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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 April 30, 1967
Part I

UNITED STATES
ATOMIC ENERGY COMMISSION
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This LA...MS report presents the status of the LASL Advanced Reactor Technology (ART) program. Previous Quarterly Status Reports in this series, all unclassified unless otherwise noted, are:

LA-3244-MS	LA-3582-MS
LA-3316-MS	LA-3625-MS
LA-3370-MS	LA-3665-MS (Part I)
LA-3431-MS	LA-3666-MS (Part II-Conf.)
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ERRATA TO LA-3708-MS

Page 4

Under Reactor Physics, Calculations for UHTREX Operation

Delete second paragraph and replace with the following:

The estimated worths of the poisons Xe and Sm at 3-MW equilibrium are \$2.27 and \$1.09, respectively. In terms of the equivalent fuel required to compensate for the poisons, the worths are 670 and 322 g, respectively. At a steady power of 3 MW, Xe will reach near-equilibrium in about two days. The required total fuel loading will then be about 10.6 kg. Since about 200 days at 3 MW will be required for Sm to reach equilibrium, a significant amount of fuel, namely 600 g, will at the same time be burned up. Thus, Sm poisoning is most appropriately a factor of the long-time fuel depletion and fission product poisoning problem.

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Report written: May 1967

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ULTRA HIGH TEMPERATURE REACTOR EXPERIMENT (UHTREX)

Reactor Components

Reactor

The reactor is completely assembled, operational tests of reactor control systems are in an advanced stage, and preparations for the cold critical experiment are nearing completion. All of the control rods and drives are installed. The plug rod drive controls and associated nuclear safety systems are ready for use.

Fuel Handling System

Installation of the fuel loader rams on the reactor vessel is finished except for the final seal welds on drive motor housings. The entire fuel system is ready for operational tests, during which numbered dummy elements will be loaded into the core. For these tests, the fuel system will be operated under combined manual and computer control to determine the most practical modes of operation for loading fuel during the cold critical tests.

Coolant Systems

Cold Flow Tests

Generally good results were obtained in an extensive series of cold flow tests run to verify the design calculations and to check the functions of the coolant systems. In the first series of tests, which involved all of the components of the primary and secondary coolant loops, 58 steady state runs were made. The He coolant pressures and flow rates were chosen so that, in at least one run for each component in both loops, the Reynolds numbers at standard design conditions were modeled and system parameters were measured with the actual control and data recording systems. The loop components performed as expected, but the tests revealed some deficiencies in instrumental hardware.

For the second part of the tests, the temperature rise in the primary loop due to blower operation alone was measured. With a 42-kW power input to the blower, the gas temperature rose about 20°F and the core solid temperature increased about 15°F within a 2-h period. It appears practical to use this method of heating the core for experimental measurements of the reactor's temperature coefficient of reactivity during the cold critical experiments.

In succeeding tests, the functions of the pressure regulation system were tested successfully, and the dynamic characteristics of the blower and fan speed-regulation systems for the heat dump were determined.

Finally, all of the safety systems that protect the coolant loops were tested under dynamic conditions. The following systems performed as expected: secondary coolant flow, primary coolant flow, primary pressure relief, secondary pressure relief, primary emergency coolant loss, and secondary emergency coolant loss. In each case, when the safety system was triggered, it operated the actual hardware with which it is associated.

Only one system, the recuperator bypass flow control, could not be tested satisfactorily, because of erratic data from differential pressure transducers. Tests on this system will be repeated.

Gas Cleanup System

All components of the gas cleanup system, with the exception of the chemical analysis instruments, were operated successfully in startup tests which were run with sections of the cleanup system on bypass, separated from the primary loop. During the tests, the computer performed its logging, monitoring, and interlocking functions.

Secondary Containment

Preparations are being made for a leak test of the secondary containment portion of the reactor building with all equipment installed. Previous tests were made on the partially empty structure.

Instrumentation

Control Room

Installation of equipment and wiring in the control room is finished. The control systems for the coolant and gas cleanup systems were checked out during operational tests, as also were the associated safety systems. Functional tests are being run on the fuel handling and reactor control systems.

Computer Systems

Essentially all input channels are connected to the computer. The analog and digital input systems and the conversion routines are working satisfactorily.

During the cold flow test the computer was used to log steady state data for analysis. Prior to the cold flow test all valve interlocks were checked and, with one revision, were found to work correctly.

New programs which perform the following jobs have been added: rotate the reactor core and sample the core thermocouples in sequence; calculate the building leak rate during a leak test; make calculations, log, and output on paper tape for flow tests; prepare a paper tape dump at all channels for permanent records; keep a fuel history and fuel history log; and make possible an increase in the number of console functions. Operations console functions are now available to change points from one scan rate to another and to change designators, conversion constants, points on logs, units and scales. These functions make it much easier to update the computer tables when changes are necessary. After a consider-

able number of revisions were made in the control system for the fuel system load rams, dislodger, preloader, and elevator, the fuel loading program was revised and is currently being checked.

