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$ explicitly includes data at as low a temperature as 230C. However, because of the intermittent acquisition of the low-temperature data (Sections V.A and VI.B), the result cannot be applied to steady-state operation below 400C. Similarly, the lower steady-state temperature for the UO_2 is 600C.

In the above statistical comparisons the only uncertainty considered was that due to the random variations in the measured data points, that is, the experimental uncertainties were not included. This technique is considered preferable when comparing two sets of data since the limits of uncertainty about each set is reduced to a minimum, and any conclusion regarding the similarity of the sets are the most conservative. It is noteworthy that utilization of this conservative technique yielded similar results in all cases. This suggests that the experimental uncertainties related to accuracy can be reduced by assuming that the same result was obtained in six separate experiments. On this basis the accuracy of the in-pile data for each material is estimated to be ± 7 percent.

C. Comparison of In-Pile and Post-Irradiation Thermal Conductivities

The regression curves through the in-pile and post-irradiation data for the 95-percent dense $\text{ThO}_2 + 3.1 \text{ w/o } \text{UO}_2$ are shown in Figure 29. The statistical comparison of the regression constants is summarized in Table 10. The analysis shows that, with 99-percent confidence, there is no statistically significant difference between the in-pile and post-irradiation thermal conductivity data for $440 \leq T \leq 1110\text{C}$.

D. Comparison of In-Pile Thermal Conductivities at Various Burnups

The bottom capsule from Test 2 (containing 98.2-percent dense $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$) was the only capsule to supply useable data beyond the first irradiation cycle. The capsule operated for three complete cycles, achieving a total burnup of 1.39×10^{19} fission/cm³. The data corresponding to various stages of depletion were fit to the linear model. The regression curves are compared in Figure 30. All curves lie within the 99-percent confidence band about the 0.8×10^{18} fission/cm³ curve implying that there is no statistically significant difference between the earliest and all subsequent sets of data. On the basis of this

result it is believed that the data from the other test capsules can be confidently extrapolated to at least approximately 10^{19} fission/cm³.

VIII. SUMMARY AND DISCUSSION OF RESULTS

Considering the analyses from the preceding section and the lower experimental temperatures for steady-state operation, the test results can be summarized as follows:

1. During steady-state operation at full power there was no statistically significant difference, with 99-percent confidence, between the in-pile and unirradiated thermal conductivity data for the UO_2 at $600 \leq T \leq 1000\text{C}$ to a burnup of up to 5.6×10^{18} fission/cm³ and for the $\text{ThO}_2 + 10$ w/o UO_2 at $400 \leq T \leq 1000\text{C}$ to a burnup of up to 13.9×10^{18} fission/cm³.
2. Including those data obtained during short (30- to 60-minutes) intermittent periods of operation at low power, there was no statistically significant difference, with 99-percent confidence, between the in-pile and unirradiated thermal conductivity data for UO_2 at $500 \leq T \leq 1000\text{C}$ and for $\text{ThO}_2 + 10$ w/o UO_2 at $230 \leq T \leq 1000\text{C}$.
3. The in-pile thermal conductivities were slightly greater than the unirradiated values by a statistically significant amount at $1000 \leq T \leq 1300\text{C}$ for UO_2 and at $1000 \leq T \leq 1700\text{C}$ for the $\text{ThO}_2 + 10$ w/o UO_2 .
4. There was no statistically significant difference, with 99-percent confidence, between the in-pile and post-irradiation thermal conductivities for $\text{ThO}_2 + 3.1$ w/o UO_2 during steady-state operation at $440 \leq T \leq 1200\text{C}$ to a burnup of up to 2.3×10^{18} fission/cm³. The post-irradiation sample operated in-pile at an arithmetic mean temperature of about 900C. As such, the post-test data are considered to be equivalent to the unirradiated thermal conductivity.

It should be noted that the levels of depletion achieved in these tests are an order of magnitude less than that required for any significant accumulation of fission products and the formation of fission gas bubbles. Therefore, the test results do not reflect any effects by these phenomena.

