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LOS ALAMOS SCIENTIFIC LABORATORY
of the
University of California
LOS ALAMOS • NEW MEXICO

Quarterly Status Report on
Ultra High Temperature Reactor Experiment (UHTREX)
for Period Ending October 31, 1967

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ULTRA HIGH TEMPERATURE REACTOR EXPERIMENT (UHTREX)

I. INTRODUCTION AND SUMMARY

During this reporting period, the program of UHTREX critical experiments (UCX-II) shifted from the initial emphasis on achieving a critical loading and calibrating control rods to performing certain zero power measurements of some system parameters. Three determinations of the power profile in the core were made. Absolute measurements of fission rate were then used to calibrate the neutron flux detectors. A preliminary value for the relative reactivity worth of U and (U-Th) fuel elements was obtained; these data will be useful in planning later experiments in which Doppler effects might be observed.

Rod drop experiments were done in order to provide a basis for comparing the calculated and observed neutron lifetime in the ^{235}U -fueled core. The two values were found to be in reasonable agreement.

Additional calculations were made to assist in determining why the critical mass observed in UCX-II differed (by about 10%) from that predicted on the basis of an earlier critical experiment (UCX-I). It was concluded that a number of differences in detail between the two experiments contributed to the inaccuracies in the prediction of the cold critical loading for the completely assembled reactor.

Considerable time was spent on training of the operating staff; the reactor console and the control room were used to provide a real-life environment. An analog computer, programmed to mock-up the

reactor system, was connected to the main operation's console and provided real-time simulation of the signals and component responses which are characteristic of the UHTREX complex.

II. POWER MEASUREMENTS IN THE UHTREX

The power generated in two specific fuel elements during a 2 kW-10 minute irradiation run of the reactor was determined by radiochemical measurements of the total number of fissions that occurred in the two elements. In order to establish the total power of the reactor, it was essential, in addition, to measure the core power profile; or, in effect to determine the ratios of the powers in all the fuel elements to the power in the two elements for which the energy release was measured absolutely. When the total power in the core was known, calibration constants could be assigned to the power-level neutron detector.

The power profile measurements were made by observing the fission product gamma activity of the fuel elements with a scintillation spectrometer. The gamma measurements were relative in the sense that no attempt was made to determine the total gamma activities of the elements. Rather, a standard, reproducible counting geometry was used so that accuracy in the relative gamma count of any fuel element compared to any other element was insured. Standard corrections for the decay in the gamma activities during the counting operations, and for small individual differences in fuel element loadings were made.

Because of the cylindrical symmetry of the core it was not necessary to determine the relative powers

of all 1248 fuel elements. If all 312 channels had been loaded identically, it would have been necessary only to measure the power profile in the 52 elements contained in one vertical row of channels. Because of the symmetry the same profile would appear in all other vertical rows of channels.

However, in order to reach the cold critical loading of 6.309 kg with essentially all control rods withdrawn, a mixture of two different channel loadings was required. With minor exceptions, all channels in a vertical row of channels were loaded identically with one or the other of the two channel loadings. Thus, by determining the power profile in two different vertical rows of channels containing the two different channel loadings, the powers in all the fuel elements in the core would be inferred.

The power profile data are plotted in Figs. 1 and 2. The individual profiles represent the variation in relative fissions per gram of ^{235}U in fuel elements from the bottom to the top of the core for a given radial position, I , in a vertical row of channels. The index $I=1, \dots, 4$ denotes the innermost to outermost radial positions, and the index $J=1, \dots, 13$ identifies vertical channel positions.

The profile in Fig. 1 is for a vertical row of channels in which the four fuel elements in each channel contain loadings 4.2, 4.8, 5.7, and 6.7 g ^{235}U per element. The profile in Fig. 2 is for a vertical row of channels containing a reduced channel loading of 3.5, 4.2, 4.8, and 5.7 g ^{235}U per element. The relative number of fissions per gram of fuel is approximately proportional to the thermal flux. Thus, the fissions per gram in the heavier channel loading (Fig. 1) are generally smaller than the fissions per gram in the lighter loading (Fig. 2). The increased absorption of thermal neutrons in heavier loadings causes a reduction in the thermal flux. However, if total fissions or relative powers per fuel elements are compared for fuel elements in equivalent radial and vertical positions in Figs. 1 and 2, the heavier loadings yield higher powers per element in spite of the reduced thermal flux.

