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LOS ALAMOS • NEW MEXICO

Quarterly Status Report on
Advanced Reactor Technology (ART)
for Period Ending October 31, 1966

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

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FOREWORD

In view of the termination of the molten plutonium fuel program at the Los Alamos Scientific Laboratory the format of the Advanced Reactor Technology (ART) Report has been changed. Part I will now deal with UHTREX, Part II with general studies relevant to Sodium Cooled Reactors, Part III with the Plasma Thermocouple, and Part IV with Heat Pipes. Some work on molten plutonium fuels is still under way and until it is completed or phased out the results will be given in Part V. Studies of ceramic plutonium fuels will be reported separately as part of the LASL Advanced Plutonium Fuels Program.

I. UHTREX

Program Status

Installation of UHTREX plant and instrumentation systems is nearly complete. Check-out and final clean-up of some systems is under way. Analytical studies of the planned program of transient experiments have begun.

Reactor Components

Reactor Pressure Vessel

The top cover is in place on the reactor pressure vessel and preparation for a pressure test of the reactor and coolant loops is nearing completion. Figure 1 shows the final stage of bolting down the cover.

Fuel Handling System

All components of the fuel handling system have been delivered and installation is progressing well. The charge conveyor tube is installed between the drybox and the fuel loader room. The three ball valves that make up the inlet gas-lock have been assembled and

Fig. 1. Bolting down of cover of UHTREX Pressure Vessel.

tested, and are ready for installation. Below the reactor, the outlet gas-lock is in place, and about half of the discharge conveyor tube is mounted.

Components of the fuel loader are being assembled and tested in preparation for mounting on the pressure vessel.

Coolant Systems

Piping

All major components and piping of the primary and secondary coolant loops are complete. The installation of thermocouples, pressure taps, and sample lines will be finished soon.

Air has been circulated, in the reverse direction of flow, through the primary loop to purge the system of loose dust and small pieces of carbon formed when assembling the internal parts of the reactor and recuperator. This was accomplished by connecting the discharge side of the primary loop gas-bearing blower to the heat exchanger outlet. The air flow passed through the heat exchanger, then through the recuperator hot side, reactor, recuperator cold side, and was finally exhausted to the atmosphere. An absolute filter was installed at the blower intake to prevent the entry of particles from the atmosphere.

Circulation was maintained for approximately 1 h after the blower reached maximum speed. The maximum fluid velocity during the test was 120% of the He velocity at normal, full-power reactor operation.

At the start of the test, a small cloud of dust was observed at the system exhaust. A few particles, mostly of dust size, were exhausted intermittently during the first 40 min of the test; none were observed during the last 20 min. One particle ejected was about 0.5-in. diam.

Auxiliaries

Pneumatic pressure tests were completed on the entire demineralized water system which cools the blowers and the water panels in the reactor room. Wet tests of the systems will be run soon.

Instrumentation

Control Room

All of the graphic panels are installed in the control room. The reactor operations console is complete, the annunciator panels are in place, and the fuel system panel has been installed. Check out of the "operate panels" for the gas cleanup system is progressing well. The chassis for the reactor and coolant loop safety systems and for the health physics systems are installed. All of the nuclear instrumentation is complete and has been tested at the control room end.

Digital Computer System

The 180-day reliability test of the digital computer was completed and the system was accepted. The computer is currently running with a basic set of on-line programs. These include the executive and operations console programs, analog scan, digital scan, annunciator scan, logging programs, alarm typewriter program, GCS matrix input, analog output, interlock, and the essential subroutines. The fuel loading program ran successfully and has been debugged with the aid of a simple mockup consisting of several valves, the core indexing mechanism, loading rams, and conveyor.

Planning for Transient Experiments

Introduction

UHTREX can be used for studying the safety characteristics of high-temperature, gas-cooled reactors (HTGRs). In such a program, a

planned series of pulse-type transients would be induced in UHTREX by using special equipment for achieving rapid control rod withdrawal. By properly tailoring the fuel loading for these experiments, the responses observed in UHTREX should provide data on the principal temperature coefficients and nonequilibrium heat transfer effects that influence the transient behavior of the HTGRs generally.

This program will require all of the calculational procedures already prepared for UHTREX. In addition, some new procedures, including methods of treating the resonance cross sections in Th, the spatial effects in transients, and the transient effects of heat transfer between coated particles, must be developed.

