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

Quarterly Status Report on
Advanced Reactor Technology (ART)
for Period Ending January 31, 1967
Part I

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

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ULTRA HIGH TEMPERATURE REACTOR EXPERIMENT (UHTREX)

Reactor Components

Reactor Pressure Vessel

The reactor vessel, with the other components of the primary and secondary coolant loops, passed a pressure test to 688 psig with N_2 and a leak test at 500 psig with He. Following the leak test, the water-cooled panel and the shielding slabs were set in place over the vessel and the control rod tube extensions were welded in place. Thermocouples were welded to the surfaces of the top cap, the elevator housing, the nozzles, and other areas where skin temperatures are to be monitored. The vessel is ready for the installation of the control rod drives, the fuel elevator, and the fuel loader tubes.

Control Rods

Both sets of control rods are ready for installation. The first rod drive was delivered and is undergoing acceptance tests at LASL. Four other drives, under test at the vendor's plant, will be delivered shortly. Instrumentation is ready at the reactor site for the immediate installation and checkout of the rod drives.

Fuel Handling System

With the exception of the fuel loader components that are to be mounted on the reactor vessel, the fuel handling system is complete and has been operated under manual control. A computer program that will control the fueling operations is written, debugged, and is operating with a mockup system. The fuel system operating panel in the control room is complete, wired, and checked out. Permanent hangers are being installed on the core indexing drive, following a test of the latter in conjunction with the core position indicator and operations panel in the control room.

The first eight of the 13 fuel loader tubes are ready for installation on the reactor vessel and the remainder await only testing and painting. Machining has been completed on the preloader tube, the last component of the system. The inlet gas lock, which mounts on the preloader, has been assembled and tested, and is ready for installation.

Coolant Systems

Leak Testing

The primary and secondary coolant loops were pressure tested to 688 psig with N_2 and leak tested at 500 psig with He. Mass spectrometer tests of all joints in both systems revealed only one leak, namely, in the seal weld of the flange which connects the 4-in. instrumentation pipe to the recuperator. During this first leak test, the blower internals were not installed, the seal weld was not made in the primary loop filter, and the pressure transducers were not installed. Another leak test will be made when the reactor vessel and coolant loops are complete.

Installation of Components

After the leak test, surface thermocouples were mounted in the loop components and insulation was installed. The primary loop bypass valve operator was installed and tested successfully from the control room. Work was completed on the heat dump and this unit, too, was operated from the control room console. When the work, now nearly done, on the control system for the secondary loop isolation valves is completed, the coolant loops will be ready for cold flow tests.

Gas Cleanup System

Leak Testing

Separately from the coolant loops, the main process He vessels and piping of the gas

cleanup system were pressure tested to 688 psig with N₂ and leak tested at 500 psig with He. During the pressure test, one leak was found in the seal weld of a transducer. Under He pressure, the valves performed well, all welded joints were acceptable, and there were no external leaks. However, leaks were detected into the buffer zones of some of the double-seal Conoseal joints. These leaks are being repaired with new gaskets and tightening of the joints. Another leak test of the entire system will be made after auxiliary systems, e.g., fuel loader purge and blower jacking gas systems, are completed.

Operational Tests

The demineralized water, glycol refrigeration, and hot air utility systems have undergone successful operations tests.

Instrumentation

Control Room

All of the graphic and operation panels and chassis are installed, wired, and checked out on the control room end. Approximately 90% of the instrumentation and control wiring is installed and tested.

The gas cleanup system panels were used during the system pressure test to operate the valves and to indicate and record system parameters. Logs of the test were prepared by the process computer.

Computer Systems

A program to save the process digital computer state tables and data tables has been added. This program allows the computer to be stopped and restarted without losing track of the core location, fuel loading status, hold amplifier assignments, status of valves, or analog inputs. A considerable number of

revisions and additions have been made to the operations console programs, the analog, digital, and annunciator scans, the utility typewriter program, and the computer tables.

A preliminary version of a closed loop flow control program and a breadboard circuit worked well together in a simulated control loop. Response to a unit step change in flow rate demand was roughly of the form $f(t) = 1 - e^{-kt}$, with the final simulated flow value controlled to within about 0.02 lb/sec of the demanded value. On the basis of the success of this trial control system, circuit diagrams for primary and secondary coolant flow control systems have been prepared.

For the analog computer, models of neutron kinetics, reactor core heat exchange, the recuperator, the heat exchanger, and the heat dump have been wired on patch boards. The neutron kinetics model was checked out previously. Static and dynamic checks have been completed on the reactor core heat exchange simulation. Several transfer functions were compared with those obtained from frequency domain analysis programs solved on the IBM 7094 and were found to be in good agreement. Work is proceeding on the other component models.