Reactor Physics

Calculations for UHTREX Operation

The DAC perturbation code was used for estimating the worths of fuel mass additions and poisons on UHTREX at design operating conditions. The estimated critical mass at design temperature (1441°C - core volume average), assuming no poisons are present, is 9.9 kg fuel ($\sim 93\%$ ^{235}U). For this loading, 58% of the channels contain the tapered loading

10.52 g, 8.99 g, 7.69 g, 6.57 g,
and 42% contain the lighter loading
8.99 g, 7.69 g, 6.57 g, 5.67 g.

The loaded mass in this configuration is increased by adding elements containing 10.52 g fuel to the channels containing the lighter loading. DAC calculations indicate that the fuel added by this type of loading change is worth $2.3 \times 10^{-5} \Delta k/g$ or 0.34¢/g.

The estimated worths of the poisons Xe and Sm at 3-MW equilibrium are \$2.27 and \$1.09, respectively. In terms of the equivalent fuel required to compensate for the poisons, the worths are 670 and 322 g, respectively. At a steady power of 3 MW, Xe will reach near-equilibrium; a significant amount of fuel, namely 600 g, will at the same time be burned up. Thus, Sm poisoning is most appropriately a factor of the long-time fuel depletion and fission product poisoning problem.

Following a shutdown of UHTREX after operation at 3 MW to Xe equilibrium, the amount of Xe will increase to a maximum of 11% above the operating equilibrium population in about 10 h. The Sm after a shutdown will increase at most 6%.

UHTREX Transient Experiments

It is expected that the first transient experiments will be done using the standard UHTREX fuel. Since this fuel is already on hand, the only additional hardware needed will be a modified drive for rapid withdrawal of a plug rod, and the transient recording instrumentation. The nuclear characteristics of the standard fuel are well understood from the experiments in the UHTREX Critical Experiment and from the considerable calculational work done on this fuel.

Transients performed with the standard fuel will provide a good test of calculational methods. In addition, data of interest to HTGRs will be obtained on the transient action of the thermal base or thermalization temperature coefficient. Since there is no Th present in the UHTREX fuel, there will be no competing Doppler temperature coefficient to complicate the interpretation of those initial experiments. However, since the Doppler effect is also of considerable importance to the safety of HTGRs, separate experiments have been proposed for studying the effect of the Doppler temperature coefficient on transients.¹

Detailed calculations of the transient response of UHTREX are in progress. In the calculations to date, the ANCON code² has been used to study the effect of various feedback models on the response of the reactor to subprompt critical step inputs of reactivity. All the transients were initiated from 1 W delayed critical with the reactor unpressurized at room temperature and no coolant flow.

The first series of calculations was done with a 2-lump feedback model for the core. In this model, the entire core was considered to be one region containing a core moderator lump and a fuel lump. Heat transfer by radiation from the fuel lump to the core moderator lump was allowed, but heat transfer from the core to the surrounding reflector regions was neglected.

Fig. 1. Variation of heat capacity of graphite with temperature.

Three cases were studied with the 2-lump model. In case A, the heat capacities and temperature coefficients of the two lumps were assumed to be constant and equal to the room temperature values. Case B was the same as case A, except that the variation with temperature of the heat capacities was that shown in Fig. 1 from data given in Ref. 3. Case C calculations were the same as case B, except that the temperature coefficients of the two lumps were assumed to decrease linearly by a factor of 1.5 between 300° and 1800°K.

Results of the 2-lump feedback model calculations are given in Table I. It can be concluded that, for the reactivity inputs considered, the variation in the lump heat capacities is more important than that in the temperature coefficients. The initial asymptotic inverse period is, by definition, the stable inverse period established before thermal feedback becomes important. Thus, this quantity is independent of the feedback model, heat capacities, and temperature coefficients.

In order to determine the accuracy of the 2-lump feedback model, calculations were done with 8- and 56-lump feedback models for case A. These models take into account the power distribution and the distributed thermal reactivity feedback in the core. The required

TABLE I
RESULTS OF 2-LUMP FEEDBACK MODEL FOR STEP INPUTS OF REACTIVITY

Step (%)	Peak Power (MW)			Time at Peak Power (sec)			Total Energy Release (MW-sec)		
	Case A	Case B	Case C	Case A	Case B	Case C	Case A	Case B	Case C
10	0.0182	0.0194	0.0195	1128	1144	1146	10.0	10.9	11.0
30	0.192	0.234	0.239	257	268	269	28.5	37.8	39.3
50	0.723	0.936	0.964	95.3	100	101	47.6	70.3	75.2
70	2.00	2.68	2.81	40.4	43.2	43.2	67.3	107	117
90	5.33	7.28	7.42	18.7	20.0	19.8	87.6	146	165