The uncertainty in a typical point on the curves is estimated to be $\pm 5\%$; it is attributable to drift in the gamma-counting equipment and to the

Fig. 1 Power profile for azimuthal position $K=7$, channel loading: 4.2, 4.8, 5.7, 6.7 g ^{235}U . Total integrated power = 1.52×10^6 W-sec. Normalization factor = 340.5 W-sec/g.

lack of a precise value for the ^{235}U content of any particular fuel element.

Calculated power distributions for the UCX-II core configurations have not yet been completed, so a comparison of predictions with the results of the experiment cannot be made. In UCX-I, the observed fission rate distributions differed somewhat from those which had been expected, and a similar situation may exist for the UCX-II.

When the power profile in the all-U core had been measured, a second, similar experiment was done after 52 of the standard fuel elements had been replaced by ones containing ^{232}Th as well as ^{235}U . The characteristics of these elements, which contain 102 g ^{232}Th and 10.2 g ^{235}U , and the methods used in their manufacture are reported in LA-3760-MS. (1) Such elements may be used in future experiments whose objectives are to study the Doppler coefficient in Th-U systems.

In this experiment, the 52 all-U elements in a vertical row of channels containing the lighter

Fig. 2 Power profile in UHTREX core channels at azimuthal position K=7. Fuel elements loadings (from center outward) were 3.5, 4.2, 4.8, and 5.7 g ^{235}U /fuel element. Total integrated power = 1.52×10^6 W-sec; normalization factor = 340.5 W-sec/g.

channel loading, 3.5, 4.2, 4.8, and 5.7 g ^{235}U per elements, were replaced by the U-Th fuel. From advance calculations of the effect of this substitution, the attempt was made to specify the loadings so that no reactivity change would occur under the substitution. Actually an average reactivity increase of 1.4% per element was observed. This was in part because the Th-U elements contained 12% more U than was planned for this experiment. Also, the Th-U elements displaced lighter loaded channels in the all-U loading. The calculations were made for a core containing uniformly-loaded channels. The above differences will be taken into account in future calculations.

After irradiation, the U-Th elements were removed from the reactor and their gamma activities measured. An absolute fission rate determination was made on one U-Th element (the innermost element from the core mid-plane). It is estimated that the absolute fission rate determination may be in error

by $\pm 10\%$.

The observed power profile is shown in Fig. 3.

Fig. 3 Power profile for U-Th fuel in azimuthal position K=7. Total integrated power = 1.63×10^6 W-sec. Normalization factor = 304 W-sec/g.

In the comparison of the profiles of Figs. 1, 2, and 3, it is seen that the thermal flux is relatively higher at the top and bottom surfaces of the core for the Th-U elements than for the all-U elements. Because of the greatly increased thermal absorption in the Th-U elements, the effect of the return of thermal neutrons from the reflectors is emphasized. Both the power plot and the reactivity measurements on the Th-U elements provide an experimental reference for future design calculations of Th-bearing fuel loadings in UHTREX.

Table I lists the normalized energy release/g. ^{235}U at the core mid-plane for three different ^{235}U contents in the fuel element. The data indicate the local variations in power which may occur in a core containing different channel loadings.

III. MEASUREMENT OF NEUTRON LIFETIME

An estimate of the prompt neutron lifetime, Λ , in UHTREX was made by observing the initial, rapid

TABLE I

COMPARISON OF ENERGY RELEASED
FROM FUEL ELEMENT IN POSITION J=7, I=1

	Irradiation Run No.		
	1	2	3
Loading, g ^{235}U	4.2	3.5	10.2
Fission energy, W-sec	1428	1288	2885
Specific fission energy, W-sec/g ^{235}U	340	368	283

decrease in power when the reactor was scrammed from a 2 kW power level by dropping one plug rod. Although this method does not permit a precise determination of Λ to be made, it can be used to give a semi-quantitative assessment of the accuracy of the calculated neutron lifetime. The observed value was found to average reasonably well with earlier estimates.

