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CALCULATION AND MEASUREMENT OF THE TRANSIENT
TEMPERATURE IN A LOW-ENRICHMENT UO₂ FUEL ROD
DURING LARGE POWER EXCURSIONS

J. E. Houghtaling, T. M. Quigley and A. H. Spano

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US ATOMIC ENERGY COMMISSION

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ABSTRACT

This report presents the results of Spert I in-pile transient tests of a rod-type, low-enrichment UO_2 fuel element. The tests were performed to investigate the possibility of damage to such long thermal-time-constant fuel rods when subjected to short-period power excursions, and to test the effectiveness of an instrumentation technique for measurement of UO_2 fuel temperatures within the rods. In an initial series of power excursion tests, in which the range of reactor periods was from approximately 1 sec to 7.5 msec, simultaneous measurements were made of the transient temperature at the center of the fuel rod and at the outer cladding surface. Fuel rod rupture occurred during the exponential rise of the 7.5 msec excursion. Similar short-period tests performed on a second fuel rod containing no internal thermocouples did not result in cladding failure, supporting the postulation that rupture of the first rod was due to waterlogging of the UO_2 as a result of the cladding penetrations made for installation of the internal thermocouples. Calculations of the transient temperature distribution in the fuel rod have been made and the results are found to be in good agreement with the experimental data obtained on the central- UO_2 and cladding-surface transient temperatures.

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CALCULATION AND MEASUREMENT OF THE TRANSIENT
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I. INTRODUCTION

Prior to the Spert I experimental program of self-limiting power excursion tests of a water-moderated, low-enrichment UO_2 fuel rod core⁽¹⁾, several in-pile transient tests of the fuel element used in the oxide core were performed in Spert I, using a water-moderated, highly-enriched, plate-type core as a neutron burst source. These subassembly tests were intended to provide (prior to the commencement of the test program on the full scale core): (a) demonstration of the ability of the oxide fuel element to undergo short-period, large power excursions without damage to the fuel rod and (b) data to evaluate a proposed thermocouple instrumentation technique for measurement of the internal fuel rod temperature under transient conditions.

Considerations leading to a subassembly test of the UO_2 fuel rod involved the low heat transfer properties of the fuel rod, which, during the course of a short-period power excursion, could give rise to large temperature gradients and cladding stresses in the stainless steel jacket, with the possibility of cladding rupture for a sufficiently severe power transient. Since the fuel rods used contained powdered UO_2 , the rupture of one or more fuel rods in the oxide core at the termination of a short period burst could result in the rapid dumping into the moderator water of nearly all of the stored energy in the rods, the subsequent generation of large steam pressures, and the possibility of violent consequences. These considerations were particularly pertinent for the case of fuel rods instrumented with internal thermocouples, since the proposed instrumentation technique required penetration of the cladding jacket for the transducer leads.

Two series of the subassembly transient tests were run: the first on a fuel rod instrumented with internal and cladding-surface thermocouples and the second on a fuel rod instrumented with surface thermocouples only, and whose cladding was intact. The first series was terminated after rupture of the fuel rod during a 7.5-msec period test. Damage was limited to the fuel rod and its aluminum container. Rupture of the fuel rod was believed to be due to a simple steam explosion resulting from waterlogging of the UO_2 . Following cleanup, a second series of power excursion tests in the asymptotic reactor period range from 56 to 7.2 msec was performed on an intact fuel rod. No failure of the fuel rod jacket occurred as a result of these tests. (In the later experimental series on the full scale oxide fuel rod core⁽¹⁾, no cladding failures

or discernible damage to the fuel rods occurred throughout the entire test range of reactor periods investigated, which included reactor periods as short as 3.2 msec.)

Experimental data on the internal and cladding surface transient temperature obtained during the first series of fuel rod tests have been compared with the results of temperature distribution calculations. These calculations and the results obtained are given in the second part of the body of this report, following presentation of the description and results of the experimental measurements.

II. EXPERIMENTAL PROGRAM

A. Description

1. Fuel Rod and Fuel Rod Instrumentation

The N. S. Savannah-type fuel rod used in these experiments consists of a welded seam, stainless steel tube (6 ft long, 0.5 in. OD, with a 0.028 in. wall thickness) containing 1600 g of 4%-enriched UO_2 powder compressed to an effective density of 9.45 g/cm^3 , which is 87% of the theoretical density of UO_2 ⁽²⁾.