The steady-state UO_2 results from this test are consistent with those obtained by Clough and Sayers (Reference 7). Using a radial heat-flow technique in conjunction with in-pile annealing studies, Clough and Sayers showed, by direct measurement, that radiation effects in UO_2 are only significant at temperatures less than 525C at a burnup of 4.3×10^{19} fission/cm³; at $525 \leq T \leq 1200\text{C}$ they observed no significant difference between the in-pile

TABLE 10. SUMMARY OF STATISTICAL COMPARISONS OF LINEAR REGRESSION EQUATIONS FOR $T \leq 1000^\circ\text{C}$

Model: $1/k(T) = \hat{Y}_i = A_i + B_i T$, $T = ^\circ\text{C}$, $k = \text{w/cm-}^\circ\text{C}$

Fuel Composition Density (Percent Theoretical Density)	ThO ₂ + 3.1 w/o UO ₂ 94.8 to 94.9		ThO ₂ + 10 w/o UO ₂ 98.2 to 98.5		ThO ₂ + 10 w/o UO ₂ 85.8 to 86.1		UO ₂ 98.5 to 98.6	
	In-Pile	Post-Irradiation	Unirradiated	In-Pile (227 to 1000C)	Unirradiated (549 to 1052C)	In-Pile (481 to 997C)	Unirradiated (526 to 1030C)	In-Pile (423 to 996C)
i	1	2	1	2	1	2	1	2
N _i = Number of observations	92	13	15	255	8	40	8	61
A _i = Intercept, (w/cm-°C) ⁻¹	13.785	14.930	7.3738	7.5620	11.733	14.377	9.0163	11.588
B _i = Slope, (w/cm) ⁻¹	2.03344 x 10 ⁻²	1.83230 x 10 ⁻²	2.38151 x 10 ⁻²	2.29686 x 10 ⁻²	2.72397 x 10 ⁻²	2.57541 x 10 ⁻²	2.23696 x 10 ⁻²	1.84783 x 10 ⁻²
$\bar{X}_i$ = Mean temperature, °C	680.75	697.31	700.27	607.50	816.62	810.87	768.12	800.05
$\bar{Y}_i$ = Mean resistivity, (w/cm-°C) ⁻¹	27.628	27.707	24.051	21.515	33.978	35.253	26.199	26.371
S _i ² = Variance in $\hat{Y}_i$	2.2578	0.58260	1.6146	2.7079	1.3227	0.8186	0.71420	1.0492
S _{xi} ² = Sum squares in $\bar{X}_i$	1.03720 x 10 ⁶	9.69131 x 10 ⁵	1.04310 x 10 ⁶	8.01163 x 10 ⁶	2.42012 x 10 ⁵	7.33774 x 10 ⁵	2.01265 x 10 ⁵	1.27770 x 10 ⁶
D = Degrees of freedom	90		266		44		65	
F ₉₉ (N ₁ - 2, N ₂ - 2) = F statistic	3.70		2.17		3.32		7.05	
t ₉₉ (D) = T statistic	2.58		2.58		2.58		2.58	
F _{calc} = F statistic for variances	3.88		1.68		1.62		1.47	
Conclusion	S ₂ ² < S ₁ ² , (0.99 confidence)		S ₁ ² ≈ S ₂ ² , (0.99 confidence)		S ₁ ² ≈ S ₂ ² , (0.99 confidence)		S ₁ ² ≈ S ₂ ² , (0.99 confidence)	
t _B = T statistic for slope	1.207		0.50		0.673		1.61	
Conclusion	B ₁ ≈ B ₂ , (0.99 confidence, that is, $\hat{Y}_1/\hat{Y}_2$)		B ₁ ≈ B ₂ , (0.99 confidence, that is, $\hat{Y}_1/\hat{Y}_2$)		B ₁ ≈ B ₂ , (0.99 confidence, that is, $\hat{Y}_1/\hat{Y}_2$)		B ₁ ≈ B ₂ , 0.99 confidence, that is, $\hat{Y}_1/\hat{Y}_2$)	
t _A = T statistic for intercept	0.879		0.47		0.391		0.24	
Conclusion	A ₁ ≈ A ₂ , (0.99 confidence)		A ₁ ≈ A ₂ , (0.99 confidence)		A ₁ ≈ A ₂ , (0.99 confidence)		A ₁ ≈ A ₂ , (0.99 confidence)	
Final conclusion	$\hat{Y}_1 \approx \hat{Y}_2$, (0.99 confidence)		$\hat{Y}_1 \approx \hat{Y}_2$, (0.99 confidence), 305 ≤ T ≤ 1000C		$\hat{Y}_1 \approx \hat{Y}_2$, (0.99 confidence), 549 ≤ T ≤ 997C		$\hat{Y}_1 \approx \hat{Y}_2$, (0.99 confidence), 526 ≤ T ≤ 996C	