Step (%)	Initial Asymptotic Inverse Period (sec ⁻¹)			$\Delta \bar{T}_{\text{fuel}}$ (a) (°C)			$\Delta \bar{T}_{\text{core}}$ (b) (°C)		
	Case A	Case B	Case C	Case A	Case B	Case C	Case A	Case B	Case C
10	0.010	0.010	0.010	64.9	64.0	64.5	3.86	4.12	4.17
30	0.053	0.053	0.053	210	204	209	11.0	13.8	14.4
50	0.156	0.156	0.156	344	344	349	18.3	24.8	26.4
70	0.391	0.391	0.391	473	457	487	25.9	36.2	39.3
90	0.890	0.890	0.890	598	578	627	33.7	47.8	53.1

(a) $\Delta \bar{T}_{\text{fuel}}$ is the maximum rise in the average fuel temperature during the transient.

(b) $\Delta \bar{T}_{\text{core}}$ is the rise in average core temperature due to the total energy release in the transient assuming that there is no heat loss from the core.

distributions were computed with the data analysis code (DAC) for the steady state case, and they were assumed to remain constant during the transient.

For the 8-lump model, the core was divided into four annular regions, corresponding to the positions of the four radial fuel elements. This model thus takes into account the radial distribution in power and thermal reactivity feedback. Each region contains a fuel lump and a core moderator lump. Heat transfer by radiation from the fuel lump to the core moderator lump in each region and heat transfer by conduction between the core moderator lumps were allowed. In the 56-lump model, the core was divided into 28 annular regions corresponding to the four radial and seven axial fuel element positions. Symmetry about the midplane of the core was assumed. This model takes into account both the radial and axial distribution in power and thermal reactivity feedback. Again,

each region contains a fuel lump and a core moderator lump. The heat transfer between lumps was as described for the 8-lump case.

Results for the 2-, 8-, and 56-lump feedback models are compared in Table II. This table shows that, for the transients under consideration, the 2-lump model is adequate. In fact, the 2-lump model agrees within 2% for all parameters compared with the 8- and 56-lump models. This is due to the fact that fuel element powers and temperature coefficients do not vary greatly with position in the core.

Self-Shielding in Fuel Particles

A comparison has been made of several methods of calculating particle thermal neutron self-shielding factors, S . One method is based upon the assumption of an isotropic surface source of neutrons coating the kernel of the

TABLE II

RESULTS OF 2-, 8-, AND 56-LUMP FEEDBACK MODELS FOR STEP INPUTS OF REACTIVITY (CASE A)

Step (¢)	Peak Power (MW)			Time at Peak Power (sec)			Total Energy Release (MW-sec)		
	2 Lumps	8 Lumps	56 Lumps	2 Lumps	8 Lumps	56 Lumps	2 Lumps	8 Lumps	56 Lumps
10	0.0182	0.0181	0.180	1128	1127	1127	10.0	9.93	9.84
30	0.192	0.191	0.190	257	257	257	28.5	28.2	28.0
50	0.723	0.722	0.716	95.3	95.0	94.9	47.6	47.1	46.7
70	2.00	1.97	1.96	40.4	40.4	40.4	67.3	66.5	65.8
90	5.33	5.33	5.29	18.7	18.7	18.6	87.6	86.3	85.5

Step (¢)	Initial Asymptotic Inverse Period (sec ⁻¹)			$\Delta \bar{T}_{\text{fuel}}$ (°C)			$\Delta \bar{T}_{\text{core}}$ (°C)		
	2 Lumps	8 Lumps	56 Lumps	2 Lumps	8 Lumps	56 Lumps	2 Lumps	8 Lumps	56 Lumps
10	0.010	0.010	0.010	64.9	64.5	63.9	3.86	3.82	3.79
30	0.053	0.053	0.053	210	209	207	11.0	10.9	10.8
50	0.156	0.156	0.156	344	343	340	18.3	18.1	18.0
70	0.391	0.391	0.391	473	471	467	25.9	25.6	25.3
90	0.890	0.890	0.890	598	596	592	33.7	33.2	32.9

particle. For an optical diameter

$$y = \Sigma_a \cdot d,$$

where Σ_a = macroscopic absorption cross section and d = particle diameter, there is obtained, for $y \ll 1$, a self-shielding factor

$$S_1 = 1 - (1/3)y + (1/12)y^2 - (1/60)y^3 \pm \dots$$

Case et al.,⁴ assume a spherical particle in an isotropic flux and obtain a slightly different self-shielding factor

$$S_2 = 1 - (3/8)y + (1/10)y^2 - (1/48)y^3 \pm \dots$$

In these treatments, particle interactions are neglected. Doub⁵ uses heuristic arguments to develop a formula which allows for particle interactions. The resulting self-shielding factor is

$$S_3 = \frac{1}{(2/3)y (V/g)} \ln \frac{1}{1 - (2/3)y (V/g) S_2},$$

where S_2 = the series given above, V = actual volume fraction particles, and g = volume fraction for tightly packed particles. Doub considered only uncoated particles and took $g = 0.740$, the value for ideally packed spheres; he found his formula agreed well with experimental transmission measurements for 37 v/o

natural B₄C spheres (87 μ mean radius) mixed with Al spheres (of comparable mean radius).