The fuel rod in the first test series was instrumented with eleven pairs of internal and outer cladding-surface chromel-alumel thermocouples, spaced at one-inch intervals along a 10-in. section of the fuel rod. Not all of the 22 transducers were employed simultaneously during the test series; the redundancy in instrumentation was intended as backup in the event of thermocouple failure. The technique used for inserting a thermocouple inside the fuel rod consisted of drilling a small (0.013-in.-diameter) hole diametrically through the fuel rod and, as illustrated in Fig. 1, arranging the thermocouple such that the two leads of the thermocouple junction penetrate the cladding on opposite sides of the rod. Positioning of the 0.005 in.-diameter welded-junction at the center of the UO_2 in the fuel rod was made with an estimated uncertainty of about ± 0.03 in. The holes in the stainless steel cladding were covered with a waterproofing epoxy resin. After the epoxy had hardened, the fuel rod was vibrated in an attempt to obtain good contact between the UO_2 powder and the thermocouple junctions. The surface thermocouples were welded to the stainless steel clad, with a separation between the 0.005-in.-diameter wires of about 0.03-in.

Fig. 1 Cutaway view showing method of installation of fuel rod thermocouples.

The fuel rod in the second test series contained no internal thermocouples and its cladding was intact. The cladding-surface temperature of this rod was measured at five points, spaced at one-inch intervals along the length of the rod, using resistance-welded, 0.005-in., chromel-alumel thermocouples.

Thermocouple location was designated by a letter M or S to signify meat or cladding-surface thermocouples, and by the number of inches above or below the center

of the 6-ft fuel rod at which the thermocouple was located. Thus, "M + 1" and "S + 1" signifies the pair of internal and surface thermocouples which were located 1-in. above the center of the fuel rod, while "M - 2" and "S - 2" was the pair located 2-in. below the center of the fuel rod. The eleven thermocouple pairs on the first fuel rod were located +5, +4, +3, +2, +1, 0, -1, -2, -3, -4, and -5 inches from the center of the rod and the five surface thermocouples on the second test rod were located at +2, +1, 0, -1, and -2 inches from the center of the rod. The centers of the first and second (intact) fuel rods were, respectively, 8.5 and 9.8 in. above the bottom of the core.

2. Neutron Driving Source

The Spert I P-core, a water-moderated, highly-enriched, stainless-steel clad, plate-type reactor (3,4), was used as a power burst source for the transient tests of the oxide fuel rod. The fuel rod was positioned by spacers along the center of a water-filled (2-5/16 in. OD, 1/16 in. wall thickness, 7 ft-long) aluminum tube, the bottom end of which was welded to a flat aluminum plate to form a container for the fuel rod. The container was then placed in the P-core, in a centrally located flux trap created by the removal of one of the fuel assemblies. The active height of the reactor was only 22-in. so that approximately 45-inches of the 67-in. active length of the test fuel rod remained outside of the reactor. The arrangement of fuel rod and driving source is shown in Fig. 2. The axial position of the fuel rod relative to the core was such that the center of the fuel rod coincided with the approximate peak of the axial flux distribution in the reactor (Fig. 3).

Fig. 2 Arrangement of fuel rod and aluminum container within the P-core reactor driving source.

Fig. 3 Positioning of the fuel rod thermocouples relative to the axial flux distribution in the reactor.

The axial flux distribution in the core and along the length of the fuel rod was determined by gold foil activation techniques.

B. Internally-Instrumented Fuel Rod Test Series

The fuel rod instrumented with internal thermocouples was subjected to several self-limiting power excursions with initial reactor periods in the range from approximately 1 sec to 7.5 msec. Data obtained in these tests on reactor period, reactor power, and fuel rod temperature at $M + 1$, $S + 1$ and $M + 5$ are presented in the Appendix in Table A-1 and in Figs. A-1 to A-7.

In contrast with the generally consistent behavior of the cladding-surface thermocouples, some of the meat thermocouples appeared to display a somewhat erratic response. This is indicated in Fig. 4 in the time plots of UO_2 temperature rise during a 25 msec period self-limiting power excursion. This behavior is, in part, related to a variable thermal contact between the UO_2 and the thermocouple junction. The size of the UO_2 crystals (of the order of 10 microns) is small compared to the 0.005-in. diameter of the thermocouple junction, and if the crystals are tightly packed against the junction, good thermal contact and smooth response can be expected. In the absence of tight crystal packing, which

Fig. 4 Experimental central- UO_2 transient temperature data during a 25-msec period test.

is a consequence of the method used for inserting the thermocouples in the fuel rod, a variable thermal contact could exist between the junction and the surrounding crystals, as a result of any thermal expansion movement of the crystals during the course of a power transient.