and unirradiated thermal conductivities. In addition, Clough and Sayers noted that their in-pile values were slightly greater than the unirradiated values, at temperatures greater than 1350C.

In experiments with $\text{ThO}_2 + 1.3 \text{ w/o } \text{UO}_2$, MacEwan and Stoute observed 30- to 50-percent reductions in thermal conductivity after irradiation to a burnup of 3.1×10^{14} to 4.7×10^{17} fission/cm³ at $T < 150\text{C}$ (Reference 8). The conductivities were measured on a post-irradiation basis at 60C. Subsequent isochronal annealing between 300 and 1200C showed a broad recovery stage between 500 and 900C.

Cursorily, MacEwan and Stoute's results conflict with the $\text{ThO}_2\text{-UO}_2$ data from the present tests which show no radiation damage at $T > 400\text{C}$. However, the applicability of post-irradiation annealing studies to in-pile operation is questionable, particularly at these relatively low temperatures where the kinetics of recovery in-pile are undoubtedly influenced by radiation enhanced defect diffusion. This is exemplified by considering similar conflicts in the UO_2 literature. Daniel reported an approximately 50-percent reduction in the conductivity of UO_2 irradiated at $T < 100\text{C}$ to a burnup of approximately 10^{19} fission/cm³. Post-irradiation annealing showed recovery stages at 150C, 300C, and 800C with only about 90-percent recovery at 1000C (Reference 9). Similar results were reported by Ross (Reference 10). In contrast, Clough and Sayers, using encapsulated heaters for in-pile annealing studies, showed full recovery of thermal conductivity in-pile at about 525C (Reference 7). These dissimilar results illustrate the nonrelevancy of post-irradiation annealing studies with respect to in-pile operation. Accordingly, the $\text{ThO}_2 + 1.3 \text{ w/o } \text{UO}_2$ data by MacEwan and Stoute are not considered germane to the in-pile data from the Bettis tests.

In Figure 31, the in-pile and unirradiated thermal conductivities for the UO_2 specimens from this test are compared directly to selected data from the literature. In those cases where the literature values were not related to 98.5-percent dense material, the data were adjusted accordingly by the Maxwell-Eucken formulation given by Belle (Reference 11). The literature data include unirradiated values by: Feith using the radial heat-flow method (Reference 12), Ainscough and Wheeler using the pulse-diffusivity technique (Reference 13), and Vogt using the radial heat-flow technique (Reference 14). Also included are the in-pile values by Clough and Sayers taken over the temperature range of interest (Reference 7).

The in-pile results from Test 5 agree quite well with the literature values over the entire range of experimental temperatures. In contrast, the Bettis unirradiated data deviate from the literature values at $T > 1000\text{C}$ in a manner similar to their deviation from the in-pile test results. It is not clear whether this similarity is fortuitous, or whether it represents a bias in the unirradiated values.

Although the possibility of a bias cannot be discounted, it is informative to consider some other explanations for the disparity between the in-pile and unirradiated test data at $T > 1000\text{C}$. Since the measurements at elevated temperature were, of necessity, restricted to the central regions of the fuel pellets, the disparities could possibly be the manifestation of physical phenomena in this region of the fuel. The prime candidates for consideration are columnar grain growth, gaseous conduction across the pores, and shifts in stoichiometry.

