**International Symposium
on Developments in
Irradiation Capsule
Technology**

**Held at Pleasanton, California
May 3-5, 1966**

**United States
Atomic Energy Commission**

Division of Technical Information

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Pleasanton, California

Edited by

Donna R. Hoffman

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PREFACE

The papers assembled herein represent, in essence, a progress report on advances in irradiation capsule technology since the first meeting on this subject, held on October 8, 9, and 10, 1963. The report on the previous meeting is entitled "Symposium on Problems in Irradiation Capsule Experiments," TID-7697, issued March, 1964. Although the title is somewhat different for this edition, the TID number has been retained since the report is on the same subject.

The Preface to the first edition made a hopeful statement to the effect that future meetings should not have to cover a large volume of material. This has not been the case. If anything, the response to the second meeting was greater. This indicates that irradiation capsule technology is still in the growing phase and improvements and refinement in techniques can still be anticipated.

Perhaps the most pertinent observation to be made on the meeting is that it showed that irradiation capsules must generally be of custom design. Because of variations in fuel composition and enrichment and in cladding composition and thickness, it is simply not possible to employ a "standard" design for all irradiation experiments. In some instances, such as irradiation of standard tensile specimens, this is possible. However, for most irradiation programs, the experimenters must start from scratch. It is believed that publication of this report should enable the designer to benefit from the past mistakes of others and to apply engineering design principles that have proven to be valid.

The program of the meeting is included as an Appendix to this report. For various reasons, not all the papers presented have been printed herein. Should some of these papers become available for publication, they will be issued as a Supplement to this report.

The meeting reported herein was jointly sponsored by the General Electric Company and the Atomic Energy Commission. The credit for the high quality of the meeting should be given to the personnel from the General Electric Company who planned the meeting and insured its smooth operation. Messrs. B. C. Meyer, L. E. Wilkinson, and G. E. Robinson, among others, devoted a great deal of their time to the meeting. This is greatly appreciated.

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TENSILE COUPON IRRADIATION TECHNIQUE

By

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This work was performed under Contract No. AT(04-3)-189 Project Agreement No. 17
between the Atomic Energy Commission and the General Electric Company.

TENSILE COUPON IRRADIATION TECHNIQUE

An irradiation technique was devised to obtain high levels of neutron exposures in fuel rod clad materials by utilizing space on an existing irradiation experiment. The UO_2 High Performance Program, PA 17,⁽¹⁾ had been using the Pressurized Water Loop of the General Electric Test Reactor to obtain information on the functional limits of operating UO_2 with gross central melting. Figure 1 shows the fuel rod assembly that was used for this purpose. The assembly consisted of four shroud tubes held together in a tie plate at the top and secured at the bottom in a lower nose piece. Each shroud tube contained a fuel rod under test.

A recommendation was made to attach tensile coupons to the fuel assembly in the space between the shroud tubes and thereby obtain irradiation effects information on several cladding materials without incurring additional irradiation costs. The AEC approved the plan and the fuel assembly was modified to accomplish both test program objectives simultaneously. The final arrangement evolved for the coupon irradiations is shown in Figure 2. This design employed a coupon holder which was shaped to hold the samples in place and to prevent their vibration in the high velocity coolant stream. Figure 3a. and 3b. show the coupons in the holder. The tensile coupon holders were attached to the fuel assembly by means of bands around the assembly. These bands held the coupon holder in place by restraint on a hold-down rod which passed down behind the bands and through the holder. The hold-down rod could be readily removed from the tensile coupon holder, and then the holder and the coupons could be detached from the fuel assembly. By this simple means, samples were taken out of the coupon holder and replaced by others at intermediate exposures.

The accumulated exposures for the samples were determined by both physics analysis for the reactor position and the materials involved, and by chemical activation analysis of sections of the hold-down rods. The hold-down rods were scanned for gamma activity and the relative gamma activities for the positions of the coupons were used to estimate the accumulated neutron exposure for each sample.

This irradiation technique provided high exposure tensile coupons free from chemical, thermal or mechanical forces normally imposed on fuel cladding for direct comparison to the fuel rod cladding irradiated in the same assembly. Although the irradiation temperature was lower than most clad operating temperatures, the results of the mechanical tests could be correlated with other high temperature results. The correlation of the tensile test results with other data can be found in GEAP-5100-9⁽²⁾.

This experiment illustrates the possibility of obtaining additional information from an irradiation experiment without affecting the primary tests by maximum utilization of space available in the item being irradiated.

References:

1. Lyons, M. F. and Weidenbaum, B., ed. "High Performance UO₂ Program Quarterly Reports," GEAP-3771-1 through -18, June 1961 - December 1965.
2. Coplin, D. H., Pashos, T. J., Lyons, M. F. and Weidenbaum, B., "Mechanical Property Changes in Zircaloy-2, Inconel, and Incoloy for Exposures to 2.5×10^{21} nvt (above 1 Mev) GEAP-5100-9, April, 1966.

Figure 1 Fuel Assembly Showing Tensile Coupon Location

1.1.4

1.1.5

Figure 2 Tensile Coupon Holder

a

b

Figure 3 Tensile Coupons in Holder

1.1.6

I. CAPSULE DESIGNS AND OPERATION WITH NON-FISSILE MATERIALS

CONTRIBUTION TO
A E C MEETING
ON
"DEVELOPMENTS IN IRRADIATION CAPSULE
TECHNOLOGY"

Presented by P. MILLIES (FRANCE)

IRRADIATION RIGS

POSSIBILITIES OF STANDARDISATION

- 1) The design of an irradiation rig must cater for specific requirements as regards experimental conditions in the reactor; temperature, flux, atmosphere, measurements, duration, etc.

At the same time, the rig is subject to certain imperatives determined by the reactor itself: shape and size, safety measures, power supply, cooling, etc.