If the number of delayed neutrons is assumed to be constant over the short ($\sim 1/2$ sec) duration of the rod-drop experiments, the reactor kinetics equations become simply:

$$\dot{N}(t) = \Lambda^{-1} \rho(t) N(t)$$

or, when integrated:

$$\frac{N(t)}{N(0)} = \exp \left[\Lambda^{-1} \int_0^t \rho(p') dt' \right]$$

In order to evaluate Λ from the observed variation in neutron population, $N(t)/N(0)$, the impressed reactivity, $\rho(t)$, must be accurately known, regardless of whether the simplified expression (above) or the exact kinetics equation is used in the analysis.

A dual beam oscilloscope was used to record simultaneously the rod position and the power level as a function of time during the rod drop experiment. Reactivity calibration of a plug rod had been performed during UCX-I, and checked over a limited range of reactivity during UCX-II. The data on position vs time, reactivity vs position, and total worth of the rod were combined to yield $\rho(t)$. Figure 4 shows a plot of this function.

Fig. 4 Reactivity input function $\rho(t)$ for scram of one plug rod.

Once the $\rho(t)$ for the rod drop experiment was known, the expected response of the reactor was calculated, using an estimated value of Λ , and compared to the $N(t)/N(0)$ observed. The trial values of Λ and β , had been computed earlier with the DAC code for the predicted cold critical loading of 5.7 kg. The delayed neutron factor, β , is relatively independent of the critical loading, so the calculated value, $\beta = 0.67851$, should be nearly correct for the observed critical loading of 6.3 kg ^{235}U . Λ , on the other hand, in first approximation, inversely proportional to the amount of fuel in the reactor. For the 5.7 kg core, Λ was computed to be 9.643×10^{-4} sec. Correcting this value to the 6.3 kg loading yields $\Lambda = 8.72 \times 10^{-4}$ sec. The reactor "rod drop" response was computed with values of Λ which bracketed that estimated (see Table II for input data). Figure 5 shows the two calculated curves of $N(t)/N(0)$ as a function of time. The observed ratios $N(t)/N(0)$ at selected times have been plotted on the same graph. Error bars on the experimental data indicate the estimated uncertainty

introduced by noise on the oscilloscope trace. Numerical results calculated with $\Lambda = 8.72 \times 10^{-4}$ sec would lie midway between the two curves of Fig. 4. The estimated value of the lifetime is therefore consistent with the results of the experiment.

TABLE II

NEUTRONIC PARAMETERS USED IN CALCULATING UHTREX
RESPONSE TO ROD DROP

$$\Lambda = \begin{cases} 9.643 \times 10^{-4} \text{ sec} \\ 7.714 \times 10^{-4} \text{ sec} \end{cases} \quad \beta = 6.7851 \times 10^{-3}$$

i	$\beta_i = a_i \beta$	$\gamma_i \text{ (sec}^{-1}\text{)}$
1	2.22551-04	1.24400-02
2	1.48187-03	3.05100-02
3	1.33395-03	1.11440-01
4	2.67808-03	3.01370-01
5	7.83000-04	1.13631+00
6	2.85653-04	3.01370+00

IV. ANALOG AND PROCESS COMPUTERS

A. Reactor Operations Simulator

An analog computer that is programmed to simulate the reactor system has been connected to the main operations console for use in reactor operator training and in control systems checkouts. The dynamic response of the analog system closely resembles that of the real system. The simulation, which operates in real time, uses a simple reactor system model based on an earlier and more sophisticated digital model of UHTREX system dynamics.⁽²⁾

The simulation system is shown schematically in Fig. 6. Abbreviations used in the figure are defined in Table III. The actual control rod drive, heat dump fans, bypass valve, main operations console and process computer are used in the overall simulation. Thus, all of this equipment receives an operational check when the system is used for operator training.