Fuel Particle Design

The course of nuclear transients in HTGRs will depend on details of the fuel elements. In planning the initial UHTREX transient experiments, the primary aim is to include all the important characteristics or mechanisms that may influence the response of HTGRs to fast excursions, rather than to mock up a particular fuel element design. The results from the initial exploratory experiments and other future developments may suggest additional transient tests in which effects of certain of the characteristics are emphasized.

The principal mechanisms affecting the response of HTGRs to fast transients are: (1) The Doppler temperature coefficient in Th. (2) The thermalization temperature coefficient associated with the heating of the moderator in the fuel element. (3) Time delays associated with the transfer of heat from the fissile material (U) to the fertile material (Th) and to the moderator (graphite) in the fuel element. All these mechanisms can probably be provided in the proposed fuel element design.

The best particles for the initial UHTREX transient experiment are most likely those proposed for the Public Service Company of Colorado Reactor (PSC). These spherical particles are of two types¹ described below.

1. Fissile Particle

Center:	200 μ o.d., UC_2 (Th:U = 4:1)
Buffer layer:	260 μ o.d., 1 g/cm ³ graphite
Outer layer:	400 μ o.d., 2 g/cm ³ graphite

2. Fertile Particle

Center:	350 μ o.d., ThC_2
Buffer layer:	410 μ o.d., 1 g/cm ³ graphite
Outer layer:	610 μ o.d., 2 g/cm ³ graphite

One change from the PSC specifications is proposed for the UHTREX application: it is planned to fabricate the central part of the fissile particle from UC_2 only instead of from a mixture of ThC_2 and UC_2 . Unless this change is made, the ratio of fertile to fissile particles in the UHTREX loading would be relatively small, and the effect of the time delay associated with the transfer of heat from fissile to fertile particles would be inappropriately deemphasized.

Preliminary calculations have been made of the critical loading and Doppler coefficient at operating temperature (1714⁰ K). These calculations were performed with the DTF code in S_4 approximation using a 1-D cylindrical, 14-zone description of UHTREX with buckling to account for axial leakage.

The critical loading at operating temperature was calculated to be 14.9 kg of ^{235}U and 63.4 of ^{232}Th ; 60 v/o of the fuel element would be occupied by the coated particles. For the two types of coated particles described above, this loading corresponds to a 1:1 ratio of UC_2 to ThC_2 . Self-shielding in the fuel element was not taken into account in this

estimate of the critical loading; the self-shielding could increase the critical mass by about 10% over that given here.

The Doppler coefficient from the Th was calculated to be $-5.2 \times 10^{-6}/^{\circ}\text{C}$ at operating temperature.

Carbon Studies

Corrosion Test Loop

Early in the UHTREX project, Grade B204 insulating carbon was heated in the corrosion test loop. During this test, outgassed volatiles were deposited throughout the loop. Subsequent small-scale experiments in furnaces established that the source of the volatile material was the insulating carbon. Comparable tests on Grade 45 porous carbon indicated that it had a much lower content of volatile material. However, large-scale experiments with Grade 45 (which is used as insulation in the UHTREX vessel) have not been conducted until recently.

In these recent tests, more than 100 lb of Grade 45 porous carbon were loaded into the corrosion loop and heated in the usual manner by the carbon resistance rod. The heat-up period was controlled in order to limit the amount of gaseous contaminants, such as CO_2 and H_2O , which accumulated in the He coolant. The heat-up period took 3 weeks; a final temperature of 3065°F (at the center of the heater rod) was reached in five steps. The H_2O level rose to about 100 ppm; this maximum was reached when the maximum temperature in the heater channel was approximately 2000°F . The CO_2 level remained below 10 ppm throughout the test.

Methane and H_2 concentrations in the loop were measured with batch samples collected two to three times daily. The CH_4 analysis indicated a maximum value (1.5 ppm) in the first sample; thereafter it was usually zero with an occasional 0.5 ppm value. The validity of the H_2 analysis is in doubt; a level between 66 and 78 ppm was indicated, continuously, but with the particular chromatographic column that was in use, the O_2 and H_2

peaks coincide. A mass spectrometer analysis showed the presence of both H_2 and O_2 but not their concentrations. It is believed that most of the H_2 indication resulted from O_2 which was accidentally introduced into the chromatograph system by the batch sampling technique.