For small values of

$$X = (2/3)y (V/g) S_2,$$

Doub's formula becomes

$$S_3 = S_2 (1 + X/2 + X^2/3 + X^3/4 + \dots).$$

A fourth method of estimating self shielding, including particle interactions for a regular lattice, results from the application of diffusion theory to a spherical cell model.^{6,7} Self-shielding factors estimated on this basis are too large. Self-shielding factors for a typical UHTREX fuel element, have been calculated by the four methods described above; the results are given in Table III.

Table III

Typical UCX Self-Shielding Factors

$$y = 0.1574; V = 0.00894$$

Designation	Method	Value
S ₁	Surface source	0.9495
S ₂	Case, et al.	0.9434
S ₃	Doub	0.9439
S ₄	Diffusion theory	0.9985

A Monte Carlo calculation was also performed with similar assumptions of particle and surrounding moderator properties. Specifically, it was assumed the particle kernels (UC₂ 0.016-cm diam) are disposed in graphite in an infinite cubic array of 0.063-cm pitch. In this geometry, only one corner of a cube, containing $\frac{1}{8}$ of the cube volume and one quadrant of the spherical particle, needed to be considered. Then, mirror reflection of the neutrons was assumed on all surfaces of the resultant cube. Monoenergetic thermal neutrons were assumed, with a spatially uniform source of neutrons.

Macroscopic cross sections for the particle were taken to be $\Sigma_a = 9.692$ and $\Sigma_s = 0.6065$, and those for the graphite were taken to be $\Sigma_a = 3.207 \times 10^{-4}$ and $\Sigma_s = 0.4110$. Despite the simplicity of the problem, fairly long computing times were involved, because many wall reflections were required, on the average, in order to obtain one nuclear collision. The mean free path of a neutron in the moderator is 2.43 cm, which is quite large compared to the cell dimensions.

The same problem constants were used in a spherical cell calculation using the DTF code in S_n approximation. The spherical particle was placed at the center of a spherical cell of moderator, with particle diameter and cell volume being kept the same as in the Monte Carlo problem. A white boundary condition was prescribed at the outer boundary of the cell. Monoenergetic thermal neutrons, with a spatially uniform source distribution, were again assumed.

A comparison of the results of the Monte Carlo calculation with DTF results at quadratures S_4 and S_8 is given in Table IV. All results are normalized to one source neutron. Regional self-shielding factors are defined, in terms of average regional fluxes, by

$$S(\text{region}) = \frac{\phi(\text{region})}{\phi(\text{cell})}.$$

Table IV.
Comparison of Particle Self-Shielding Analyses

Region	Absorptions			Average flux $\times 10^{-4}$			Self-Shielding Factors		
	Monte Carlo	S_4	S_8	Monte Carlo	S_4	S_8	Monte Carlo	S_4	S_8
Fuel	0.9959	0.9961	0.9961	4.5935	4.5834	4.5834	0.9484	0.9467	0.9459
Moderator	0.0039	0.0039	0.0039	4.8457	4.8436	4.8478	1.0005	1.0005	1.0005
Cell	0.9998	1.0000	1.0000	4.8434	4.8412	4.8454			

The results agree well among themselves and with those given in Table IV. The conclusion, therefore, is that interaction effects between these particles are small and that the simpler methods are adequate for calculational purposes. Calculations on more closely packed particle arrays are in progress.

Fuel Testing

Fuel Migration

Out-of-pile laboratory experiments were continued in a study of the migration of U from PyC-coated fuel particles at high temperatures. UHTREX fuel elements, made with triplex-coated UC₂ fuel particles, were heated to 3150°-3220°F in a graphite furnace under a He atmosphere, held at temperature for 2 weeks, cooled, and removed for examination. Fuel migration from the particles and an accumulation of U at the surface of a fuel element were revealed by microradiographs of sections cut from an element. Since migration has occurred at a lower temperature, a test run at furnace temperatures in the 3300°-3400°F range was discontinued.