Only rigs which meet requirements of both categories are acceptable. This implies close collaboration between research scientist, project engineer and the team responsible for reactor operation.

- 2) Rig design and fabrication involve a number of widely differing techniques. The project engineer must be familiar not only with nuclear, heat and mechanical engineering design, but also with electronics, constructional problems, use of special materials, etc.

Development of an operational prototype is long and requires lengthy testing. Except for very simple equipment, 18 months can well elapse between the draft project and normal in-pile operation.

As a result, interest may be lost in the experiment, the reactor is not put to good use, the final cost of the rig is high and operation is often not satisfactory.

- 3) It follows that, before specifying a new rig for a new application, a check should be made that an existing rig would not be suitable for the job. This presupposes detailed knowledge of the rigs that have already been developed.

Taken a step further, this reasoning would suggest series manufacture of small batches of rigs as in the rare cases when a number of identical irradiations are carried out. However, experiments are generally all different as regards sample type and form and experimental conditions; moreover, reactors are far from all being the same.

Nevertheless experience shows that it is possible, particularly in the case of technological research, to fix ranges within which experimental conditions are contained; e.g. the temperatures of sheath and structural materials are usually between 150 and 900°C; sample shape and size are defined by existing laboratory equipment; thermal energy of nuclear origin is determined by the reactor, the flux of which is known.

If standard rigs can be designed to cater for experimental conditions within each range, series production can be envisaged. Only 3 to 4 months are then needed to produce the rigs, the cost is reduced, reactors are more rationally used and industry can be interested in this type of construction.

- 4) Two conditions are indispensable for standardisation. These are additional to normal design considerations; they are:

4.1 - Separation of samples from rig

- Continuity of irradiation is ensured, since, in case of accident, samples can be recovered and placed in another rig.
- Rigs which are still in working order at the end of an experiment can be used again.

- It is possible to treat samples separately and time factors are no longer added together. Problems of compatibility, measurement, degasification, etc. are easier to solve.

4.2 - Maximum distributed control power

- It is necessary to achieve precise, constant, homogeneous temperatures. The basic principle is to evacuate, through a localised thermal barrier, a heat flux higher than that produced by nuclear flux in the materials. An auxiliary control heating arrangement compensates for the margin of imprecision due to the dispersion of thermal characteristics, the inexactitude of the forecast of nuclear heating, and the variations of flux in the course of a cycle.
- The relationship between electric and nuclear heating in the rig is given by the formula:

$$W_{\text{electric}} = \left(A + \frac{B}{T} \right) W_{\text{nuclear}}$$

in cases where thermal conduction is preponderant.

In the above formula:

A = sum of constant relative errors affecting factors such as density, coefficient of conductivity, measured flux, etc.

B = sum of constant absolute errors due to machining tolerances, etc.

T = required temperature.

It is found in particular in the medium temperature range that the control power must be all the higher as temperature is low and nuclear heat flux high.

In the case of very high temperatures, in excess of 1000°C, the heating section is placed in an area where the temperature is lower; it is thus possible to obtain a considerably longer life, but coupling is not so good and power requirements are higher.

Consequently the highest power capacity which it is technically possible to achieve is installed.

It should be noted that the energy available for experiments must be all the higher as the research reactor is more powerful.

Each irradiation is considered as a special application.

- 5) It is not possible to use standardised rigs for all irradiation work.

There are cases where experimental conditions fall outside the range limits mentioned above, and special rigs have to be produced. The construction of special rigs should be based on proven techniques, and, where possible, use should be made of sections of standard rigs.

Such cases are, however, not frequent.

- 6) Series production of some types of rigs is today on the up-trend.

18 standardised rigs are in simultaneous use in Siloé. They are used without modification in Melusine, Triton, Avogadro.

They are used in reactors such as BR2 (Belgium), H.F.R. (Holland), F.R.G. (Germany) with a modified connection layout; this has little effect on series production.

They are also in production for the future Osiris reactor.

- 7) These results require:

- placing design and manufacture of rigs for a given field of applications in the hands of a single team.
- perfect acquaintance by this team of research programmes and evolution of these.
- rational choice at the appropriate time of a technical solution and systematic application of this.
- close collaboration with industrial concerns entrusted with rig production, and precise knowledge of their means of production.

TYPE OF RIG	SOME APPLICATIONS
CHOUCA B	<ul style="list-style-type: none"> - Graphite 350 - 450 - 550°C, 4 W/gr - Be 450 - 650°C 4 W/gr - Stainless steel and metallic alloys 580 - 650°C 4 - 5 W/gr - Uranium alloy enriched 4,8% 450 - 560°C in Sodium or NaK pressure 40 kg/cm² - Gaz loop for study of corrosion
CHOUCA D	<ul style="list-style-type: none"> - Be 500°C 15 - 20 W/gr 20 à 30 kg/cm² - Stainless steel 450° - 550°C 15 à 20 W/gr 20 à 30 kg/cm² in Sodium
HEBE	<ul style="list-style-type: none"> - BeO 1300°C 2 - 3 W/gr - Molybdenum and thermocouples 1400°C 2 - 3 W/gr - Graphite 900°C 2 - 3 W/gr - Refractories materials 450°C 2 - 3 W/gr
CYRANO	<ul style="list-style-type: none"> - UO₂ enriched 600°C on the specimen surface (500 W/cm³) pressure 60 kg/cm²
HF 3	<ul style="list-style-type: none"> - UO₂ enriched 20%, 2000°C - UC " 20%, 2000°C
HF 5	<ul style="list-style-type: none"> - UO₂ enriched 20%, 1400°C

"CHOUCA" FURNACE -

Descriptive note -

1 - Introduction

The "Chouca" furnace is designed for the irradiation of structural and fissile material in piles of the swimming pool type. It operates over a temperature range which can be controlled between 150 and 900°C. The rig is in series production.