C. Fuel Rod Cladding Rupture

During the exponentially rising portion of a 7.5-msec period power excursion, a 12-in. split in the stainless steel cladding occurred along the length of the fuel rod, approximately in line with the series of holes previously drilled in the cladding for the installation of the thermocouples. In addition, the thinly-welded seal at the bottom of the surrounding aluminum container was broken away, allowing an estimated 700 g of the UO_2 powder from the ruptured fuel rod to be dispersed in the reactor vessel water. Fig. 5 shows the ruptured fuel rod and aluminum container; Fig. 6 illustrates the broken end of the aluminum container; and Fig. 7 is a closeup photograph of the ruptured cladding. In this test the maximum power density obtained in the fuel rod was estimated to be about 50 kw/cm^3 . Evidence that the splitting occurred during the initial portion of the burst is given by the times of failure of the thermocouples, which are indicated in the temperature curves of Fig. 8 by the sudden transition of the curves from smooth to noisy traces, followed shortly by the complete disappearance of the traces. Thermocouple failure appears to occur initially at $M - 5$ (approximately 4 msec before peak power), and then at $M + 1$, $M - 4$, $M - 3$, $M + 2$, $M + 4$, and $M + 5$. Aside from the indicated $M + 1$ time of failure, this order of failure of the thermocouples implies that the split in the rod proceeded from the bottom of the line of thermocouples to the top. The temperature data also show that the peak of the axial power distribution occurred at the $M - 3$ position and that the temperature rise at the time of rupture was nearly 400°C . Inspection of Fig. 7 shows that the point of greatest cladding distortion occurs at the $M - 3$ position, where the split in the cladding goes through the thermocouple hole in the cladding. The indication here is that initial rupture in the cladding occurred at the $M - 3$ hole and not at the $M - 5$ hole, with the split quickly spreading within a few milliseconds upward and downward along the fuel rod.

While the temperature and photographic data do not clearly indicate the exact sequence of events, the results of the inspection of the damaged cladding indicate that it is probable that the cause of the rupture was related to the cladding penetrations made for installing the internal thermocouples. (This contention is also supported by the results of the second series of tests on the intact fuel rod and of the entire series of integral core tests for the Spert I oxide core⁽¹⁾.) It appears likely that, as a result of, perhaps, a combination of radiation breakdown of the epoxy and of differential thermal stresses between the epoxy and the steel cladding, small cracks were created in the cladding, which allowed water to seep into the powdered fuel. The time between the next-to-last transient and the last (7.5 msec) transient was two days. It is presumed that during that time sufficient waterlogging took place to

Fig. 5 Photograph of ruptured fuel rod and aluminum container following cladding failure.

Fig. 6 Photograph of lower end of the aluminum fuel rod container following cladding failure.

Fig. 7 Photograph of the ruptured portion of the internally-instrumented fuel rod with indicated positions of internal thermocouples.

Fig. 8 Experimental central-UO₂ transient temperature data during the cladding-rupture test.

permit a steam explosion to occur which was violent enough to rupture the fuel rod and to blow out the bottom of the aluminum container.

D. Intact-Cladding Fuel Rod Test Series

Following the decontamination of the reactor vessel, a series of twelve power excursion tests was performed on a fuel rod with intact cladding. In this series the initial asymptotic reactor period was varied from 57 to about 7.2 msec. Test data obtained in this series on reactor period, peak power, integrated power to time of peak, and fuel rod surface temperature are given in Table A-2 in the Appendix.

No discernible damage or distortion to the fuel rod occurred as a result of these power excursions, supporting the above postulation that rupture of the first fuel rod was related to the thermocouple cladding penetrations.

III. FUEL ROD TRANSIENT TEMPERATURE DISTRIBUTION CALCULATIONS

A. Computational Method

The central- UO_2 and cladding-surface temperature data obtained in the transient tests described earlier permit comparison with calculated transient temperature distributions in the fuel rod. Such calculations have been performed using an IBM-650 program⁽⁵⁾ to solve the one-dimensional partial differential equation for the radial diffusion of heat in the fuel rod. This program, a modification of the IBM-704 code written by Heestand, Schoeberle and Miller⁽⁶⁾, solves the heat conduction equation in cylindrical geometry for a multiple-region model, using regional, temperature-dependent values of the conductivity and specific heat and a time-dependent source of the fission heat density derived from the experimental transient reactor power.