Thermal Properties of Pyrocarbon Coatings

A value of 0.4 Btu/h-ft² (°F/ft) for the thermal conductivity of a PyC buffer layer around a UC₂ kernel was estimated from the first set of calculations based on data from an experiment done in the fast burst reactor, Super Kukla, at the Nevada Test Site. For the experiment, the coated particles had UC₂ kernels, 100-200-μ diam, coated with a 30-60-μ thick layer of buffer of 1.0 g/cm³ density. The latter was enclosed in a 3-μ thick coating of

dense C covered with a 2-4 μ outer layer of W, to which chromel and alumel thermocouple wires were welded. The instrumented particles were exposed to a series of neutron bursts in the reactor and the temperature rise in the particles was recorded. From these heat-up curves, by comparison with curves calculated with various values of buffer layer thermal conductivity, k, the experimental value was estimated. Analyses are being made of data from later experiments in which particles with various PyC coating characteristics were exposed to neutron bursts.

Two sets of experiments are being prepared to obtain data for a comparison of the values of k derived from the fission experiments with those measured by the flash technique. Specimens for both sets of experiments were prepared simultaneously in a coating bed which contained UC₂ particles and thin, $\frac{1}{8}$ -in. diam disks of graphite. A 200- μ thick coating of dense PyC was applied to both sets of specimens. The coating on the disks will be examined to determine crystal orientation, isotropy, and density before they are used in the flash technique tests. The particles will be coated with W, fitted with thermocouples, and exposed to a neutron burst in Super Kukla.

Fuel Element for Transient Studies

A UHTREX experiment has been proposed in which UO₂ and ThO₂ particles would be combined in a single fuel element and exposed to a temperature-power transient. In order that the interaction between the two types of fuel can be studied, it is desirable that the influence of temperature changes in the graphite moderator be excluded. Therefore, an internally insulated element is being designed with the same external dimensions as a standard UHTREX element. The fuel is in the form of eight pellets, each 0.594 in. long x 0.422 in. diam, enclosed in a graphite case. Inside the case there is 0.164 in. of space for insulation around the pellets and end room for shock mitigation materials. Compressed graphite felt, originally 0.22 in. thick, was

chosen for insulation. A prototype element survived drop tests which simulated the fall of an element out of the reactor core. Since the mechanical design appears to be sound, the present configuration is being used as the basis for heat transfer and neutronic calculations for the transient experiment.

Fuel Elements for UHTREX Experiments

A series of experiments related to the safety of high-temperature gas-cooled reactors is planned for the UHTREX project. Included are physics perturbation experiments, transient experiments, Doppler experiments, HTGR mock-up tests, and ²³³U experiments. The fuel elements for the first four of these experiments will be fabricated at LASL.

Extrusion experiments are being conducted to form 1-in. diam x 5.5-in. long solid graphite elements of the same size with a small axial hole. The binder content and curing cycle have been varied in attempts to produce crack-free extrusions. To date, the best results have been obtained using 29 pph Varcum binder (a partially polymerized furfuryl alcohol resin) and the standard 63-h curing cycle. The knowledge gained from these tests will be used to develop an extrusion process for the highly loaded (ThO₂-UO₂) fuel element for the physics perturbation experiments. A maximum of 100 elements is required.

A fabrication process for the HTGR mockup tests will be developed using glass beads in place of the U. These fuel elements are also highly loaded with fuel and processes other than extrusion are being investigated. About 2000-3000 fuel elements will be required.

Fuel elements for the Doppler experiment consist of a core of ThO₂-UO₂ with ThO₂/UO₂ = 10. This core is to be encapsulated in thermal insulation which, in turn, will be placed in a graphite sleeve. The overall dimensions are 1 in. diam x 5.5 in. long; the fueled core is 0.3

in. diam x 4.9 in. long. Approximately 4000 fuel elements are required.

Two methods of fabricating the $\text{ThO}_2\text{-UO}_2$ fuel elements have been considered, extrusion and dry pressing, with the latter being favored for practical reasons at this time. A calcined coprecipitated product and a mechanical mixture of the two oxides are being considered as starting materials. Upon receipt of these materials, sample fuel elements will be fabricated and evaluated, and a decision will be made as to the best raw material for this application.

The transient experiments will use fuel elements with the existing UC_2 beads and loadings plus one dummy B_4C element. Little development work is anticipated for these elements.

Instrument Development

Thermocouple Life Tests

Thermal cycle tests on UHTREX core thermocouple prototypes are being continued. The test units are still functioning after 6250 h of exposure to thermal cycles between 1500° and 2500°F in an electrically heated graphite block under a He atmosphere.

Neutron Flux Monitor

Development and testing of the fission-couple thermopile device for use as a flux monitor in UHTREX are in progress. Improvements in the design of the device were made to avoid fabrication troubles. A Nb expansion joint was added to the thermocouple sheaths to overcome the extension and fracture of the 0.0005 in. diam thermocouple wires during welding of the seal cap. Ceramic cement, which had been considered necessary to support the beads and wires of the thermopile, was eliminated to permit the assembly to move under the force of thermal expansion.