Leading features are:

- standardisation
- lower (active) part can be uncoupled from upper part
- high useful volume
- uniformity of temperature (6 independent heating coils)
- facilities for measurement of sample temperatures (up to 18 thermocouples)
- operating pressure up to 10 kg/cm²
- ease of introduction and withdrawal of samples without damaging furnace.

2 - Principle of operation

The rig is designed for samples releasing considerable calorific energy of nuclear origin, due in particular to the absorption of γ rays from the core of the pile by the samples to be irradiated and the structural material. It requires a well-defined thermal resistance and a high power electric heating supply.

This auxiliary control heating system is provided to compensate for the margin of imprecision due to the dispersion of the thermal characteristics of the furnace, the difficulty of making an exact forecast of the nuclear heating process, and the variations of flux during the cycle.

The furnace is designed on the basis of the following factors:

- quantity of samples to be irradiated
- quantity and type of structural materials constituting the sample holder
- arrangement of samples
- thickness and type of heat insulation
- length and number of heating elements with a view to obtaining as a result of nuclear heating alone, a temperature in the sample slightly below the required temperature.

The electric resistors, which are placed under the control of a precision governor, come into action to bring the temperature up to the required value and act as temperature controllers.

Each one of the resistors has its own independent power supply and independent governor, to ensure efficient correction of the temperature gradient due particularly to the non-uniformity of the γ flux.

3 - Main characteristics -

Temperature range

The "Chouca" furnace is designed to cover a range of temperatures of from 150 to 900°C with a mean γ flux of 50 W/cm².

Within these limits, the temperature depends on the mass to be irradiated and on the γ flux.

Precision of temperature control

Within the above limits, temperature can be maintained within $\pm 2^\circ\text{C}$ approx. of the setting. In optimum conditions, longitudinal temperature variation can be limited to 1% in spite of the flux gradient.

Pressure

The furnace is filled with gas (Helium, Argon or, for temperatures below 500°C, CO₂) at a pressure of between 2 and 10 kg/cm².

Useful volume

The part of the furnace which can be used for samples to be irradiated is cylindrical in form. The useful volume covered by the flux is 26 mm in diameter and 400 mm high.

It should be noted that two "Chouca" furnaces can be placed side by side in one fuel element.

Note

The useful length of 400 mm can be increased to 586.5 mm if longitudinal uniformity of temperature is not essential.

Heat flow

The furnace is suitable for calorific energy of nuclear origin of up to some 50 W/cm³.

Relatively high degrees of thermal flow can be evacuated at moderate temperatures (a few hundred degrees C): e.g. 20 W/cm² at the wall in contact with the water when operating at 300°C.

4 - Description of "Chouca" rig -

The installation comprises:

- the furnace itself, or lower, active part.
- the connection box. This contains the connectors between the upper and lower parts of the rig, and also serves as a biological protection baffle.
- the down tube. The rig head, with connections to the control bay, is fixed at the top of the down tube.
- the control bay and electrical connection (not shown on figure).

Description of lower part

Note -

The stainless steel used for the "Chouca" furnace is a low-carbon 18/8 steel to American AISI standard 304 L and French AFNOR standard Z3 CN 18/10. This is referred to below by the Ugine steelworks reference NS 22 S.

The lower part of the furnace is in the form of a tube made of NS 22 S, closed by a leak-tight NS 22 S plug welded on the bottom.

At the top end of this tube are located:

- a side opening for the furnace atmosphere control pipe
- a leak-tight passage, also on the side of the tube, for heating resistor power circuits and furnace thermocouple leads.

An NS 22 S fitting on the top of the tube comprises:

- a lifting hook on the vertical centreline of the tube
- a side opening for sample thermocouple leads

The leak-tight passage for sample thermocouple leads is located on the tube sealing plug which is itself prolonged downwards by the sample holder.

The lower part of the tube containing the furnace itself has an outside diameter of approx. 42 mm (exact O.D. depends on thickness of insulation.) Two furnaces can thus be lodged in a single water box having the outside dimensions of the MTR type combustion element.

The furnace is cooled in the pile by water circulating in the annular space between the 42 mm dia. pipe and its housing in the water box.

Description of the furnace itself

The figure represents an exploded view of the furnace and shows:

- the outer furnace tube (O.D. 42 mm approx., NS 22 S), closed by a plug fitted with a cone which served to centre the furnace in the water box.

- the heating tube (I.D. 26.1 mm, length 680.5 mm up to upper connection level, NS 22 S)

The heating tube comprises, from the bottom upwards:

- a plug welded on the tube, with two holes to ensure free circulation of furnace gas.
- six successive stages of Sodern type 1 NC1 15 hairpin shaped heating resistors.

2 Sodern type 2 AB1 15 thermocouples are fitted on each stage of resistors. The resistors and thermocouples are held in mechanical and thermal contact with the support tube by means of a thin magnetically restricted NS 22 S sheath. Heat insulation is provided by a layer of hafnium-free zirconia sprayed on to a nickel-chrome base and machined to the required diameter.

The outer furnace tube is magnetically restricted on the layer of zirconia to seal the assembly.

The junctions between the Sodern heating conductor and the Sodern type 1 CZ AC 20 ~~copper~~ power supply conductors are located at a distance of 50.5 mm from the top resistor, i.e. in an area away from the influence of electric and nuclear heat. These junctions are bedded between two layers of sprayed alumina. A layer of nickel-chrome sprayed between the junctions and the top resistor provides perfect thermal contact and ensures cooling in this area. The height of the furnace in the rig is determined by a shoulder in the nickel-chrome layer, below which downward expansion is free.