Reactor Physics

Calculations for UHTREX Startup

The UHTREX startup procedure includes an initial core temperature increase to about 650°C. This will be performed with the heat dump fan running at standard speed, and the primary and secondary blowers both set at about 40% of their maximum speeds. Under these conditions, it is estimated that a power of 0.78 MW will be required to stabilize the core at a volume-averaged temperature of 653°C. Corresponding inlet and outlet core temperatures are 382° and 577°C, respectively.

The critical mass at 653°C has been calculated with procedures developed earlier in the calculations of the room temperature and operating temperature critical masses.¹

Results for the three temperatures are given in Table 1. The following temperature fit of the data

$$M_c = 5.63 + 2.19 \times 10^{-3} T + 5.35 \times 10^{-7} T^2$$

where M_c is the critical mass in kg of 93% enriched U and T is the temperature in °C, should give reasonable interpolations for other temperatures.

Table 1. UHTREX Critical Masses

T(°C)	M_c (kg 93% ²³⁵ U)
20	5.68
653	7.29
1441	9.90

The critical mass estimates apply only to configurations where all control rods are withdrawn. In addition, corrections for Xe poisoning have not been included. The Xe effects will be small at 653°C, but at 1441°C the critical mass will increase to about 10.5 kg as Xe in equilibrium at 3 MW is approached.

UHTREX Transient Experiments (UTE)

Cores for the UHTREX Transient Experiments (UTE) can be designed so that all the important mechanisms influencing the transient behavior of high-temperature, gas-cooled reactors (HTGRs) are present. The fuel elements for such cores would contain

coated particles of the types proposed for the Public Service Company of Colorado reactor (PSC). Three different designs based upon the use of these particles are summarized in Table 2. With each type of fuel element, the UHTREX would be loaded with a full load of 1248 fuel elements. Each fuel element would be a solid cylinder 1 in. in diam and 5.5 in. long. The fuel elements can contain up to 60 v/o coated particles admixed with graphite binder material.

Design A will emphasize the effects in fast transients of time delays associated with the transfer of heat from the fissile particles. Since these particles contain no Th, heat must be transferred from the fissile kernel to the fuel moderator or the fertile particles before the moderator or Th Doppler coefficients can act.

The advantage of design B is its simplicity. Since only one kind of particle, the fissile particle, is present, the transient particle heat transfer effects are relatively uncomplicated. Furthermore, except for the addition of Th to the particles, and the fact that the fuel elements are solid, the design B fuel elements are similar to the standard UHTREX fuel elements.

The first core fabricated especially for UTE will most likely contain design C fuel elements, because this design will provide the most realistic mockup of the response of large HTGRs to fast transients. By reducing the thickness of the isotropic layers on the particles,

Table 2. Fuel Element Designs for UTE

Design	Fissile Particles		Fertile Particles	
	Kernel (200-μ diam)	Coatings	Kernel (350-μ diam)	Coatings
A	UC ₂ only	30 μ buffer 70 μ isotropic	ThC ₂ only	30 μ buffer 100 μ isotropic
B	UC ₂ + ThC ₂ (Th/U=4)	30 μ buffer 100 μ isotropic	None	None
C	UC ₂ + ThC ₂ (Th/U=4)	30 μ buffer 30 μ isotropic	ThC ₂	30 μ buffer 30 μ isotropic

as is done in design C, the carbide fraction in the fuel elements is greatly increased without exceeding the limit of 60 v/o particles. It should be possible to make the Th/U atom ratio relatively large in the design C elements.

The relative ease with which new cores can be fabricated and loaded in UHTREX makes it reasonable to consider testing a variety of cores in the transient experiments. Each core will be designed to emphasize the action of a particular mechanism. For example, in a core of design A fuel elements, the particle heat transfer effects are accentuated.

Another fuel element design would accentuate the Doppler effect in Th and suppress the thermal base coefficient of the carbon moderator. The central core of the fuel element would be fabricated from pressed disks or rods of mixed oxides of U and Th. A central core diameter of 0.3 to 0.5 in. would provide critical loadings of U and sufficient space for Th/U ratios of 8:1 to 12:1. The remaining radial thickness of the fuel element of 0.25 to 0.35 in. would be used to provide a radiation gap, an insulating layer of porous carbon or ceramic insulator, and a thick-wall graphite container.