The possibility of columnar grain growth as a cause can be discounted in light of the times and temperatures involved. Test 5 containing UO_2 , operated at 1200 to 1300C for 658 effective full power hours. Tests 3 and 4 containing $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$ (98.5- and 86-percent dense, respectively) operated at 1600 to 1700C for about 100 effective full power hours. These time-temperature combinations are well below the minimums required for columnar grain growth as given by Nichols (Reference 15).

The effects of gaseous (hydrogen) conduction across the pores have also been considered and discounted. At the onset, it is uncertain whether there is any gas in the pores at all; although the pellets were sintered in a hydrogen atmosphere, they were subsequently vacuum annealed at 800C for about 4 hours. However, assuming the presence of gas, calculations indicate that its contribution to the bulk conductivity would be insignificant at the temperatures involved. In addition, if gaseous conduction were the sole contributor, there would be a significant difference in its effect on the high- and low-density materials. No such difference was observed.

The possibility of a shift in stoichiometry as the cause of the larger conductivity values in-pile at $T > 1000\text{C}$ cannot readily be discounted. There is considerable experimental evidence in the literature which shows that the thermal conductivity of UO_2 is extremely sensitive to stoichiometry; small increases in the oxygen-uranium ratio at $T < 900\text{C}$ cause significant reductions in conductivity; whereas, reductions in the oxygen-uranium ratio result in increases in conductivity at $T > 900\text{C}$ (Reference 14). Thus, there is little doubt that an increase in conductivity would result, if a reduction in the

oxygen-metal ratio were to occur at the center of the fuel. Of interest, therefore, is the mechanism for this shift in stoichiometry.

Fryxell has shown experimentally that thermal gradients can provide the potential for shifts in composition in uranium dioxide (Reference 16). In his experiments, Fryxell established gradients in stoichiometry in unirradiated UO_{2-x} by subjecting the material to thermal gradients of 120 to 200C/in. with peak temperatures of 2200 to 2500C. Under these conditions, the oxygen-uranium ratio was reduced at the hotter regions whereas the colder portions became stoichiometric. However, it should be noted that Fryxell could not reproduce this effect with initially stoichiometric UO_2 . (It is noteworthy that MacEwan and Stoute observed what they considered to be an irradiation-induced enhancement in thermal conductivity of UO_{2-x} at $T > 1000C$ [Reference 17]. In light of Fryxell's work, this may well have been the manifestation of a reduction in the oxygen-uranium ratio at the center of their samples).

In comparison to Fryxell's experimental parameters, the operating thermal gradients during irradiation were 1600 to 2400C/in., and the peak temperatures were 1300 to 1700C. It is uncertain whether these conditions could, in fact, induce a compositional shift in initially stoichiometric material. However, in considering the possibilities, one cannot exclude the effects of radiation and fissioning on the diffusion processes involved.

Clearly, any reduction in the oxygen-metal ratio at the center of the fuel would result in a corresponding increase in the oxygen-metal ratio at the cooler periphery with a subsequent reduction in conductivity in this region. However, this effect would be masked somewhat in the Bettis tests, since all measurements were restricted to radial positions, ≤ 0.7 of the full pellet radius. However, in considering the total fuel pellet, if shifts in stoichiometry did occur, the increase in conductivity at the center was balanced by reductions at the periphery since no significant changes in the center temperature were observed.

In conclusion therefore, there are two possible explanations for the increase in conductivity in-pile at $T > 1000C$ as observed in these tests; it is either a bias in the data or a manifestation of a shift in stoichiometry to a lower oxygen-metal ratio in the center of the fuel. If the latter case be true, there would be a reduction in conductivity at the periphery of an in-service fuel rod due to an increasing oxygen-metal ratio. It is assumed that this would not result in any significant change in the center temperature. Therefore, in either case, the observed increase in conductivity in-pile at $T > 1000C$ cannot be considered a meaningful enhancement.