Two devices were completed and one was subjected, without failure, to temperature cycles between 1000° and 1800°F in a furnace. Fol-

lowing this test, the device was exposed in the Omega West Reactor to a flux of 10^{11} neutrons/ $\text{cm}^2\text{-sec}$. The thermopile output, very low but real, established an apparent low limit of detectable flux. The device will be tested again in a region where the flux is 10^{13} neutrons/ $\text{cm}^2\text{-sec}$.

A second experiment to study the effect of bead size was performed in the OWR. The results of the experiment could not be compared with those of a previous experiment, which related bead size to thermoelectric output, but provided valuable information about the effects of heat loss through conduction. Large and small beads made of both $^{235}\text{U-Nb}$ alloy and pure Nb were tested. In the test device, the large beads were in contact with 20-mil diam lead wires, but the small beads were not. On exposure to 10^{13} neutrons/ $\text{cm}^2\text{-sec}$ flux, both sets of beads produced heat, but the size-emf output relationship was not consistent because the wires conducted heat away from the large beads. Performance of the pure Nb beads was observed for the first time. A temperature of 240°F was produced in a Nb buffer bead by γ -ray absorption and the effect of the thermal environment. This compares with the 684°F temperature of an adjacent fissile bead.

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MISCELLANEOUS ACTIVITIES

High-Temperature Neutron Detector Test

The testing of neutron detectors for use at high temperatures was undertaken as part of a now inactive program. The program required reactor startup counters that could operate at 600°C. It was believed that such a counter was available commercially and a calculated risk was taken by assuming that evaluation of state-of-the-art counters from several manufacturers would indicate at least one source of acceptable detectors. A pair of neutron detectors was purchased from each of three manufacturers for evaluation. The detectors have ²³⁵U-coated electrodes and integral, mineral-insulated leads (cables). Specifications require operation of both the detector and integral lead at 600°C.

Testing will determine the effect of temperature on the following: (1) pulse height distribution; (2) neutron counting sensitivity; (3) cable and detector resistance; (4) cable and detector capacitance; and (5) cable and detector noise.

Detectors with their integral cables are installed in four 17-ft high furnaces that contain a He atmosphere at a few psig. Electrical heaters on the outside of the furnace tube are wired so that those in the detector region and those in the cable region are independently controllable.

The integral cable is connected to a preamp which drives an amplifier whose output drives a scaler and a 400-channel pulse height analyzer. Ten minute counts are taken with and without a neutron source just outside the furnace insulation. Temperature is raised by 100°C increments. The testing schedule is shown in Table V. Some preliminary data have already been obtained and these and others will be reported when definite conclusions can be drawn.

Table V.
Schedule for Testing Neutron Detectors

Temp. °C		10-min counts			Measure R and C
		O. V. -	O. V.	O. V. +	
25-50	N	x	x	x	x
	n	x	x	x	
100	N	x	x	x	x
	n	x	x	x	
25-50	N		x		
	n		x		
200	N	x	x	x	x
	n	x	x	x	
.	.	.	.	.	.
.	.	.	.	.	.
.	.	.	.	.	.
600	N	x	x	x	x
	n	x	x	x	
25-50	N		x		x
	n		x		
600	N	x	x	x	x
(500 h)	n	x	x	x	

O. V. = Mfr recommended operating voltage

R = Resistance of detector and lead

C = Capacitance of detector and lead

N = No source counting

n = Neutron source counting.

Equation of State Data of Reactor Fuels

Fast reactor accident analysis requires equation of state information for fuel materials at high temperatures. Currently this information is obtained by extrapolation of available low-temperature data using an assumed model. Results of such an extrapolation are dependent on the model and there is little to indicate which model gives correct information.

Shock transient-time measurements have been used to calculate the equation of state for materials at high pressures and low temperatures. By reducing material density, the same technique may be able to give equation of state data of reactor fuels for temperatures and pressures of interest in reactor accident analysis.

A program is under way to investigate the feasibility of: (1) Fabricating reactor fuel samples

with a range of densities down to about one-fourth normal density. (2) Making shock transient-time measurements on materials of different density to find if the method will yield the equation of state in the range of interest.

UO₂ has been made having a density in the range from 2.9 to 10.9 g/cm³. Other fuel materials, and UC in particular, will be used in further studies.

Thermal Convection Loop Code

A thermal-hydrodynamic code (CONTAM) has been written to consider natural convection Na loops in steady-state condition. Debugging is nearly completed. Variable parameters include the following: (1) loop size and configuration; (2) heat exchanger number, size, position and type; (3) heating element number, size and position; (4) optional cold traps; (5) tee and elbow fittings; (6) system temperatures; and (7) system heat fluxes. The program will establish the relationship between Reynolds number and heat flux for a given physical apparatus.