When the test loop was opened no deposits were found in any part of the loop. There was no carbon deposit on either the resistance rod or the channel walls, nor were any salts found in the heat exchanger. The loop interior was clean.

It is concluded, therefore, that the UHTREX primary coolant loop will not be contaminated with deposits from the porous carbon insulation if the contaminant levels are limited by controlling the heat-up rate of the reactor. The clean-up capacity of the loop will then be greater than the rate of generation of contaminants.

Reference

1. Private communication from R. Dahlberg, General Atomic Division of General Dynamics.

II. SODIUM-COOLED REACTORS

Introduction

Several problems of general and continuing interest in the sodium-cooled reactor program are being studied. The work includes instrumentation testing and development, heat transfer investigations, and formulation of computer codes for treating the dynamic response of entire reactor plants.

High Temperature Instrumentation

Neutron Detectors

As a part of the Fast Reactor Core Test Facility instrumentation program, evaluation of high temperature neutron counters supplied by five different manufacturers had been planned. Ten counters (two from each vendor) were ordered and the evaluation will be made when they are delivered. Two of the ten are on hand. Four temperature controlled furnaces and two counting channels have been installed for these tests. The evaluation will include the measurement of neutron sensitivity, gamma sensitivity, cable resistance, and cable capacitance as a function of temperature up to 1200°F.

Pressure Transducers

Proposals for supplying pressure transducers for Na service have been solicited from two manufacturers. One proposal has been received. Both companies have a line of pressure transducers utilizing principles of operation new to the Na instrumentation field. The proposals will be evaluated and, if satisfactory, prototype transducers will be purchased for testing.

Linear Induction Flowmeter

The possibility is being investigated of applying linear induction techniques to a liquid-metal flowmeter. The scheme is based on the matching of linear induction wave velocity to Na velocity. The method is theoretically sound but may be difficult to apply in a practical instrument. The primary advantage is that flow velocity is measured directly by a null balance method in which temperature does not affect the null.

Coils for a bench test have been wound and put into a wooden stator form. A current problem is the procurement of a simple and inexpensive variable-frequency, 3-phase power supply.

Sodium System Technology

Heat Removal by Natural Convection

Calculations of core-to-coolant heat transfer following loss of pumping are hindered by absence of heat transfer data in the natural convection flow regime. Limits on the parameters which could be determined by experiment in a natural convection heat transfer loop must be established by theoretical analysis. To this end, a steady-state natural convection code has been written to simulate the proposed loop. The code calculates equilibrium flow rates, pressures, and temperatures as functions of variables such as piping diameter and heat input to the test sample. Sample calculations have been made which include as parameters loop size, piping diameter, core test module flow characteristics, heat flux, and control valve characteristics. No further calculations are planned at this time; however, results of work done to date will be available should further interest develop in a natural convection.

Misalignment Effects in Reactor Fuel Assemblies

Design of an experiment is under way to measure the velocity field in a bundle of cylindrical pins spaced on a triangular pitch. The

following three questions have been investigated:

1. How many pins must be provided in a bundle so that the middle pin will simulate a pin in an infinite bundle? Velocity distribution calculations for laminar flow (in both finite and infinite pin bundles) have been made using the codes described by Axford.^{1,2} It is concluded that the central pin of a 7-pin bundle will not simulate one in an infinite bundle and it will be necessary to provide an assembly of at least 19 pins.

2. What is the maximum diameter of an impact probe in relation to the flow channel width so that perturbation of flow by the probe can be neglected? It is thought that the impact probe diameter should be no larger than about 10% of the flow channel width. The basis for this opinion is not very firm, however, and it is recommended that at least three different diameters be tried to attempt to determine the effect of probe diameter.

3. What entrance length must be provided to assure that flow is fully developed? For heat transfer to liquid metal in turbulent flow, it has been reported³ that the entrance effects are negligible after about 35 hydraulic diameters downstream. For laminar flow, an entrance length of $L_e = 0.035D_e Re$, where D_e is the hydraulic diameter and Re is the Reynolds number, is recommended.^{4e} It appears that velocity distribution measurements should be made at various distances downstream from the inlet to the tube bundle. The results should establish the appropriate site for making valid measurements of the velocity field.