Figure 1. Capsule for In-Pile Thermal Conductivity Studies

Figure 2. Assembled Test Train

Figure 3a. Radial Isotherm Contours in Fuel with Thermocouples

Figure 3b. Radial Isotherm Contours in Fuel without Thermocouples

Figure 4. Thermal Conductivity of Type 304 Stainless Steel

Figure 5. Distribution in Fission Power Density, $\text{ThO}_2 + 3.1 \text{ w/o } \text{UO}_2$, 94.9-Percent Dense, U^{235} Enrichment: 93.2 Percent

Figure 6. Powder Preparation and Pellet Fabrication,
ThO₂ + 3 w/o UO₂, Batch 6274

Figure 7. Powder Preparation and Pellet Fabrication,
 $\text{ThO}_2 + 8 \text{ w/o } \text{UO}_2$, Batch 5953

Figure 8. Powder Preparation, ThO₂ + 10 w/o UO₂ Fuels

Figure 9. Pellet Fabrication, ThO₂ + 10 w/o UO₂, Batch 6735

Figure 10. Pellet Fabrication, $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, Batch 6783

Figure 11. Powder Preparation and Pellet Fabrication,
100-Percent UO_2 , Batch 6959

Figure 12. Fuel Pellet for In-Pile Thermal Conductivity Test

Figure 13. Distribution in Fission Power Density,
ThO₂ + 7.7 w/o UO₂, 74.1-Percent Dense, U²³⁵
Enrichment: 61.8 Percent

Figure 14. Distribution in Fission Power Density,
 $\text{ThO}_2 + 10 \text{ w/o UO}_2$, 98.5-Percent Dense, U^{235}
 Enrichment: 28.7 Percent

Figure 15. Distribution in Fission Power Density,
 $\text{ThO}_2 + 10 \text{ w/o UO}_2$, 86-Percent Dense, U^{235}
 Enrichment: 39.5 Percent

Figure 16. Distribution in Fission Power Density,
 UO_2 , 98.5-Percent Dense, U^{235} Enrichment:
 10.0 Percent

NOTE:
 BATCH 6274
 94.9 - PERCENT DENSE
 AS POLISHED
 LONGITUDINAL

NOTE:
 BATCH 5953
 74.1 - PERCENT DENSE
 AS POLISHED
 TRANSVERSE

Figure 17. Top: Photomicrograph of $\text{ThO}_2 + 3.1 \text{ w/o UO}_2$
 Bottom: Photomicrograph of $\text{ThO}_2 + 7.7 \text{ w/o UO}_2$

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NOTE:

BATCH 6735
98.5-PERCENT DENSE
AS POLISHED
LONGITUDINAL

100 X

NOTE:

BATCH 6735
98.5-PERCENT DENSE
AS POLISHED
LONGITUDINAL

400 X

Figure 18. Photomicrographs of 98.5-Percent Dense $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$

EP R758-19

NOTE:

BATCH 6783
86-PERCENT DENSE
AS POLISHED
TRANSVERSE

100 X

NOTE:

BATCH 6783
86-PERCENT DENSE
AS POLISHED
TRANSVERSE

400 X

Figure 19. Photomicrographs of 86-Percent Dense ThO_2 + 10 w/o UO_2

EP R758-20

NOTE:
BATCH 6959
98.5-PERCENT DENSE
AS POLISHED
LONGITUDINAL

150 X

NOTE:
BATCH 6959
98.5-PERCENT DENSE
AS POLISHED
LONGITUDINAL

500 X

Figure 20. Photomicrographs of 98.5-Percent Dense UO_2

Figure 21. Locations of Bettis Tests in the ETR Reflector

Figure 22. In-Pile and Unirradiated Thermal Conductivity Data, $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, 98.2- to 98.5-Percent Dense

Figure 23. In-Pile and Unirradiated Thermal Conductivity Data, $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, 85.8- to 86.1-Percent Dense