Biological protection baffle

The main purpose of the baffle is to prevent radiation in a straight line from the core to the top of the pile. The dimensions of the baffle are standardised, but can be adapted in special cases if necessary. The baffle is formed by the flat rectangular connection box containing the plug-in connectors between the conductors from the furnace (heating resistors and thermocouples) and the connecting leads to the rig head.

The Sodern type 2 AB 1 10 sample thermocouple connections are lead out through the top of the furnace tube. A maximum of 18 thermocouples can be connected to the rig head by means of a PVC sheathed conductor system.

Down tube -

The down tube forms the rigid connection between the furnace and the platform above the surface of water in the pile. (In some special cases the down tube can be replaced by a flexible tube, provided that this is leak-tight and will withstand the furnace pressure.

In this case a separate arrangement must be provided to maintain the furnace in the irradiation position.)

On the standard rig, the down tube consists of two pipes:

- rigidity is provided by a leak-tight NS 22 S pipe, I.D. 36 mm, O.D. 40 mm, housing the resistor power circuits and the control thermocouple leads
- an NS 22 S pipe, I.D. 12 mm, O.D. 17 mm, is provided for furnace atmosphere control.

The two pipes are connected by NS 22 S gussets welded between them to ensure perfect rigidity. The PVC sheath, I.D. 18 mm, O.D. 23 mm, containing the sample thermocouple leads, is fixed to the down tube.

Rig head -

The rig head occupies a rectangular space 110 x 220 mm overall. From the bottom upwards it comprises:

- 1 pressure switch. On the standard rig this:
 - . trips a remote alarm if furnace pressure falls below 1.5 kg/cm^2
 - . cuts the heater supply if pressure falls below 0.65 kg/cm^2
- 1 pressure gauge showing pressure in the connection box
- 1 3-pin plug (Socapex type EM 134) for pressure switch connections

- 1 39-pin plug (Socapex type EM 439 R) for connection of sample thermocouple and pressure switch leads to control bay
- 1 39-pin plug (Socapex type EF 439 B) for connection of furnace thermocouple leads to control bay
- 1 12-pin plug (Socapex type EM 413 AB) for heater resistor power circuits
- 1 valve for pressurisation of connection box
- 1 valve for pressurisation of furnace

5 - Furnace and sample handling

Provided sufficient space is available close to the core of the pile, the "Chouca" furnace can be lowered into place and raised while the pile is operating. This pre-supposes the following precautions before starting the pile:

- installation, in the irradiation position, of a water box fitted with an orifice calibrated for normal water flow
- a good estimate of the antireactivity of the furnace

The furnace is suspended from the rig head for handling. A hook is provided at the top of the lower part for a guide wire.

Introduction and withdrawal of samples

Non-active samples can be introduced very easily, while the lower part of the furnace is held so that the upper plug is clear of the water.

The "Chouca" rig is designed so that samples can be withdrawn either in the hot cell or by means of a withdrawal shield. In the latter case, the furnace is held at the surface, as for sample introduction.

6 - Features which can be modified to cater for specific requirements -

Thermal features

- Type and thickness of insulation (Zirconia, Alumina, Helium, Neon, argon, etc.)
- If necessary, number and length of heating elements; i.e. heater power consumption.

Mechanical features (adaptation for different piles)

- Length of active part
- Shape and dimensions of upper part

7 - Basic Design Information

The following information is required to design a "Chouca" furnace:

Furnace design

- gamma, fast neutrons, thermal flux required
- degree of irradiation or integrated flux (duration)
- type and size of samples
- design of sample holder:
 - . compatibility of samples with structural materials that can be used for sample holder
 - . shape and number of samples
 - . method of withdrawing samples
- type and pressure of gas. Gas circulation if any.
- temperature required and precision of temperature measurement

Mechanical features

The following three main dimensions must be precisely defined, taking into account the size of the pile and the means available for handling the furnace and withdrawing samples (overhead crane, hot cell, etc.):

- length of down tube from connection box to rig head (L)
- length of active part (M)
- length of baffle (N)
- operating pressure

Electrical factors

- supply voltage for control bay
- connections available with control room for safety and alarm circuits

8 - SOME TYPICAL APPLICATIONS

Type of sample	Graphite	Ni	UO ₂ dispersed in graphite	U Alloy
Sp. gravity	1.65 - 1.85	8,9	1.8 overall	
Weight	150 gr	300 gr	12 gr UO ₂	200 gr
Sample holder	Aluminium turret	stain- less steel	Graphite	NaK Capsule
Flow	$\dot{V} = 3.6 \text{ W/gr}$ $\phi_r = 5.6 \cdot 10^{13}$	$\dot{V} = 2 \text{ W/G}$ $\phi_r = 3.3 \cdot 10^{13}$	$\phi_r = 4,5 \cdot 10^{13}$ $\phi_{th} = 17 \cdot 10^{13}$	$\phi_{th} = 3,7 \cdot 10^{13}$
Thickness of Zirconia	2 mm	1,2 mm	2 mm	
Temperature	450°C	625°C	850°C	400-600°C
Number of hours Operation in pile	>4000	>3000h	1000 h - Irrad. in progress	Irradiation in progress

CONTROL BAY FOR "CHOUCA" B FURNACE

(See description of control bay for "Chouca" D furnace)

- 1 - Furnace span shut-off valve.
- 2 - Upper section shut-off valve
- 3 - Power circuit connector
- 4 - Furnace thermocouples outlets
- 5 - Samples thermocouples outlets
- 6 - Pressure governing contactor connector
- 7 - Upper section pressure gauge
- 8 - Lower section pressure gauge
- 9 - Down-tube
- 10 - Flexible thermocouple tube
- 11 - Furnace tube
- 12 - Sample-thermocouples connections
- 13 - Sample removal fixture
- 14 - Furnace
- 15 - Connection box
- 16 - Alumina metallizing insulation
- 17 - Centering stud
- 18 - Nickel Chrome metallizing
- 19 - Magnetically restricted tube
- 20 - Dehafniumized zirconia metallizing insulation
- 21 - Outer furnace tube
- 22 - Sample holder
- 23 - Heater element thermocouples
- 24 - Heater thermocoax
- 25 - Double water-box
- 26 - Electrical connections