The analog computer is made up of two TR-48 Electronic Associates, Inc. computers, with 115 amplifiers, 10 integrators, 14 multipliers, and

Fig. 5 Calculated response to rod drop for two different neutron lifetimes. Plotted points are those observed in the experiment.

5 diode function generators. A three-group, one-point, reactor neutron kinetics model is used. The reactor heat source is represented by a two-lump model; one-lump models are used for the recuperator, main heat exchanger and heat dump. These one-lump models, whose transfer functions match those of more accurate models only at frequencies less than 0.1 Hz, are relatively crude approximations to the physical systems. However, each component of the control system acts as a low pass filter in transmitting signals to the next component, and when all analog components are connected, little high frequency energy is present in the system. Therefore, since component model behavior is good at low frequencies, the component simulation response is good. The response of the analog computer model to system events such as loss of primary flow and increase in bypass flow agrees well with the reactor response predicted by detailed calculations run on a digital computer. For instance, the analog reactor neutronic power decrease on loss of primary flow is indistinguishable from the digital program response.

Fig. 6 UHTREX simulation system.

TABLE III

ABBREVIATIONS USED IN FIGURE 6

BS_1	Primary loop blower speed
BS_2	Secondary loop blower speed
FB	Fraction of primary flow bypassed around recuperator
P_1	Primary loop coolant pressure
P_2	Secondary loop coolant pressure
Q_n	Reactor neutronic power
T_{avg}	Average reactor solid temperature
T_{dci}	Heat dump cold inlet air temperature
T_{dho}	Heat dump hot outlet coolant temperature
T_i	Reactor inlet coolant temperature
T_o	Reactor outlet coolant temperature
T_{xci}	Heat exchanger cold inlet coolant temperature
T_{xco}	Heat exchanger cold outlet coolant temperature
T_{xhi}	Heat exchanger hot inlet coolant temperature
T_{xho}	Heat exchanger hot outlet coolant temperature
$\dot{W}_1$	Primary loop coolant flow rate
$\dot{W}_2$	Secondary loop coolant flow rate
$\dot{W}_3$	Heat dump air flow rate
X_b	Recuperator bypass stem position
X_r	Reactor control rod position
ρ	Net reactivity

The simulation is now being used in an extensive reactor operator training and qualification program. Each operator must demonstrate his ability to make changes in system operation, to start up the reactor, and to identify and act on a variety of system failures. All training is conducted from the actual main operations console. Meters, recorders, and annunciators respond just as they should in the actual circumstances.

B. Process Computer

Basic computer programs to monitor analog and digital inputs, to annunciate and alarm, to log, to interlock, to calculate, to load fuel, and to record fuel history are complete and are working

well in their present form. The computer has controlled and monitored the loading of more than 2500 fuel elements into the reactor and has recorded the loading successfully. It has also performed well in the reactor critical experiments, in the checkout of the coolant loops, and in other system component tests.

The programming and hardware for systems that permit use of the process computer for control of primary and secondary coolant flow have been finished and checked in a simulated mode. These systems give a rate-limited response, without overshoot, to operator demands. An understanding of the stability of these systems has been obtained through the use of z-transforms. An adequate stability margin and a variety of self-checking features are incorporated in each control system to assure its proper operation.

IV. REACTOR PHYSICS

A. Critical Experiments

The UHTREX cold critical mass was observed to be 6.3 kg ^{235}U during the course of UCX-II. The predicted value, which had been normalized to the results of the first critical experiment, UCX-I, was 5.7 kg. Apparently the measures which had been taken to bring calculations and observations into agreement in UCX-I were inadequate in the case of UCX-II, which was carried out in the completely assembled reactor.