It is proposed to measure velocity distributions for three pitch-to-diameter ratios; 1.1, 1.2, and 1.3. The initial measurements will be made in a finite bundle of seven tubes in both laminar and turbulent flow with no misalignment. The laminar flow measurements can then be compared with the velocity distributions predicted by the method of Axford.² For a pitch-to-diameter ratio of 1.1 and an impact probe

diameter of 0.030 in., the test section will consist of seven 3-in. diam pins on a 3.3-in. triangular pitch inside a 10-in. pipe shell. Length of the test section will be about 10 ft. Laminar flow measurements will be made in the 150-gpm H₂O loop.

Boiling Heat Transfer

A survey of the literature on boiling heat transfer and two-phase flow, initiated on July 1, 1966, has been completed. Review of some of the more recent publications has revealed that researchers in this area have turned to a study of the basic mechanism of boiling. These studies have shown that the conventional concept of nucleate boiling upon which all prior correlations were based is not correct. Three and possibly four regimes of nucleate boiling have been demonstrated, with only the low heat flux range corresponding to the ideas held previously about nucleate boiling. The new views and their consequences have been explored; a summary of the literature and specific recommendations for future work have been prepared.

Reactor Plant Dynamics

Further alterations have been made to the fast reactor plant dynamics computer code (formerly CTFDYN; now called RAPID, for Reactor and Plant Integrated Dynamics). Six delayed neutron groups are now considered (three were used formerly); printouts may be demanded at any time interval (previously 1 sec was the minimum); and information pertaining to the neutron kinetics is printed when the program judges the approximations made in the neutron kinetics calculation to be invalid.

A comparison has been made of calculations performed by RAPID, (with a good heat transfer model but a poor neutronics model), and the modified ANCON⁵, which employs a good neutronics model but a poor heat transfer model. The results are shown in Figs. 2, 3, and 4. The agreement

Fig. 2. Reactor power following \$5/sec ramp.

Fig. 3. Reactor power following \$1/sec ramp.

Fig. 4. Reactor power following 20¢/sec ramp.

of the curves is satisfactory, but two details were somewhat unexpected. First, it was considered that the calculation resulting in curve A of Fig. 2 would be stable for a time step of 0.0561 sec, but it was not. Second, the neutron kinetics approximation was declared by the code to be invalid beyond 0.2 sec for the calculation leading to curve B of Fig. 2, yet that calculation appears to be satisfactory up to 10 sec. These anomalies were eventually traced to two errors in the neutron kinetics portion of the program. That portion has been removed to a separate subroutine, which allows consideration of much smaller time steps in the neutron kinetics calculations without appreciably increasing total computation time. A separate neutron kinetics subroutine is the first step toward possible inclusion of a modified ANCON program in RAPID. At present, it appears that such inclusion will cause the storage requirements for RAPID (with ANCON) to exceed the capacity of the IBM-7094, but these problems could be run on the CEC-6600.

A 10-sec, real-time problem (Fig. 2) requires about 10 sec of IBM-7094 time for the coarse (two heat transfer node) option of RAPID, 4 min for the detailed (25 heat transfer node) RAPID calculation, 10 sec for the modified ANCON calculation (which is restricted to $\rho \ll \beta$), and 50 min for the exact ANCON calculation. The exact ANCON calculation is, nonetheless, faster than any other neutronics calculation of comparable accuracy.

Helium Cover Gas Cleanup System

Work on a He cover gas cleanup system consists mainly of the procurement and testing of a hermetically sealed gas compressor (pump) and a He diffusion cell. The compressor is a resonant piston type. The pump specification calls for a compression ratio of 2.0, but the highest value achieved by the vendor is 1.8. A flow of 0.4 scfm is produced with an inlet pressure of 50 psia and an outlet of 72 psia.

The He diffusion cell consists of a shell and tube arrangement of SiO_2 capillary tubing in a pressure vessel. At operating temperatures (about 750°C) the SiO_2 walls are slightly permeable to He but to no other gas. Such a cell can therefore be used to purify He.

The 3000-tube $4\frac{1}{2}$ -ft diffusion cell has been operated at temperatures between 480° and 760°F at a constant differential pressure of 165 psi. The He flow rate has been measured at each temperature to establish the diffusion coefficient of the cell. This coefficient was observed to be three times as large at 760° as at 490°F .

A 500-tube prototype diffuser using a new, all-epoxy tube bundle seal has been received from the contractor; it will be tested as soon as studies with the 3000-tube unit have been completed.