The geometric model, shown in Fig. 9, consisted of an inner heat-source region of UO_2 , a clad region of stainless steel, and an outer water-coolant region. The radial heat source distribution in the UO_2 due to fissions by slow and fast neutrons is indicated in Fig. 9. This distribution was obtained by a four-group neutron diffusion calculation, using Muft- and Soffocate-derived regional constants, with the hardened thermal constants obtained in terms of the homogenized-cell neutron flux spectrum.

Fig. 9 Three-region model used for radial temperature distribution calculations in oxide fuel pin showing calculated heat source distribution.

For each time step, the program computed temperature-dependent input values for the thermal conductivity and specific heat in each of the (42) sub-regions of the model by solving equations of the form

$$x = \frac{A}{\theta} + B + C\theta + B\theta^2, \quad (1)$$

where θ is the temperature in $^{\circ}\text{C}$, and the coefficients of the respective conductivity and specific heat equations were obtained by least-squares fits of each equation to the published experimental data. Values for the conductivity of UO_2 as a function of temperature up to 1000°C were taken from Kingery's data⁽⁷⁾, with extrapolation to higher temperatures made by means of Bate's radiation heat transfer approximation⁽⁸⁾. In these calculations, conduction of heat from the cladding to the water was

assumed to take place by heat diffusion alone, with no account taken of convection or boiling-conduction effects. The conductivity and specific heats obtained for (9.45 g/cm³) UO₂ and for stainless steel are given in Table 1. For water, the value of 0.998 cal/cm³-°C was used for the specific heat and 0.00144 cal/cm-sec-°C for the thermal conductivity.

TABLE 1

Least-Squares-Fitted Conductivity and Specific Heat
Data in the Temperature Range 20°C ≤ θ ≤ 2500°C

Thermal Conductivity (cal/cm-sec-°C)

$$K(\theta) = \frac{0.507}{\theta} + 1.54 \times 10^{-2} - 1.43 \times 10^{-5} \theta + 4.95 \times 10^{-9} \theta^2$$

UO₂

$$K(\theta) = 3.96 \times 10^{-2} + 2.23 \times 10^{-5} \theta$$

Steel

Specific Heat (cal/cm³-°C)

$$C(\theta) = 5.6 \times 10^{-1} + 4.36 \times 10^{-4} \theta - 1.83 \times 10^{-7} \theta^2$$

UO₂

$$C(\theta) = 0.883 + 8.07 \times 10^{-4} \theta$$

Steel

B. Results

1. Transient Behavior

The results of the temperature distribution calculations performed for transient tests with reactor periods of about 0.5 sec or less are shown in Figs. 10 to 18.

In Figs. 10 and 11, the calculated and experimental results are shown for the time behavior of the M + 1 central-UO₂ and S + 1 cladding-surface temperature rises during the 25- and 49-msec period power excursions, respectively. The degree of agreement between calculation and experiment that is indicated in these two figures is typical of the results obtained for the other transients. A single normalizing factor, relating the total power of the reactor driving source to the energy density in the test fuel rod at the M + 1 position, was applied to the M + 1 central-UO₂ temperature at t = 0.5 sec for the 25-msec period test (Fig. 10), and this factor was used in all calculations. As

Fig. 10 Comparison of calculated and experimental transient temperatures for a 25-msec period test.

Fig. 11 Comparison of calculated and experimental transient temperatures for a 49-msec period test.

indicated in Figs. 10 and 11, the computed and experimental central meat temperatures are in reasonably good agreement throughout the entire excursion, including the post-burst region, whereas, good agreement for the surface temperature is obtained only up to the time at which boiling starts on the surface of the fuel rod. Thereafter, the computed surface temperature, which takes no account of convection or boiling heat transfer, continues to rise while the actual surface temperature remains near saturation. The results of additional calculations, involving increases in the value for the diffusivity of water in order to maintain the calculated surface temperature at saturation after the onset of boiling, indicate that the effect of a constant saturation temperature at the cladding surface on the central-UO₂ temperature during the post-burst period is to reduce the central temperature by a few percent or less (depending on the time after peak power and the reactor period of the transient). The result of this change in boundary condition is to improve, somewhat, the agreement between experimental measurement and calculation.

In Fig. 12, the calculated and experimental cladding-surface temperature rises up to the time of onset of boiling are shown for six transient tests. In Fig. 13, comparison of the calculated and experimental meat and surface temperature rises at the time of peak power indicates reasonably good agreement throughout most of the reactor period range studied.