One thing which was not explicitly included in the normalization of UCX-I calculations with results of the experiment was the reflector-effect associated with the concrete walls of the small cell in which the experiment was done. Failure to include the concrete in the normalization of the calculational model could lead to an underprediction of the UHTREX (UCX-II) critical mass. To evaluate this effect in UCX-I, new calculations were performed. A 1-D (cylindrical) model of the reactor assembly was used, the dimensions of which are given in Table IV. From two sets of k-calculations (one with the concrete included in the reactor representation, and the other with it omitted) the worth of the concrete walls was found to be $\Delta k = +0.46\%$. The 16-group Hansen cross sections were used, since the most important effect of the concrete is on fast neutrons.

TABLE IV

ONE-DIMENSIONAL (CYLINDRICAL) REPRESENTATION
OF UCX-I AND THE CONCRETE WALLS OF THE
CRITICAL ASSEMBLY CELL

Region	Outer Radius (cm)
Plug	30.0
Core	85.9
Graphite Reflector	106.7
Dense Carbon Reflector	132.9
Void	182.9
Concrete Reflector	289.6

It appears that use of the simple, symmetrical model implied by Table IV will lead to an overestimate of the worth of the concrete as a radial reflector. But this is compensated for to some extent by the exclusion, in the model, of axial reflectors, also of concrete, which were present in the experiment. It appears that the concrete walls were worth at most +0.5% Δk . If this had been included in the normalization, it would have led to no more than a 2% increase in the predicted UCX-II critical mass. The "concrete wall effect" therefore appears to be too small to account entirely for the larger than predicted critical mass observed in UCX-II.

The possibility that a change in the absorbed moisture content in the UHTREX components could affect the critical mass in UCX-I and UCX-II was also considered. Perturbation calculations indicated that an increase of 0.36 w/o in water content would be required to explain the discrepancy in the experiment. A change of this magnitude does not appear to be credible.

In order to further investigate the discrepancy, a more accurate one-dimensional model for UCX-I was designed using a radial trace of the UCX-I two-dimensional model. A reference calculation was run for the 7-kg critical UCX-I core using six energy groups, S_4 approximation, and a buckling height of 150 cm. The six groups include one thermal group and five fast groups, and the space model consists of two plug regions, four core

regions, and two reflector regions. Selected variations in the reference calculation were then made in an attempt to isolate the causes of the discrepancies discussed above. Results are given in Table V, which indicates the perturbations introduced and changes which they produced in k_{eff} relative to k_{eff} for the reference calculations.

The variations considered in problems 2 to 9 make little or no difference in k_{eff} , nor do they affect the flux slope. The effects considered in problems 13 to 15, when taken together, would yield approximately the required adjustment to k_{eff} but have no effect on the flux slope. Problem 10, in which the radial reflector is reduced to 31.0 cm (from 46.2 cm), gives approximately the correct adjustment to k_{eff} and a flux slope in agreement with experiment. However, an arbitrary chopping off of the radial reflector is unsatisfactory from a physical standpoint. The same effects can be obtained by a reduction in the graphite density, as shown for problem 12. Problem 11 results show that the outlet coolant plenum, which varies from 1.5 to 2.5 in. in width, has a significant effect on both k_{eff} and the flux slope.

The results of this investigation led to a re-examination of graphite densities used in previous calculations. Using the actual weights for the various graphite components and detailed engineering drawings, it has been found that the homogenized graphite densities in both the core and reflectors are somewhat less than those previously used in the computation model. Calculations with these new data are in progress.

Thermal cross sections for all the calculations reported above were obtained with the SUMMIT and THERM codes. As a check on these codes, calculations are being made with the LASL equivalents of these codes, the TOR and GLEN codes.

B. Critical Experiment - The Cell Problem

Experimental data now on hand strongly indicate that the remaining discrepancies between calculation and experiment on UHTREX critical masses are most likely due to inadequate treatment of the heterogeneities in the UHTREX core in previous analyses. In UHTREX, as in most reactors, the cell approximation is used for calculating the effects of the heterogeneities. In the cell approximation, it is assumed that the core is divisible into a large number of