"CHOUCA D" FURNACE

Descriptive note

I - INTRODUCTION

The "Chouca D" furnace is designed for the irradiation of structural material of fuels at temperatures of 150 to 900°C in high-flux reactors such as BR2. It is very similar in design to the "Chouca B" furnace, but, since it is intended for use in piles where gamma heating is of the order of 15 to 25 W/gr, it is equipped for high heat evacuation and consequently requires very high power auxiliary electric heaters (approx. 75 W/cm²)

The rig is in series production

Leading features are:

- standardisation
- lower (active) part can be uncoupled from upper part
- high useful volume
- uniformity of temperature (6 independent heating coils)
- facilities for measurement of sample temperature (up to 18 thermocouples)
- operating pressure up to 60 kg/cm²
- ease of introduction and withdrawal of samples without damaging furnace

II - PRINCIPLE OF OPERATION

The auxiliary control heating system is provided to compensate for the margin of imprecision due to the dispersion of the thermal characteristics of the furnace, the difficulty of making an exact forecast of the nuclear heating process and the variations of flux during the cycle.

The furnace is designed on the basis of the following factors:

- quantity of samples to be irradiated
- quantity and type of structural materials of which the sample holder is made
- arrangement of samples
- thickness and type of heat insulation
- length and number of heating element; i.e. power requirements for electric heating

Note

The length of 380 mm can be increased to 600 mm if longitudinal uniformity of temperature is not essential.

The electric resistors, which are placed under the control of a precision governor, come into action to bring the temperature up to the exact value required and act as temperature controllers.

Each one of the resistors has its own independent power supply and independent governor, to ensure efficient correction of the temperature gradient (non-uniformity of γ flux)

III - MAIN CHARACTERISTICS

Temperature range

The "Chouca" furnace is designed to cover a range of temperature of from 150 to 900°C in the presence of thermal flux due to the release of calorific energy of nuclear origin which, in certain conditions, may reach 150 W/cm².

Within these limits, the temperature depends on the thickness of heat insulation with respect to the samples to be irradiated, and on the release of calorific energy.

Precision of temperature control

Within the above limits, the temperature can be governed to within approx. $\pm 3^\circ\text{C}$. In optimum conditions the longitudinal temperature variation can be limited to a few °C in spite of the flux gradient.

Pressure /

The furnace is filled with gas (helium, argon or, for temperatures below 500°C, CO₂) at a pressure of between 2 and 60 kg/cm².

Useful volume

The part of the furnace which can be used for samples to be irradiated is cylindrical in form. The useful volume covered by flux is 20 mm in diameter and 380 mm high.

Thermal flux

The furnace is suitable for calorific energy of nuclear origin of up to some 300 W/cm³. Relatively high degrees of thermal flux can be evacuated at moderate temperatures (a few hundred C); e.g. 50W/cm² at the wall in contact with the water when operating at 300°C.

IV - DESCRIPTION OF "CHOUCA D" RIG

The installation comprises:

- the furnace itself, or lower, active part
- the connection box. This is cylindrical in shape and comprises a flange for fixing to the reactor tank.
- the rig head and connections between the control bay and the submerged part in the pile.
- the control bay and electrical connections

Description of lower part

Note

The furnace is built of stainless steels as follows:

- an ordinary 18/10 low carbon steel to American AISI standard 304 L and French AFNOR standard Z3 CN 18/10
- a 25/20 steel to American AISI standard 314 - 310 and French AFNOR standard Z 15 CN S 25/20. This steel is creep-resistant to 900°C.

The lower part of the furnace is in the form of a tube made of Z3 CN 18/10 steel, closed by a leak-tight plug of the same grade of steel, welded on the bottom. A leak-tight passage is provided at the top of the tube for the heating resistor power circuits and the furnace thermocouple leads.

The sample thermocouple leads are lead through a sealed passage located on the tube sealing plug which is itself prolonged downwards by the sample holder. The lower part of the tube containing the furnace itself has an outside diameter of approx. 26 mm.

The furnace is cooled in the pile by water circulating in the annular space between the 26 mm dia. pipe and an outer housing tube (I.D. 30 mm, O.D. 34 mm)

Description of the furnace itself

The figure represents an exploded view of the furnace and shows:

- the furnace tube (O.D. 26 mm approx. Z3 CN 18/10), closed by a plug fitted with a cone which serves to centre the furnace in the water box.
- the heating tube (I.D. 20.1 mm, length 380 mm up to connection level, Z 15 CNS 25/20)

The heating tube comprises, from the bottom upwards:

- a plug welded on the tube, with two holes to ensure free circulation of furnace gas.
- six successive stages of hairpin shaped heating resistors.

The resistors are stainless steel sheathed and MnO insulated. They are 1.5 mm in diameter, and their resistance is of the order of 5.1 /m.

Two 1.5 mm dia. stainless steel sheathed chromel/alumel thermocouples are fitted on each stage. The resistors and thermocouples are held in mechanical and thermal contact

with the support tube by a layer of sprayed molybdenum which is then machined to a diameter determined by the thickness of insulation required. Heat insulation is provided by a layer of dehafniumised zirconia sprayed on the assembly and machined to the required diameter.

The outer furnace tube is magnetically restricted on the layer of zirconia to seal the assembly.