With this thermal insulation, the response to large reactivity insertions should be almost entirely controlled by the Doppler effect in Th. Such transients could be superimposed on steady-state temperatures from 20° to 1800°C. By providing coolant flow in the fuel channels, bulk moderator temperatures can be kept relatively low while elevated steady-state temperatures in the fuel cores are achieved. The coolant flow would also improve the effectiveness of the thermal insulation during transients.

By using the tapered loading technique developed for UHTREX, the steady-state temperature can be varied continuously without resorting to control rod motions. For

safety reasons, it may be desirable to perform a gradual displacement of a known loading by the Doppler loading while transient tests are being performed. This is readily accomplished with the UHTREX fuel loading equipment.

In addition, fuel elements containing ^{233}U can be tested. The fuel handling facilities in UHTREX provide gamma-shielded, alpha-tight containment. Thus, no special modifications would be required for handling fuels containing ^{233}U . By varying the temperature in the assembly and observing the corresponding variations in critical masses and temperature coefficients, integral data on the thermalization effects in ^{233}U with ^{235}U additions would be sufficient to obtain soft spectrum data. With a more abundant supply of ^{233}U , harder spectrum data could be obtained, and the effects of elements for thermal spectrum adjustment, such as Rh or ^{240}Pu , could be evaluated.

It is expected that the first transient experiments will be done with the standard UHTREX fuel elements because they are already on hand, and because such experiments will provide valuable data on the contribution of the moderator temperature coefficient to HTGR shutdown mechanisms.

Preliminary calculations have been completed for the design A fuel. The steady-state results in Table 3 were obtained with the one-dimensional DTF code² in S_4 approximation, with 25-group cross sections. The most significant conclusion is that sufficient Th can be loaded in a UHTREX fuel element to achieve a substantial ^{232}Th Doppler temperature coefficient, although less than 20% of the particle volume is occupied by the particle kernels, and the particles occupy only 60% of the fuel element volume. (The proportion of kernel volume is 13% in the fissile particle and 19% in the fertile particle.)

Table 3. Summary of 1-D Results for U-Th Loaded UHTREX

	Room Temp (293°K)	Design Temp (1714°K)
²³⁵ U loading (kg)	8.4	14.9
²³² Th loading (kg)	71	63.4
Doppler coefficient ($\Delta k/k$ per °C)	-1.7×10^{-5}	-5.2×10^{-6}
Fuel moderator coefficient ($\Delta k/k$ per °C)	-1.4×10^{-5}	-1.1×10^{-5}
Bulk moderator coefficient ($\Delta k/k$ per °C)	-1.18×10^{-4}	-1.44×10^{-4}
Effective delayed neutron fraction	6.85×10^{-3}	6.84×10^{-3}
Effective neutron generation time (sec)	8.43×10^{-4}	5.05×10^{-4}

Transient calculations were also performed on the design A fuel. The heat transfer in the fuel was treated in terms of an average fissile particle and its associated binder material and fertile particles. The temperature coefficients and the kinetics were treated similarly in terms of a one-point model. In spite of these approximations, the results given in Table 4 should be adequate for design evaluations. The reactivity inputs and various model assumptions leading to the results in Table 4 are as follows:

Case 1. \$2 reactivity step input with reactor initially at 1 W delayed critical.

Case 2. Same as Case 1, except that heat transfer from central UC₂ to UC₂ buffer layer is by radiation only.

Case 3. Same as Case 1, except that UC₂ buffer conductivity is 10 times reference value.

Case 4. Same as Case 1, except that heat transfer time delays are neglected.

Case 5. Same as Case 1, except that Doppler coefficient is 30% less.

Case 6. Same as Case 1, except that fuel moderator coefficient is 30% less.

Case 7. \$2/sec ramp terminated after 1 sec; otherwise same as Case 1.

Case 8. \$3 step; otherwise same as Case 1.

Case 9. Same as Case 8, except that heat transfer time delays are neglected.