TABLE V

UCX-I CALCULATIONS, SHOWING THE EFFECT OF SELECTED PERTURBATIONS
ON K_{eff} FOR A 7 kg CRITICAL CORE

<u>Variation from Reference Calculation</u>	<u>Δk_{eff}</u>	<u>Relative Core Thermal Flux Slope*</u>
1. None - reference calculation		-0.0089
2. Doubled space mesh	+0.0002	-0.0090
3. Twenty-five energy groups (13 thermal, 12 fast)	-0.0015	-0.0089
4. Pointwise convergence in DTF	0	-0.0089
5. Run with S_8	-0.0004	-0.0089
6. Run with S_{16}	-0.0004	-0.0089
7. Leakage (DB^2) in THERM	+0.0001	-0.0089
8. Leakage ($H = 55.88$ cm) in DTF cell	-0.0006	-0.0089
9. Self-shielded resonance cross section for ^{235}U and ^{238}U **	+0.0010	-0.0089
10. 31.0 cm radial reflector***	-0.0288	-0.0105
11. First 2 in. of radial reflector replaced by void	-0.0132	-0.0094
12. Graphite density in radial reflectors reduced by 10%	-0.0137	-0.0096
13. Doubled B concentration in all out-of-core regions	-0.0069	-0.0089
14. σ_{5+6} for C reduced by 10%	-0.0056	-0.0089
15. Re-evaluated thermal cross section for ^{235}U and C	-0.0066	-0.0089

* The relative core thermal flux slope calculated from UCX-I data is -0.0110.

** Reference calculation done with unshielded resonance cross sections.

*** Reference calculation has 46.2 cm reflector.

identical regions or cells. Then calculations are performed on a single cell, with the tentative assumption that the core consists of an infinite array of such cells. These calculations provide neutron fluxes as a function of position within the various regions of the cell. The cell fluxes, in turn, are used as weighting functions on the neutron cross sections of the materials in the cell to obtain the equivalent homogenous medium cross sections. These cell-average cross sections are used in finite-geometry calculations of the core and surrounding media.

For UHTREX, the cell approximation is applicable in the vertical and azimuthal directions. In the vertical direction there are 13 identical, equidistant fuel channels separated by homogenous carbon moderator. In the azimuthal direction, with the fuel channels separated by equal angular increments of 15° , the cell approximation should be rigorous.

In the radial direction, however, the variation in loading and the geometrical divergence invalidate the cell approximation. Nevertheless, in the past the cell approximation was used in the radial direction because no computational facilities existed for treating the problem otherwise.

Experimental plots of the power profile in the UHTREX core show that the radial structure of the core is not treated adequately by the cell approach; a greater radial slope in the power is observed than is predicted by the calculations. This would indicate that the radial neutron leakage is underestimated. Measured and calculated vertical power profiles agree reasonably, supporting the cell approximation in the vertical direction.

A geometry for UHTREX cell calculations which does not require the cell approximations in the radial direction is shown in Fig. 7. The boundary conditions may be described in terms of a three-

dimensional cylindrical R, θ, Z geometry. In the vertical direction, Z , and the azimuthal direction, θ , the boundary conditions are such that the vertical neutron current, J_z , and azimuthal current, J_θ , vanish at the cell boundaries; but, in the radial direction, the entire structure of the reactor must be included with a vanishing flux, ϕ , or vacuum boundary condition at the outer surface.

Fig. 7 UHTREX cell geometry.

An unfortunate aspect of this geometry, aside from its three-dimensional character, is that the R, θ, Z coordinate system which fits the boundary conditions is not suitable for specifying the fuel element geometry, which requires a cylindrical axis oriented along the fuel channel.

It is apparent that a specialized code is needed to solve this problem. A multigroup diffusion theory approach appears feasible. Examination of the numerical difference equations and the details of constructing a suitable code, leads to the conclusion that an X, Y, Z , Cartesian mesh of about 25,000 points is more suitable than cylindrical geometry. Although frequent access to data stored peripherally on the computer will be required, code running times should not be prohibitive. Work on the code is in progress.