Fig. 12 Comparison of calculated and experimental cladding-surface temperature rise up to time of onset of boiling for six transient tests.

Fig. 13 Comparison of calculated and experimental temperatures at the time of peak power vs reciprocal period.

Fig. 14 Calculated radial temperature distributions for a 25-msec period test with time as a parameter.

Fig. 15 Calculated radial temperature distributions at time of peak power with initial reactor period as a parameter.

2. Radial Temperature Distribution

The calculated radial temperature distribution in the fuel rod and water is shown in Fig. 14 for three different times during the 25-msec period test. At time $t = 0.5$ sec, approximately 0.17 sec or 7 reactor periods after peak power, the temperature distribution in the UO_2 is seen to be still about the same as it was at peak power. At about 1.5 sec after the burst, however, which is a time delay of the order of the thermal time constant of the fuel rod, the distribution has approximately the asymptotic relaxation shape, with the total energy in the UO_2 decreasing exponentially with a period of about 4 sec. Fig. 15 shows the calculated radial distributions at the time of peak power for five different transients. From these data and the temperature-dependent specific heat data, the fraction of the fission heat generated in the UO_2 up to the time of peak power which has been transferred to the cladding and moderator can be computed. The results are shown in Fig. 16, where the percent heat loss from the UO_2 at the time of peak power is plotted as a function of the asymptotic reciprocal period of the transient. The results indicate that for a 100-msec period self-limiting power burst, approximately 10% of the heat generated in the fuel rod to time of peak power has been transferred, whereas, for a 10-msec period burst only about 1% of the heat has been transferred.

3. Fuel Rod Cladding Thermal Stresses

The transient temperature gradient that is developed across the cladding wall during a power excursion is of importance in considering

Fig. 16 Calculated percent heat loss from the UO_2 at the time of peak power as a function of reciprocal reactor period.

Fig. 17 Calculated transient temperature difference across the fuel rod cladding for the 25-msec test.

Fig. 18 Calculated maximum temperature difference across the fuel rod cladding as a function of reciprocal reactor period.

the likelihood of cladding rupture. As indicated in Figs. 14 and 17, the temperature difference across the cladding during a transient is greatest at about the time of peak power. The temperature difference at peak power increases as the asymptotic reactor period of the self-limiting power burst decreases. In Fig. 17, the temperature difference across the cladding during the 25-msec period test is plotted as a function of time. The maximum temperature difference obtained is approximately 60°C and it occurs shortly after the peak of the power burst. In Fig. 18, the maximum transient cladding temperature difference is plotted as a function of the reciprocal period of the transient. For the case of a 7.5-msec period transient, the calculated temperature difference is about 165°C. This is the reactor period test which, in the initial test series, resulted in rupture of the fuel rod. It is of interest to see if this temperature difference is equivalent to thermal stresses sufficiently large to rupture the stainless steel cladding of the fuel rod.

For the fuel rod which ruptured, the fission gas pressure buildup inside the fuel rod as a result of its previous reactor history was negligible. Also, if it is assumed that the estimated 0.4% differential increase in UO_2 volume by thermal expansion at the time of the power peak of the 7.5-msec test is taken up entirely in the 13% void volume of the porous fuel, then the circumferential stress in the cladding by internal pressure can be neglected, and the principal stress in the cladding will arise from the thermal stress created by the radial temperature gradient across the clad. The diameter-to-wall thickness ratio of the fuel rod cladding is approximately equal to 18, and hence, the thermal stress in the clad can be approximated (assuming the applicability of elastic

theory⁽⁹⁾) by

$$\sigma(\theta) = \frac{\alpha E}{2(1-\nu)} \Delta \theta \quad (2)$$

where

$\sigma(\theta)$ = thermal stress in the cladding (psi)

α = linear coefficient of thermal expansion = $9.2 \times 10^{-6}/^{\circ}\text{F}$

E = modulus of elasticity = 29×10^6 psi

ν = Poisson's ratio = 0.28

Substitution of these values for stainless steel into (2) yields $\sigma(\theta) = 3.3 \times 10^2 \Delta \theta$, where $\Delta \theta$ is the difference in $^{\circ}\text{C}$ across the cladding wall. Thus, for a 7.5-msec test in which $\Delta \theta_{\text{max}} = 165^{\circ}\text{C}$, $\sigma_{\text{max}} = 5.4 \times 10^4$ psi. (In the actual test, rupture of the fuel rod occurred about one-half reactor period before peak power, at which time $\Delta \theta \approx 100^{\circ}\text{C}$, so that $\sigma_{\text{max}} \approx 3.3 \times 10^4$ psi.)