The junctions between the heating resistors and the 2 mm dia. stainless steel sheathed copper power supply conductors are located at a distance of 50 mm from the top resistor, i.e. in an area away from the influence of electric heat. These junctions are bedded between two layers of sprayed alumina. A layer of molybdenum sprayed between the junctions and the top resistor provides perfect thermal contact and ensures cooling in this area.

V - FEATURES WHICH CAN BE MODIFIED TO CATER FOR SPECIFIC REQUIREMENTS

Thermal features

- type and thickness of insulation (alumina, zirconia, helium, neon, argon, etc.)
- if necessary, number and length of heating elements; i.e. heater power consumption.

Mechanical features (adaptation for different piles)

- length of active part
- shape and dimensions of upper part

VI - BASIC DESIGN INFORMATION

The following information is required to design a "Chouca" furnace:

Furnace design

- gamma, fast neutron and thermal flow rates required
- degree of irradiation or integrated flow (duration)
- type and size of samples

- design of sample holder:
 - . compatibility of samples with structural materials that can be used for sample holder
 - . shape and quantity of samples
 - . specific gravity
 - . method of withdrawing samples
- type and pressure of gas. Gas circulation if any
- temperature required and precision of temperature measurement

VII - TYPICAL APPLICATIONS FOR BR2

NATURE OF SAMPLES	STAINLESS STEEL	Zr-Cu
Density	7.9	6.6
Mass	220	185
Sample-holder	Sodium capsule	Sodium capsule
Flux	$\phi_r 5 \times 10^{14}$	$\phi 5 \times 10^{14}$
Thickness of zirconia	0.45 mm	0.5 mm
Temperature	550 and 650°C	650°C
Number of in-pile operating hours	1,000 hours irradiation in progress	irradiation in progress

DESCRIPTIVE NOTES ON THE CONTROL BAY FOR THE "CHOUCA"
FURNACE

This bay, constructed for the power-supply to, and control of a "CHOUCA" type in-pile irradiation furnace, performs the following functions:

- Controlled power supply to the six heating elements which make up the furnace;
- Measurement of the voltages and amperages on the elements;
- Switching off the heating and emission of a warning in the event of failure of the device or non-fulfilment of the conditions necessary for proper functioning, viz. perturbation of the supply network or insufficient throughput of cooling water.
- Recording of temperatures and indication of operating pressures.

For these purposes it comprises:

- A "power-panel which supplies the furnace;
- An "instrument panel" containing a voltemeter and an ammeter;
- A "distribution board" enabling the various elements to be interconnected as required;
- A "safety drawer" which acts on the "power panel" in the event of a failure or malfunction and emits the appropriate warning signals;
- A "regulating drawer" which likewise acts on the "power panel" and regulates the supply to the furnace according to the desired temperature;
- A multi-curve recorder.

I - DESCRIPTION (See drawings SIL-GE 350)

1 - Power panel (see drawing D 1)

- This panel, located in the bottom third of the cabinet, comprises:

1.1 - Protection and adaptation equipment as follows:

1.1.1 - Fuses protecting the power and safety supply circuits.

- . On the power circuits, 3 porcelain cartridge fuses, one on each phase (32 Amps for 380 V or 50 Amps for 220 V).
- . On the safety circuit, 2 glass cartridge fuses (European series, 5 x 20 mm) of the delayed action type (3 Amps for 220 V or 5 Amps for 110 V).

1.1.2 - A manually operated main switch, which serves to energize the cabinet. This is located on the fuse box access panel, at the foot of the front of the cabinet.

1.1.3 - A row of terminals which enable the cabinet to be quickly adapted to a supply voltage of either:

- 380 V, 3-phase + neutral, or
- 220 V, 3-phase

1.1.4 - A variable voltage autotransformer giving a 127-V supply in the cabinet from any 50 cps, a.c. supply of from 100 to 260 V.

1.1.5 - A contactor which acts solely on the furnace power supply and is actuated by the "Start-Stop" control of the furnace and by the safety drawer.

1.2 - Six identical "supply units", one for each heating element, each unit consisting of:

1.2.1 - An individual protection fuse (20 amp fuse in a porcelain cartridge);

1.2.2 - A 3.5 KVA insulator-type transformer (see drawing D 6.7) with the following characteristics:

- . Primary winding 220 volts, maximum current 22 amp;
- . Secondary winding with multiple tapings between 0 and 240 volts in 10-volt steps. Maximum current 20 amps at any tapping.

1.2.3 - Two controlled silicon diodes, type 2N 689, mounted head to tail, for varying the power applied at the heating element.

1.2.4 - An ultra-fast fuse (type 20 A4 UR) on the secondary winding of the transformer to protect the controlled diodes.

1.2.5 - A thermocouple ammeter shunt for intensity measurements.

1.2.6 - Connections for measurement of the voltage at the heating-element terminals and at the transformer terminal.

1.3 - A ventilation unit which forms the lower half of the rear panel of the cabinet.

2 - The instrumentation and voltage balancing panel
(see drawing D.2) This panel, which forms the bottom third of the front of the cabinet comprises:

2.1 - A current-intensity measuring unit which is mounted on the left-hand side of the panel and consists of:

2.1.1 - A twelve-way switch (an "off" position between each of the six positions used) for selecting the element on which the measurement is performed.

2.1.2 - A "high-frequency" type ammeter with thermocouple shunt (see 1.2.5 above). This is rendered necessary by the non-sinusoidal form of the current at the output of the controlled diodes.

2.2 - A voltage measuring unit mounted on the right-hand side of the panel and consisting of:

2.2.1 - A selector switch identical with that for the intensity measurement;

2.2.2 - A class 1.5 voltmeter graduated from 0 to 250 volts.

2.2.3 - A press-button change-over switch mounted in the centre of the panel between the two selector switches. This is used for switching over from measurement of the voltage at the heating-element terminals to measurement of the voltage at the terminals of the corresponding transformer.