Table 4. Summary of Transient Results

Case	Time to peak power (sec)	Peak power (MW)	Energy to peak power (MW-sec)	Maximum UC ₂ temp (°K)	$(\rho_0 - \beta)/\Lambda$ (sec ⁻¹)	Initial asymptotic inverse period (sec ⁻¹)	Fraction of feedback due to Doppler coeff. at peak power
1	2.359	236	54.4	1950	8.13	8.47	0.49
2	2.344	269	57.8	3756	8.13	8.47	0.46
3	2.355	222	51.8	1020	8.13	8.47	0.50
4	2.353	205	48.6	1010	8.13	8.47	0.54
5	2.379	275	63.7	2231	8.13	8.47	0.40
6	2.378	280	64.1	2256	8.13	8.47	0.59
7	2.978	234	53.8	1939	8.13	8.47	0.48
8	1.312	981	114	6560	16.3	16.5	0.11
9	1.304	755	91.6	1500	16.3	16.5	0.54

Since there is no Th in the fissile particles in design A, the fission heat generated in the particles must be transferred to the fuel moderator before the fuel moderator coefficient can act, and to the fertile particle kernels before the Th Doppler coefficient is effective. A comparison of Cases 1 and 4 shows that the peak temperature attained in the fissile particles is indeed changed substantially by the time delays in heat transfer, but other power pulse parameters, such as the time to the peak power and the magnitude of the peak power, are not strongly affected.

Other conclusions from the results in Table 4 are:

1. The conductivity of the buffer zone in the fissile particles strongly affects the peak temperatures reached in the particles (Cases 2, 3, 4).
2. Rod withdrawal times of about 1 sec will be sufficiently rapid for initiating transients (Cases 1 and 7).
3. Observations on the initial asymptotic period should provide good values for the neutron generation times in room temperature loadings (fifth and sixth data columns, all cases).

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Fuel Elements

Bead Damage

The results of bead damage, gamma count, and density measurements for the current UHTREX core loading were completed. The extrusion mix was reextruded several times to aid in the mixing of the ingredients. During these reextrusion cycles, bead damage was encountered. It was found that the lower loadings (148 to 202 mg U/cm³) reextruded four times did not show excessive bead damage; about 0.04% of the total U was leached from the element. After the fifth extrusion of a batch, the bead damage increased to as much as 1.8% U removed by leaching for the 202 mg U/cm³ loading. Slightly higher U losses were encountered after four extrusions of the 237 mg U/cm³ lots.

Loading and Density

The results derived from gamma counting and density measurements are presented in Table 5.

Bead Coating Studies

Tungsten coated UC₂ beads are required for use in UHTREX and for other reactor thermocouple applications. A thin (0.2 mil) W coating is needed to allow W wires to be welded directly to each bead. Various methods of applying the coating were investigated. Initial attempts made involved applying a thin pre-

Table 5. Loadings and Density of UHTREX Fuel Elements

Loading (mg U/cm ³)	Grams U per Element		Average Density (g/cm ³)	Carbon Density (g/cm ³) *
	Specified	Average		
148	7.69	7.66	1.920	1.772
173	8.99	8.93	1.935	1.762
202	10.52	10.48	1.961	1.759
237	12.31	12.09	1.965	1.728

* Obtained by subtracting the U loadings from the density.

coat of W by the pyrolysis of $W(CO)_6$ followed by a dense W coating deposited by the H_2 reduction of WF_6 . Although coatings could be applied by this method, the beads were not weldable.

Consequently, pre-coats of PyC and Ni were used to prevent the fluoride from reacting with the UC_2 . Problems with discontinuities in the Ni coatings were encountered (see below) which enabled the fluoride to reach the UC_2 ; similar problems were associated with porous PyC coating. Successful coatings have been obtained using a pre-coating of porous PyC and dense PyC $\sim 10 \mu$ thick. This duplex coating allowed the W to be deposited on the nominal 200- μ diam UC_2 beads from WF_6 without attack of the UC_2 by the fluoride.

Fuel Testing

Fuel Migration

Migration of U from PyC-coated fuel particles at high temperatures is being studied in out-of-pile laboratory experiments. In the first of a series of tests triplex-coated UC_2 fuel particles, bonded in a UHTREX-type fuel element, were exposed to temperatures between 2950° and 3100°F in a high-temperature, He-filled furnace. Microradiographs of element sections 0.030 and 0.004 in. thick revealed no migration of U from the kernel of the fuel particle.

A second test, in which the fuel element was heated to 3100°-3300°F has just been completed and examination of the disks cut from the element will begin shortly. A third test is under way, in which the fuel element will be heated to 3300° and 3400°F, the limit of the furnace.

Thermal Properties of Pyrocarbon Coatings

A fission couple device was used to study the thermal properties of a PyC buffer layer around a UC_2 kernel in an experiment done in

the fast burst reactor, Super Kukla, at the Nevada Test Site. The coated particles consisted of UC_2 kernels, 100 to 200 μ diam, coated with a sooty deposit of buffer layer carbon about 50 μ thick. For the attachment of thermocouple wires, the particles were coated with a thin Ni layer and then an outer W shell about 3 μ thick. Unfortunately, the Ni layer was not coherent and HF, from the W coating process, penetrated and attacked the UC_2 . The properties of the resulting particles were not uniform. Consequently, the burst at Super Kukla gave scattered results which could not be resolved to give a unique value for thermal diffusivity.