Activation Analysis for Oxygen in Germanium

In the early manufacture of Ge(Li) radiation detectors, the success achieved in drifting Li was quite variable. It was suspected that the O content of the Ge ingot affected the drifting, but the determination of O in Ge by normal analytical techniques was difficult. Activation by ³He provides a very sensitive nondestructive analytical method for measuring O and C in matrices of high atomic number. Hence, in 1964 a piece of Ge that had been a good detector was irradiated to determine its O content. The Ge was etched immediately before irradiation and counted with a coincidence detector system after irradiation. The O concentration was estimated to be about 160 ppm. The sample was then analyzed by inert carrier gas fusion which gave about 300 ppm O. The difference is not

considered significant, because the activation analysis was not well calibrated.

At about the time of the foregoing experiment, the reliability of detector production improved, and so the investigation of correlation of O concentration with Li driftability was discontinued. However, the production of good detectors is still a function of some unknown and uncontrolled variable (or variables). Attempts by others to correlate driftability with impurity concentrations have been inconclusive. It is still suspected that the O concentration in the ingot may be important, but O measurements based on infrared techniques have not supported this conjecture. Since the infrared measurements presume chemical bonding of the O, there may be more O present than is indicated. To investigate this possibility, the ³He activation experiments have been resumed, and two Ge samples have been irradiated. Postirradiation etching techniques are being developed to minimize the effects of surface contamination, and so the concentrations measured should be minimum values.

A Ge sample, 678 E(35) 1-S, was irradiated for 1 h with 3 μ A of 6-MeV ³He ions. About 0.001 in. (or less) of material was etched off the irradiated surface with HF to remove surface contamination. An autoradiograph, resulting from exposure to ¹⁸F ($t_{1/2} = 110$ min) from the ¹⁶O(³He, p)¹⁸F reaction, was then taken. The O content of the Ge sample, which appeared to be uniformly distributed, was estimated to be 20 ppm, but with the uncertainty introduced by the etching it could be significantly larger.

HEAT PIPES

Fluid Flow Through Porous Wall Boundaries

Until recently it was thought that the theory of heat pipe operations, particularly the concept of vapor flow within the pipe, was understood rather well. It appears now, theoretically at least, that under certain ranges of radial Reynolds number a heat pipe may experience

Fig. 2. Heat-pipe assembly placed in orbit.

reverse-directed vapor flow, and there may even be some operating conditions where no steady-state flow is possible.

The full three-dimensional hydrodynamic equations of motion admit of certain similarity solutions. For these the basic differential equations are of third order, and are nonlinear to such a degree that it requires IBM digital computations to find solutions. A 7094 program has been coded that searches for these solutions by Runge-Kutta methods. This program, EIGFLO PROGRAM, is now operating, and it has recently confirmed one analytically known solution.

Zero-g Environment Test

The object of the zero-g experiment was to determine if a heat pipe will operate in the absence of gravitational forces. The necessary evidence to substantiate this fact was considered to be isothermal operation of a heat pipe while in an earth orbit. A water heat pipe was launched into orbit from Cape Kennedy on April 5, 1967 aboard an Atlas-Agena launch vehicle. The satellite module providing the telemetry, power supplies, and associated circuitry was supplied by the Lockheed Missile and Space Corporation under an Air Force contract.

The heat pipe was a Type 347 stainless steel tube 12-in. long and $\frac{3}{4}$ -in. o.d. containing 7.5 g of water. The wick structure consisted of three layers of 100 mesh stainless steel screen. Heat was supplied to the heat pipe by a 10-W electric heater wound around the pipe and soldered in place with pure Sn for good mechanical and thermal bond. Two stainless steel brackets, attached to an Al base plate, supported the heat pipe and formed a complete assembly as shown in Fig. 2. Resistance thermometers were used to measure temperatures along the axis of the heat pipe. The thermometers were coupled to a voltage divider network which provided an output from 0-5 V as a function of thermometer temperature. Since heat pipe operation is essentially isothermal, an end-to-end system calibration of the thermometers was made by operating at several different temperatures and plotting telemetry output voltage vs temperature as measured by thermocouples attached at each end of the heat pipe.

The satellite module consisted of the heat pipe, two batteries, transmitter, voltage control oscillator, electronics commutator, dc/dc converter, timer, antenna, and associated circuitry. The two batteries have a total capacity of 3500 W-h and were expected to provide an experiment life of 1.3 days. The transmitter had an output of 10 W and transmitted on a frequency of 228 MHz. A 60-point, 0.5-rps commutator was used to switch each of the data outputs to the telemetry system.

The heat pipe was launched on an Atlas-Agena vehicle as a piggy-back experiment with the ATS-A satellite. Upon separation of the Atlas from the Agena, a separation switch started a timer which turned on the heat pipe experiment 11,760 sec later. At that time, the Agena had completed its role in placing the primary payload in its orbit and had been deboosted by retrorockets so that the Agena's orbit would not coincide with that of the payload. The resulting orbit was elliptical with an apogee

of 6063 nautical miles and a perigee of 128 nautical miles with a period of 220 min.