2.3 - A voltage-balancing unit comprising six pairs of 100 k Ω resistors, one for each heating element. In each pair the resistors are mounted in series and connected to the terminals of the corresponding heating element. The mid-point thus obtained is earthed to the furnace. By means of this arrangement it is possible to maintain a correct balance of the voltages at the extremities of each element with respect to earth, even when there is a big difference between the insulation of these two extremities. This reduces the danger of arcing between the resistance wire and earth as a result of overvoltage.

3 - The " distribution board" (see drawings D 32)

This board consists of a panel made of insulating material on which are mounted jacks which can be interconnected as on a telephone switchboard. It is located at the rear of the bay and is accessible via the top rear panel. It is mounted on hinges so that it can be opened to give access to the back. On this panel are:

3.1 - The recorder input lines. The recorder has 16 input channels and the corresponding terminals are arranged in a column at the centre of the panel, numbered from 1 to 16 from top to bottom. The cords attached to jacks enable one or more channels to be connected in parallel.

3.2 - The measurement and safety lines from the furnace

3.2.1 - The temperature measurement lines comprise 30 pairs, viz.,

- 12 for the furnace thermocouples (2 for each heating element), and
- 18 for the sample thermocouples.

3.2.2 - The pressure measurement lines comprise the supply and signal lines of the potentiometer pick-ups.

3.2.3 - The leakage-detection lines, which are connected to the three electrodes mounted in the bottom of the furnace.

3.3 - The safety and warning lines from the reactor control room.

- The bay must receive a signal which indicates the absence or insufficiency of the cooling-water throughput and causes the heating to be switched off. This signal can consist in the opening or closing of a contact or, if desired, in the appearance or disappearance of a DC or AC voltage of between 21 and 300 volts.
- Furthermore, it is possible to send to the control room:
 - . a pre-alarm signal (one pair)
 - . an alarm signal (over 2 pairs)These signals can correspond to the opening or closing of a contact, as desired.

3.4 - The input lines of the regulating drawer (see section 5 below), comprising:

3.4.1 - Six pairs for the "fast chains". The corresponding terminals are mounted near those of the thermocouples which have to actuate them.

3.4.2 - Six pairs for the "slow chain". For the same reason the relevant terminals are mounted at the top of the board, immediately to the left of the sample thermocouples.

By means of cords attached to the jacks the regulators can be connected in parallel with the recorder input channels.

3.5 - The output lines of the regulators and the supply control lines for the heating elements.

3.6 - The fuses protecting:

- the three phases of the "safety drawer" power circuit
- the supply to the recorder
- the supply to the thermostat enclosure
- the supply on the safety circuit of the safety drawer" and of the "regulating drawer".

4 - The "safety drawer" (see drawing D 41)

This drawer, mounted immediately above the power panel, contains the safety logical chain and the elements which actuate the controlled diodes.

Its functions are as follows:

- Giving warning of damage and transmitting the appropriate pre-alert or alert signal.
- Working out the actuation impulses for the controlled diodes of the power pack.
- Manual control of the power supplied to each heating element.
- Shutting off the heating in the event of the situations provided for in the following table (see drawing D 41.b)

NOTE: Wiring and sub-assembly layout are designed with a view to adaptability to different operating conditions. The table of safety arrangements can therefore be modified as required.

Inputs	SIGNALS			ACTION					Remarks
	Warning lamp	Orange flasher	Red flasher	Automatic signal hold-in	Normal shutdown of heating	Emergency shut-down of heating	Automatic action hold-in	Pre-alerting of control room	
PRESSURE									
3 pick-up									
Minimum pre-alarm	X			X					Threshold approx. 15 bars.
Minimum alarm	X	X		X	X			X	Threshold approx. 13 bars
Maximum alarm	X	X		X	X			X	Operation at 2/3 Threshold approx. 20 bars, 2/3 operation
TEMPERATURE									
Maximum									
- recorder									
- slow chain									
Pre-alarm	X			X	X				
Alarm	X	X		X	X	X	X	X	
COOLING WATER	X				X	X			
- contact-closed									
MAINS	X				X	X			On the 3 phases. 2 sec. time lag.

The front of the drawer contains:

- the fault warning lights;
- a press-button for resetting the safety devices;
- a two-button "stop"- "star" switch for the furnace power supply
- the six potentiometers for regulating the power supplied to the heating elements;
- a galvanometer graduated from 0 to 20 kg/cm² to indicate the pressure inside the furnace.

On the right-hand side of the drawer and accessible via the side panels are:

- the contact galvanometers for regulating the thresholds of the minimum and maximum pressure pre-alarm and alarm,
- the shunt switches and press-buttons for testing the temperature pre-alerts and alerts and the mains safety circuit.
- seven switches, each with three positions: "test", "operate", "shunt", acting on the safety devices:
 - . pressure, pick-ups 1, 2 and 3
 - . cooling water
 - . sodium leakage, electrodes 1, 2 and 3

The logical chain set up in order to obtain the operation desired is shown on drawing D 41. The general principle is as follows:

- on each input a shaping circuit brings the signal up to the necessary level for the logic employed (SONEA negative logic).

Signal 0 between 0 and - 1.32 volts

Signal 1 between - 4.83 and - 12 volts

These signals are then processed by sub-assemblies: storage units, 2/3 grouping, or counters which actuate the output summation functions.

The various functions circuits and sub-assemblies are mounted on plug-in plates.

The shaping circuits which make use of conventional components are printed. The functions sub-assemblies are built up from standard SONEA plug-in units.

Eight different models of plate are used.

Fool-proof slides and pins reduce the risk of error when the plates are plugged in.

The outputs of each sub-assembly are connected to marked jacks which serve to facilitate any necessary tests.

5 - The "regulating drawer"

This drawer is located immediately above the "safety drawer".