References

1. Axford, R. A., "Multiregion Analysis of Temperature Fields and Heat Fluxes in Tube Bundles with Internal, Solid, Nuclear Heat Sources," LA-3167, Los Alamos Scientific Laboratory, 1965.
2. Axford, R. A., "Longitudinal Laminar Flow of an Incompressible Fluid in Finite Tube Bundles with $m+1$ Tubes," LA-3418, Los Alamos Scientific Laboratory, 1965.
3. Kalish, S. and Dwyer, O.E., "Heat Transfer to NaK Flowing in-line Through a Bundle of Circular Rods," ASME Paper 65-HT-11.
4. Bird, R. B., Stewart, W. E., and Lightfoot, E. N., "Transport Phenomena," John Wiley & Sons, Inc., p. 47.
5. Vigil, J. C., "Solution of the Nonlinear Reactor Kinetics Equations by Continuous Analytic Continuation," LA-3518, Los Alamos Scientific Laboratory, 1966.

III. PLASMA THERMOCOUPLE

Cesium Vapor Pressure Experiment

Fifty-six measurements of Cs vapor pressure were taken as a function of temperature using a heat pipe which was connected to a pressure measuring system. Argon was used as an inert gas fill for the heat pipe, and pressure measurements were observed from 52.7 down to 0.10 torr.

Comparison of these new data with the commonly accepted values¹ of the vapor pressure of Cs is given below. The chromel-alumel thermocouples used on the Cs heat pipes are yet to be calibrated.

<u>Vapor Pressure,</u> <u>torr</u>	<u>RCA</u> <u>T(°K)</u>	<u>LASL</u> <u>T(°K)</u>
0.5	526	520
1.0	550	543
10.0	646	635
20.0	681.4	670
50.0	733.6	722

Triple Layer Evaluation

Investigation has been reopened into the evaluation and construction of the triple layer materials surrounding a thermionic fuel rod. Desirable characteristics of this triple layer are that it reject heat with a $\Delta T \sim 50^\circ\text{C}$ at fluxes of 50-100 W/cm², and that it support a potential of 100 V at 700°C.

Past experience has shown that specimens manufactured by plasma spraying Al_2O_3 on Nb-1Zr tubing and then magnaforming (magnetic swaging) a cover tube on the outside have a thermal contact resistance between the metal and the ceramic that is too high to perform satisfactorily under the required conditions. Evaluation has been made on four tubes which had plasma-sprayed 325 mesh Nb powder substituted for the Nb magnaformed tubing. The steady state thermal conductivity of Al_2O_3 in these four specimens was on the average $\sim 0.5\%$ of the published thermal conductivity of dense Al_2O_3 , and about a factor of 20 better than the thermal conductivity of the magnaformed specimens. Post test examination of the sprayed material showed it to be about 90% theoretical density.

Sections of these tubes have been submitted for transient thermal analysis by the laser technique to verify the values of the thermal conductivities obtained by the steady state method. Sections of the tubes are also being polished for microhardness testing, and photomicrographs will be made of the test sections.

Attempts have been made to plasma-spray 10 mils of Al_2O_3 on a typical collector-insulator-base configuration. The first three test pieces leaked and subsequently were destructively examined to determine the cause of the leaks. It was found that the insulator in the region of the V braze joint was being fluxed by the Zr present in the system, thus creating a glassy phase in the insulator which contained some connected porosity.

Specimens of vapor-deposited Al_2O_3 and BeO on Nb-1Zr tube have been purchased from commercial vendors to evaluate the thermal conductivity of this type of ceramic.

Thulium Oxide-Molybdenum Cermet

The number of potential isotopic sources with the capability of reaching thermionic temperatures and power densities is very limited. One such candidate is ^{170}Tm in the form of the oxide. To increase the

thermal conductivity, the Tm_2O_3 may be combined with Mo as a cermet. The advantages of constructing the cermet from normal (stable) ^{169}Tm (100% abundance) in the form to be used and producing the ^{170}Tm by irradiation in a high thermal neutron flux are obvious.

A minor effort is being made to investigate some of the fabrication problems and stability characteristics of this cermet. Samples of Tm_2O_3 -Mo cermet cut from a cold-pressed and sintered block will be heated in vacuum electrically for 1000 h at 1700°C to establish material compatibility. These samples, which are $\frac{1}{4}$ in. in diam and $\frac{1}{2}$ to $\frac{3}{4}$ in. in length were coated with 2 to 3 mils of vapor-deposited Mo to prevent evaporation of the oxide from the surface porosity of the cermet. Another batch of Tm_2O_3 -Mo samples have been coated with 2 to 3 mils of vapor-deposited W. Samples from this batch will be heated for 1000 h at 2000°C . Weight and dimensional changes will be determined and micrographic examination will be made.