V. UHTREX TRANSIENT EXPERIMENTS

As part of the planning of an HTGR-oriented program for UHTREX and its associated facilities, a study was made of the feasibility of constructing a critical assembly capable of simulating the

transient behavior of high temperature gas-cooled reactors. A well-shielded critical assembly building (used for UCX-I) is available and might be suitable for such a series of experiments. In an evaluation of the building's usefulness for experiments involving high power nuclear transients, an estimated was made of radioactivity which would be induced in the steel walls of the shielded cells, and in the steel vessel which would contain the critical assembly. If the activity level were low, the experiment might be designed so that "contact maintenance" or modification would be possible, once the fuel were withdrawn from the assembly.

To obtain an estimate of the activity induced by a 100 MW-sec power pulse, an S_4 calculation (spherical geometry) was performed for a core with C:Th and C:U ratios of 250 and 5000, respectively. The model included the core, the graphite reflector, heaters, stainless steel insulation, and thermal shield, the reactor vessel, a 20-in.-thick void, the steel pit-wall liner, and three feet of concrete. The main contributors to the dose rate (48 hours after the transient) were found to be ^{59}Fe and ^{60}Co . Each of these isotopes contributes approximately equally to the dose rate immediately after the burst. Without taking into account the decay of the ^{59}Fe and ^{60}Co (half-lives of 45 d and 5.26 y, respectively), the dose rate at a point midway between the vessel and the pit-wall liner was estimated to be 3 mrem/h from the vessel and internals and 0.6 mrem/h from the pit-wall liner. It appears that entry of personnel into the critical assembly cell would therefore not be restricted by the induced radioactivity in the cell and vessel walls.

VI. FABRICATION OF SPECIAL FUEL ELEMENTS

The fuel elements for the proposed Doppler experiments contain a mixture of Th and U oxides, described in LA-3760-MS. Pressings of mixed oxides prepared by various methods indicated the best material was a 2.18 μ coprecipitated powder prepared from the peroxides. Trial pressings, $\sim 3/8$ -in.-diam x 0.395-in.-high, were made on a Stokes pill press and fired at 1750°C for 16 h in air. Total shrinkage was 11-13%; the fired density was ~ 8.35 g/cc. This density value compares favorably with that reported in the literature for material of similar composition.

Microscopic examination of the fired samples indicated that presence of agglomerates with some separation between the agglomerates and the surrounding material. Chemical analysis indicated that some segregation had occurred. The U content varied from 7.34 to 7.60% and the Th content varied from 85.6 to 87.4%.

As can be seen from these results, the final Th/U ratio was not 10.2 as it was prior to firing. This segregation occurred either during the original precipitation or during waxing. If it occurred during waxing, a higher calcining temperature might produce a more stable product, thus preventing segregation during subsequent operations. The Uranium Chemistry Group is preparing a batch of similar material using a higher calcining temperature.

Graphite sheathing for the Th/U core in the Doppler experiment was extruded and processed through graphitization. The pieces were 0.993 in. o.d. x 0.733 in. i.d. and had an average density of 1.86 g/cc.

VII. FLUID CIRCUITS

The use of fluid circuits in the measurement of high temperature is being studied. This application requires that circuit components be miniaturized and be fabricated from materials such as niobium. During this report period, a stacked-foil fluid circuit assembly was diffusion bonded. The assembly was fabricated by first bonding the top, bottom, and center sections separately. This was accomplished by vapor depositing 0.00002 in. Ti on the chemically milled Nb foils, stacking the foils and applying a 60 psi load, and heating to 1600°C in vacuum for 1 h. Vapor deposited Ti was then applied to these sections bonded in a similar manner to form the fluid circuit assembly. Two separate diffusion bonding cycles were required to get good bonding of the foils over the channels. Though the foils were "fabrication rejects," the completed assembly in this trial assembly can be used for certain tests. A second assembly fabricated from acceptable components is in process.

References

1. LA-3760-MS, "Quarterly Status Report on Advanced Reactor Technology (ART) for Period Ending July 31, 1967, Pt. 1." August 24, 1967.
2. "System Dynamics Study for the Ultra High Temperature Reactor Experiment," H. B. Demuth, K. Duerre, F. Schilling, and C. Stiles, LA-3561, August 1966.