A second test is planned for the near future with particles having improved W coatings prepared in the manner described in the preceding section.

Instrumental Development

Thermocouple Life Tests

Thermocouples, of materials and design similar to those to be used in UHTREX, were cycled between 1500° and 2500°F for a cumulative time of 4050 h. The developed emf outputs continue to be satisfactory and are equivalent to the temperatures read with optical pyrometers.

UHTREX Neutron Flux Monitor

A fission-couple "thermopile" for use as a neutron flux monitor in UHTREX is in preparation. The fissile beads in the device, shown in Fig. 1, are 0.030-in. -diam cylinders cut from a length of wire made of an alloy of 20% U (93% enriched) and 80% Nb. The advantages of this material are: (1) The alloy melts at 2300°C (4170°F). (2) It is inert in air at room temperature; therefore no coating is needed prior to attaching the thermocouple wires. (3) The cylindrical shape is compatible with the cylindrical hole in the Al_2O_3 disk which houses the particles. (4) The welding of the fine thermocouple wires is easier on the flat end of a cylinder than on the curved surface of a sphere.

Fig. 1. Fission-couple thermopile.

In the past, to stabilize single bead units, the beads and wires were coated with a clear plastic cement. During the first experiment at the Omega West Reactor (OWR), the emf output was observed to decrease steadily with time. This effect on the signal and an examination of the unit after the experiment, indicated that the plastic cement had carbonized, become slightly conductive, and short circuited an increasing portion of the thermocouple output. An Al_2O_3 cement was substituted for the plastic cement in the UHTREX fission couples at the expense of more difficulties in handling the assemblies.

In a second OWR test an attempt was made to measure the effect of bead size on the output of a fission couple. Two small beads and two large beads (~ 500 - and 900 - μ spheres, respectively) were exposed to $\sim 2 \times 10^{13}$ neutrons/ cm^2 -sec. The output was found to be directly proportional to particle diameter with a constant of about $0.69^\circ\text{C}/\mu$ of UC_2 . To compensate for the effect of bead size on output, the distribution of particle sizes in each unit is the same. To effect a further improvement, the buffer bead was changed from W to Nb when the fissile bead was changed from UC_2 to the

U-Nb alloy. The Nb bead more nearly matches the gamma heating characteristic of the alloy than does W.

Three of the fission-couple assemblies are to be placed in the source hole of UHTREX. A fourth unit will be available for placement outside the reactor vessel near the more standard neutron counters. One of the fission couple assemblies for UHTREX will be tested in a furnace out-of-pile. If time permits, it will be given a short exposure in the OWR to obtain an indication of the output level with neutron flux. More extensive OWR experiments are planned with a fifth assembly.

Strain Gauge Testing

Rokide-coated strain gauges, prepared as evaluation samples by BLH Electronics (West), were tested at temperatures up to 300°F . One set of samples (indicated by S-6 in Fig. 2) was

Fig. 2. Effect of temperature on strain gauges.

matched to a mild steel base and another was for use on Al (S-13). Both were attached to a mild steel plate similar to the material of the UHTREX pressure vessel. The former show deviations from zero, due to temperature rise only, at a rate of $+1.77$ microstrain/ $^{\circ}\text{F}$ and the latter at a rate of $-3.07/^{\circ}\text{F}$. To attain a zero rate, other units had to be selected for testing; simple interpolation between the two results suggested that an S-9 unit might be satisfactory. However, when tried, a disappointing -2.6 microstrain/ $^{\circ}\text{F}$ was obtained. An additional test on another S-6 unit gave a rate of $-0.76/^{\circ}\text{F}$, showing the irregularity between gauges. The indications, plotted in Fig. 2, were observed under conditions of zero strain.

The tests conducted locally were limited to temperatures in the vicinity of 300°F . BLH Electronics experiments predicted further complications are to be expected when, at temperatures just above 300°F , the linearity disappears and the slopes of the curves reverse in sign. The solution to the strain gauge problem is not in sight and no further tests are planned at present.

PLASMA THERMOCOUPLE RESEARCH

and

HEAT PIPES

The results obtained during the current quarter on plasma thermocouple and heat pipe research are classified and are described in the ART Report Part II, LA-3666-MS (Confidential).

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