These stresses are smaller than the nominal minimum yield point of 60,000 - 100,000 psi for commercial, cold-worked, 347 stainless tubing. Further, hardness tests of the fuel rod cladding indicated an ultimate strength for the steel tubing of about 150,000 psi. The results of this analysis indicate that rupture of the fuel rod cladding during the 7.5-msec test was probably not a consequence of thermal gradients in the cladding.

IV. CONCLUSIONS

The experimental and calculational results obtained suggest that:

(1) Rupture of the fuel rod cladding during the initial test series was related to the penetrations made in the cladding for the installation of thermocouples and was probably caused by waterlogging of the UO_2 during the time between the next-to-last transient and the last (cladding rupture) transient. This conclusion is supported by the non-destructive results of the second series of short period transient tests on the intact fuel rod and by the numerous short-period tests on an entire oxide core comprised of 592 intact fuel rods. In addition, a calculation of the thermal stress in the cladding wall during the fuel rod rupture power excursion indicates that the thermal stress at the time of rupture was insufficient to explain cladding rupture.

(2) The technique of sealing the cladding penetrations for installation of the internal thermocouples to prevent UO_2 waterlogging and consequent probable fuel rod damage during power excursions has to be improved. In addition, improvement in thermal contact between thermocouple junction and UO_2 is needed to provide reliable transient temperature measurements of the UO_2 .

(3) Reasonable agreement between calculated and experimental transient temperature distributions is obtained in the transient tests studied. The assumption made in the calculational program that heat transfer across the cladding-water interface is by conduction alone (with no account taken of convection or boiling-conduction processes) is not unreasonable in calculations of the reactivity compensated at the power peak due to heat transfer from fuel rods to moderator during short-period, self-limiting power excursions with no forced flow. For long-period transients (of the order of the 4-sec thermal time constant of the fuel rod) or for relatively long times after the peak of a power burst, the assumption of heat transfer by conduction alone tends to break down, resulting in calculated cladding-surface temperatures which continue to rise above the boiling point, in contrast to the actual temperatures which remain near saturation.

APPENDIX

Tables A-1 and A-2 present data summaries of the transient tests conducted on the internally-instrumented and intact fuel rods. Data plots of the experimentally-measured UO_2 transient temperatures at the $M + 1$ and $M + 5$ thermocouple positions and of the $S + 1$ cladding-surface temperatures obtained during the initial test series are shown in Figs. A-1 to A-7.

TABLE A-1

Internally-Instrumented Fuel Rod Test Series Data Summary

Test No.	Reactor Period (sec)	Reciprocal Period (sec ⁻¹)	Peak Power (Mw)	Clad-Surface Temperature Rise to Peak $\theta_{(S+1)}(^{\circ}\text{C})$	UO ₂ Temperature Rise to Peak $\theta_{M+5}(^{\circ}\text{C})$	UO ₂ Temperature Rise to Peak $\theta_{M+1}(^{\circ}\text{C})$
1	1.1	0.91	0.49	67	105	165
2	0.54	1.9	0.78	48	98	160
3	0.25	4.0	1.7	44	75	117
4	0.23	4.3	1.8	32	70	102
5	0.049	20	16	22	90	158
6	0.043	23	22	14	108	156
7	0.025	40	47	20	126	225
8	0.0075	130	300	-	-	-

TABLE A-2

Intact Fuel Rod Test Series Data Summary

Test No.	Reactor Period (sec)	Reciprocal Period (sec ⁻¹)	Peak Power (Mw)	Surface Temperature Rise to Peak Power $\theta_S(^{\circ}\text{C})$
1	0.057	17.6	17	20
2	0.056	17.9	12	24
3	0.033	30.3	37	17
4	0.023	43.5	56	12
5	0.023	43.5	56	19
6	0.022	45.5	55	18
7	0.020	50.0	65	18
8	0.015	66.6	112	16
9	0.011	91	158	12
10	0.011	91	189	13
11	0.0075	133	308	13
12	0.0072	139	318	15

Fig. A-1 Transient temperature data for 1.1-sec period test.

Fig. A-2 Transient temperature data for 0.54-sec period test.

Fig. A-3 Transient temperature data for 250-msec period test.

Fig. A-4 Transient temperature data for 230-msec period test.

Fig. A-5 Transient temperature data for 49-msec period test.

Fig. A-6 Transient temperature data for 43-msec period test.

Fig. A-7 Transient temperature data for 25-msec period test.

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