Fuel Pin Fabrication

Seven 60 v/o Mo-40 v/o Tm_2O_3 pins were prefired at 1800°C for 5 min in vacuum and coated with W by H_2 reduction of WF_6 at 450°C . The coatings were stress-relieved at 1200°C for 20 min in vacuum and the pins delivered for irradiation experiments.

Cell Design Calculations

Further calculations have been made using the computer code for determining the performance of a cylindrical diode. Results have been obtained using a set of current voltage characteristics obtained by TEECO^2 for a W emitter, Mo collector diode operating with a 7-mil spacing and a Cs reservoir temperature of 598°K . For this diode the maximum efficiency for a 1.2-cm diam emitter occurs for an emitter length of 4

cm when the maximum temperature is 2100°K . The maximum specific power of an epithermal reactor system occurs for an emitter length of 3 cm. Although the TEECO diode has a lower performance than the Wilson-Lawrence diode, the qualitative behavior of maximum power output, maximum efficiency, and maximum specific power with emitter length remains the same.

Calculations have also been made for a series of 15 thermionic diodes all carrying the same current, but with a symmetric fission power distribution varying as a cosine curve. The Wilson-Lawrence current voltage characteristics were used for an emitter length of 4 cm and the central cell was constrained to a maximum temperature of 2100°K . For the case in which the fission power of the outer two cells is 80% of that in the central cell, the emitter support resistances can be adjusted so that the maximum efficiency and the maximum power output are 95% and 88%, respectively, of the same quantities for a flat power distribution (for a 4-cm emitter length).

References

1. R. E. Honig's data from RCA Rev. June 1957 was fitted to a four-parameter equation.
2. S. S. Kitrilakis, M. E. Meeker, and N. S. Rasor, TEE-4015-3, June, 1962.

Several papers related to the Plasma Thermocouple program were presented at the IEEE meeting at Houston, Texas, Nov. 2-4, 1966. The titles and references will be reported after publication.

IV. HEAT PIPES

Heat Pipe Capability Experiments

Heat pipe performance will usually depend to a large extent on wick configuration since other factors, such as heat pipe size and choice of working fluid, will often be determined by the specific application. Several wick improvement experiments have been conducted with Na and K working fluids in heat pipes having a diameter of 1.8 cm and a length of 30 cm.¹ The results of these experiments are presented in a report which describes heat pipe construction, final preparation, and the experimental procedure.¹ Pertinent fluid properties are given for Na and K as a function of temperature. Heat pipe equations are listed and the experimental results are compared with calculated heat transfer limits. General heat pipe behavior is discussed with particular emphasis on startup and performance limitations.

Tests with 30-cm long heat pipes were discontinued after the heat transfer capability of some of the improved systems exceeded the 4000 W which could be removed with the experimental equipment. At present, some 80-cm long heat pipes are being tested. Because of the increased area available with these pipes, it should be possible to achieve heat removal rates in excess of 10,000 W before equipment limitations occur. These longer systems also provide greater resistance to fluid flow so that substantial wick improvements should be required to attain high heat transfer rates.

The first 80-cm long heat pipe tested with Na as a working fluid showed a heat transfer limit of 5500 W at 850°C. The wick consisted of 39 axial channels, approximately 30 mils wide x 30 mils deep,

evenly spaced around the inside circumference of the heat pipe tube. The channels were covered with a single layer of 100-mesh screen. Although measured heat transfer limits were higher at all test temperatures than those predicted for open channels, the screen cover was not nearly as effective as was found in earlier tests. Examination of the heat pipe after the test showed that the screen had buckled away from the wall in several locations and no longer covered the channels. This heat pipe was formed by joining three tubes and the internal alignment was not ideal. Screen insertion was difficult and some screen distortion probably occurred during this procedure. Attempts are now being made to solve the assembly problem.

A useful feature of the tests with longer heat pipes concerns the determination of heat transfer limits with different length evaporator and condenser zones. This is accomplished by insulating one end of the heat pipe over a significant section so that it is essentially inactive. Heat is added to a portion of the pipe next to the insulated section and removed from the remainder of the system. During operation, the heat pipe is moved so that more of it becomes active. Heat transfer limits can be determined at various locations and the procedure repeated with heated areas of different sizes to show the influence of heat pipe length and heat input flux.