Figure 24. In-Pile and Unirradiated Thermal Conductivity Data,
 UO_2 , 98.5- to 98.6-Percent Dense

Figure 25. In-Pile and Post-Irradiation Thermal Conductivity Data,
 $\text{ThO}_2 + 3.1 \text{ w/o } \text{UO}_2$, 94.8- to 94.9-Percent Dense

Figure 26. In-Pile and Unirradiated Thermal Conductivity,
 $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, 98.2- to 98.5-Percent Dense

Figure 27. In-Pile and Unirradiated Thermal Conductivity,
 $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, 85.8- to 86.1-Percent Dense

Figure 28. In-Pile and Unirradiated Thermal Conductivity,
 UO₂, 98.5- to 98.6-Percent Dense

Figure 29. In-Pile and Post-Irradiation Thermal Conductivity,
 ThO₂ + 3.1 w/o UO₂, 94.8- to 94.9-Percent Dense

Figure 30. In-Pile Thermal Conductivity at Various Stages of Depletion, $\text{ThO}_2 + 10 \text{ w/o } \text{UO}_2$, 98.2-Percent Dense

Figure 31. Summary Curve, In-Pile and Unirradiated Thermal Conductivity, Polycrystalline UO_2 , 98.5-Percent Dense

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APPENDIX A

FORCING THE SOLUTION OF THE NEUTRON DIFFUSION EQUATION TO FIT A GIVEN DISTRIBUTION IN POWER DENSITY

Solution of the neutron diffusion equation for solid cylindrical geometry yields (A1a) as an expression for the radial distribution in fission power density:

$$Q(r) = \beta I_0(Kr) \quad (A1a)$$

where:

$Q(r)$ = power density by fission at radius r

K = reciprocal neutron diffusion length

$I_0(Kr)$ = zero-order modified Bessel function with argument Kr .

Since $I_0(0) = 1$ and $\beta = Q(0)$, Equation (1a) may be rewritten as:

$$Q(r) = Q(0) I_0(Kr) . \quad (A1b)$$

Let K_a be an adjusted reciprocal neutron diffusion length such that Equation (A1a) is forced to fit a given distribution in power density. Using K_a in a series approximation to the zero-order modified Bessel functions:

$$I_0(K_a r) = 1 + \sum_{n=1}^{\infty} \left[\frac{(K_a r)^{2n}}{4^n (n!)^2} \right] \quad (A2a)$$

$$= 1 + \frac{[(K_a r)^2]}{4(1!)^2} + \frac{[(K_a r)^2]^2}{4^2(2!)^2} + \frac{[(K_a r)^2]^3}{4^3(3!)^2} + \frac{[(K_a r)^2]^4}{4^4(4!)^2} + \dots \quad (A2b)$$

Therefore,

$$\left[\frac{Q(r)}{Q(0)} - 1 \right] = \frac{[(K_a r)^2]}{4(1!)^2} + \frac{[(K_a r)^2]^2}{4^2(2!)^2} + \frac{[(K_a r)^2]^3}{4^3(3!)^2} + \frac{[(K_a r)^2]^4}{4^4(4!)^2} + \dots \quad (A3)$$

Applying series reversion,

$$(K_a r)^2 = AY(r) + BY^2(r) + CY^3(r) + DY^4(r) + \dots \quad (A4)$$

where:

$$Y(r) = \left[\frac{Q(r)}{Q(0)} - 1 \right]$$

and:

$$A = 4$$

$$B = 4/(2!)^2 = -4(0.2500)$$

$$C = 4[1/(2!)^4 - 1/(3!)^2] = 4(0.097225)$$

$$D = 4[5/(2!)^2 - 1/(4!)^2 - 5/(2!)^6] = -4(0.0451389), \text{ etc.}$$

Therefore,

$$K_a \approx \frac{2}{r} [Y(r) - 0.250000 Y^2(r) + 0.097225 Y^3(r) - 0.0453789 Y^4(r) + 0.023078 Y^5(r) - 0.012540 Y^6(r) + \dots]^{1/2} . \quad (A5)$$