Data were received at five different ground stations during 14 orbits after which battery voltage dropped too low for signal transmission. Although complete data have not been received and analyzed, the spot check information indicates that the heat pipe operated normally and that the zero-g environment did not affect its performance. There appears to be no reason why heat pipes using other fluids should not behave in a similar manner.

Vertical Heat Pipe Operation

The heat removal problem for power reactors requires heat pipes of large heat transfer capacity. A vertical heat pipe using gravity to augment the liquid return forces appears to be interesting; however, the heat must not be applied to the pool of working fluid since this would result in instabilities with slugs of liquid being pushed up the tube. In addition to leaving a pool below the evaporator section, the use of an inert gas column to assist in startup is being investigated.

A 2.3-cm o.d. x 315-cm long heat pipe has been operated vertically with Na as working fluid. The pipe was heated with a 62-cm long induction coil. A 28-cm section extended below the coil was insulated and the remaining 135-cm length was left bare so that it could radiate and be cooled by air convection.

A fine screen at the container-liquid interface was used to wick liquid from the pool into the evaporator as it was needed for heat transfer. The entire length of the heat pipe was lined with three additional layers of progressively coarser screen toward the liquid-vapor interface to provide increased surface for vapor condensation and a controlled path for liquid return.

The heat pipe was started with an internal He pressure of 20 torr. This gas prevented the vapor from leaving the evaporator until its temperature was 600°C. The input power was then increased in steps to check the behavior of the system. During these increases, the vapor moved through the insulated section compressing the He ahead of it. When the vapor entered the cooled region, its temperature was 635°C. Eventually, the He was compressed into the top 15 cm of the pipe and the remainder of the system was essentially isothermal at 825°C. The heat removal rate due to radiation and convection was estimated to be 8 kW. Although the pool diminished with each power increase, 15 cm remained, thus indicating a greater system capability. Provisions have now been made for forced cooling of the condenser section and measuring heat removal rates with a calorimeter.

A second test was run with the same He pressure, but with the insulation removed to allow greater heat transfer rates at lower vapor temperatures. During startup, the pool diminished normally as the power was increased and steady-state conditions were achieved until the system reached 650°C. At this time, the pool level continued to fall at constant power. When the pool disappeared, the evaporator section overheated near the bottom. Although the maximum rate of heat removal was only about 4 kW, the vapor velocity was about 600 ft/sec compared to 175 ft/sec in the first test.

A third test was run, again without insulation, after the system was evacuated to remove the He gas. The vapor was free to move throughout the system at very low density. The pool disappeared in this test while the vapor was moving through the system at temperatures less than 500°C. Since steady-state conditions could not be attained at any temperature, heat transfer and vapor velocity estimates were not made.

Although these tests demonstrate that high-velocity vapor can interact with returning liquid, past experiments have shown that the interaction can be reduced by separating the liquid flow region from the vapor passage with a fine-mesh screen. It is likely that fewer difficulties would have been encountered in the tests if this had been done. However, some interaction was desired to show the effectiveness of non-condensable gas control. This gas makes the heat pipe less dependent on its environment since it limits heat removal while the vapor density is low. Moreover, it assures a shorter liquid return path during startup, so that the total system pressure drop is lower than in an evacuated system. This is particularly important in a horizontally operated heat pipe since liquid must be returned by capillary forces.

It should be stated that gas control can be necessary when a working fluid has a significant vapor pressure at its melting point. This has been demonstrated by starting a heat pipe with water from -60°C . Without gas, the system temperature increased isothermally until the ice has sublimed out of the evaporator, but then this section overheated since there was no liquid return. When a small amount of H_2 was added, the ice melted before it could leave the evaporator and the vapor front moved through the condenser at temperatures sufficient to melt the water.

Fabrication Development

Five heat pipes were made from grooved Type 347 stainless steel sheet. The starting material was 0.060-in. thick with 100 grooves 0.069-in. apart and 0.030-in. deep $\times$ 0.012-in. wide. The edges of the grooved sheet were machined to a 14° angle and the sheet formed into a tube with the grooves on the i.d. and welded. New tooling has been received and checked out to form heat pipes 0.710-in. to 0.750-in. o.d., 0.040- to 0.060-in. wall, and 4-ft long; the maximum length which can be

formed with the old dies was 12 in. An empirical formula, $D = 0.330(W)$ in., relating the width (W) of the flat sheet to the o.d. (D) of the finished tube, was derived from data accumulated during the manufacture of heat pipes. By means of this formula the width of the starting material can be calculated for a required tube diameter.

PLASMA THERMOCOUPLE RESEARCH

The results obtained during the current quarter on plasma thermocouple research are classified and are described in the ART Report Part II, LA-3709-MS (Confidential).

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