Tungsten Cylinder Fabrication

Tungsten, 0.03 in. thick, was deposited in 12-h coating time onto a Kovar mandrel $\frac{1}{8}$ -in. diam by 9-in. long by the reduction of WF_6 . Kovar has approximately the same coefficient of thermal expansion from 0-400°C as does W; thus its use minimizes the stresses produced in the W coating during heating and cooling. The Kovar will subsequently be removed by dissolution and the hollow W cylinder will be used in heat pipe experiments.

Reference

1. Kemme, J.E., "Heat Pipe Capability Experiments," LA-3585-MS, Los Alamos Scientific Laboratory, 1966.

V. LIQUID PLUTONIUM FUELS

Irradiation Experiments

Introduction

To study the effects of thermal neutron irradiations on liquid Pu fuels and their containers, fueled capsules have been irradiated in the 6.5 MW Omega West Reactor (OWR) at LASL. This program is now nearly complete. The Omega West Reactor Experiment (OWREX) in-pile apparatus consists of two separate parts: first, a semi-permanently installed environmental cell, which contains a naturally convecting Na loop, thermocouples, heater units, and a gas annulus (heat leak) for controlling heat removal and temperature; and second, a fueled and instrumented test capsule insert which is installed in, and removed from, the environmental cell as a unit. At present, there are two environmental cells in the OWR. Several inserts containing fuel have been irradiated.

Program Status

On Aug. 30, the fission gas trap for the OWREX-8 capsule showed a slight increase in activity. Because this increase was small, the capsule was kept in operation until Sep. 2, when the activity increased rapidly and the reactor was shut down. OWREX-8 was removed on Sep. 19; the fuel had accumulated about $2\frac{1}{4}\%$ burnup since installation, Jan. 31, 1966.

The capsule from OWREX-8 (No. 1082) is being gamma-scanned with slit and pinhole collimators, using both 400-channel and 4096-channel pulse-height analyzers. It appears that there is an average of about 50% Ta wall thinning in the region of the fuel-gas interface.

Another capsule, OWREX-9, was installed in the reactor on June 27, 1966. This contains 32 g of Pu instead of the usual 24 g. In addition, the insert was modified to contain additional thermocouples so that the behavior of the volatile fission products could be determined during reactor operation by measuring changes in the fuel column height. An initial sharp decrease in height was observed and is attributed to the "percolation" of cover gas which had been trapped in the initial fuel melting. Equilibrium fission gas bubble content of the fuel during irradiation appears (on the basis of fuel height measurements) to be 3 or 4%.

On Oct. 21, the He gas flow in OWREX-9 (which sweeps over the fuel capsule and is monitored for radioactivity) stopped. After it was determined that radioactive gas had escaped from the capsule, the experiment was terminated by turning off electrical heat to the cell and scrambling the reactor. It is believed that the sweep gas line was plugged with Na which was injected by sudden release of gaseous fission products from the test capsule.

Container Studies

Metallurgical Development

"DR" Ta sheet is advertised to be fine-grained due to controlled fabrication procedures and to retain the finer grain size to some extent regardless of further processing. A program was established to determine the effects of heat treatment, welding, and cool working on the structure.

Samples of this material were heat treated in vacuum for 1 h at temperatures between 1400° and 1800°C. Slight grain growth was observed at 1600° with pronounced growth occurring at 1700°C. Standard Ta given a similar heat treatment exhibits extensive grain growth above

1500°C. Material was cold worked (~20% reduction) and subjected to similar heat treatments. Some grain growth occurred at 1400°C and increased with temperature through 1800°C; the hardness of the as-rolled material dropped from 185 to 125-135 DPH for the vacuum heat-treated material.

These investigations did not indicate any significant advantages of the "DR" type Ta over standard Ta sheet. Therefore, further study of this material has been discontinued.

Three Nb base alloys were being evaluated for capsule materials. Tubes of these alloys, F-85 (Nb-27W), C-129Y (Nb-10Hf-10W), and K-719 (Nb-10W), were fabricated and evaluated by nondestructive testing. The K-719 (Nb-10W) alloy gave the highest yield of sound parts as expected, since this exhibited the best formability and weldability of the alloys examined.

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