5.1 - Principle (see drawing D 51):

The control system adopted consists of a double loop:

- the first loop (proportional action) is piloted by a thermocouple which is mounted directly on the heating element controlled.
- the second loop (integrating action), which is piloted by a thermocouple attached to the corresponding samples, corrects the drifts of the first loop.

This system is reproduced six times, once for each heating element.

- the six proportional-action loops, known as "fast chains", are independent.
- the six integrating-action loops are combined in a single "slow chain" which is actuated successively by the six selected sample thermocouples and which acts in turn on the six corresponding "fast chains".

5.2 - The fast chains operate on the all-or-nothing system linearized by modulation.

Principle (see drawing D 521)

The voltage supplied by the pilot thermocouple is compared to a reference voltage (temperature setting).

The difference, amplified and superimposed on a saw-tooth voltage, is applied at the input of a threshold flip-flop circuit. In this way voltage crests obtained for which the ratio of the duration to the recurrence period is a function of the difference between the heating-element temperature and the temperature setting.

The voltage crests are used to override the trip relays of the controlled diodes which feed the heating elements.

Since the saw-tooth recurrence period (0.1 - 0.12 sec.) is short in comparison with the time constant of the heating elements (approx. 5 sec.), the latter "integrate" the power crests supplied, and their temperature is a function of the average value of these crests.

The power correction resulting from a deviation of the operating temperature from the temperature setting is consequently a function of the latter.

5.3 - The "slow chain"

The "slow chain" was designed with a view to making the maximum use of the elements shared with the "fast chains" and the "regulating drawer".

Principle (see drawing D 531)

A programmer piloted by the same clock as the saw-tooth modulator controls, at a rate six times slower, the synchronized switching operations of the thermocouples, the reference voltages and the outputs of a chain identical with the "fast chains".

The output crests are used to override two trip relays which actuate controlled diodes feeding the two windings of a split-field motor. One of the trip relays is overridden directly, the other after inversion of the crests.

The motor therefore rotates in one direction during a crest and in the opposite direction during the interval between that crest and the next. The resulting rotation and, consequently, the sign and the value of the average speed of the motor depend on the average value of the crests, which, as we have seen, is a function of the difference between the operating temperature and the temperature setting.

This motor is used to drive the potentiometer which provides the reference voltage of the corresponding fast chain.

The motor-potentiometer unit, the speed of which is a function of the temperature difference, performs the time integration of this difference.

5.4 - Presentation

The front of the drawer comprises six identical panels mounted side by side, corresponding to the six channels of the "slow chain".

Each panel comprises:

- a potentiometer for regulating the reference voltage (sample temperature setting). Potentiometer locks in 10 turns
- a switch with two positions, "manual" and "automatic", for overriding the "slow chain" on the channel in question.
- two push-buttons, "up" and "down", which, when the switch is in the "manual" position, directly actuate the motor-potentiometer unit which regulates the reference voltage of the corresponding "fast chain".
- two pilot lamps indicating the direction of rotation of the motor.

A common switch for the whole assembly enables the programmer to be stopped on one of the channels.

6 - The recorder

The recorder normally used on this bay is a 16-channel "MECI" Speedomax S.134 (built under Leeds and Northrup licence. It is graduated from 0 to 10 with sensitivity switching from 0 to 1, 2, 5, -5, -10 or 25 millivolts and origin change (shift from 0 to -57 mV).

CHOUCA D

- 1 - Support tube
- 2 - Heating resistors
- 3 - Furnace thermocouples
- 4 - Thermal bond: sprayed molybdenum
- 5 - Insulation: sprayed zirconia
- 6 - Electrical connections, supply cables - heating resistances
- 7 - Sprayed insulation (Schoop process)
- 8 - External casing (outer tube)
- 9 - Pressure-tube anchoring
- 10 - Sample-holder
- 11 - Sample thermocouples
- 12 - NaK safety devices
- 13 - Sodium recovery pan
- 14 - Samples
- 15 - Samples
- 16 - Samples
- 17 - Thermocouples
- 18 - Thermocouples
- 19 - Thermocouples
- 20 - Thermocouples
- 21 - Na or NaK level

1.2.36

"HEBE" FURNACE

DESCRIPTIVE NOTE -

1 - INTRODUCTION

The "Hébé" furnace is designed for the irradiation of structural material in piles of the swimming pool type. It operates over a temperature range which can be controlled between 350 and 1400°C. The rig is in series production.

LEADING FEATURES ARE:

- Standardisation
- Lower (active) part can be uncoupled from upper part
- High useful volume
- Uniformity of temperature (4 independent heating coils)
- Facilities for measurement of sample temperatures (up to 18 thermocouples)
- Operating pressures up to 60 Kg/cm²
- Ease of introduction and withdrawal of samples without damaging furnace.

2 - PRINCIPLE OF OPERATION

Basically, operation of the furnace depends on nuclear heating due to the absorption of γ rays from the core of the pile by the samples to be irradiated and the structural material. ✓

The furnace is designed on the basis of the following factors:

- Quantity of samples to be irradiated
- Quantity and type of structural materials constituting the sample holder
- Arrangement of samples
- Gas with which furnace is filled (Helium, Argon, N₂, CO₂) taking into account chemical compatibility at operating temperature, with a view to obtaining, as a result of nuclear heating alone, a temperature in the sample slightly below the required temperature.

The electric resistors, which are placed under the control of a precision governor, come into action to bring the temperature up to the required value and act as temperature controllers.

Each one of the resistors has its own independent power supply and independent governor, to ensure efficient correction of the temperature gradient due particularly to the non-uniformity of the γ flux.

3 - MAIN CHARACTERISTICS

Temperature range -

The "Hébé" furnace is designed to cover a range of temperatures of from 350 to 1400°C with a mean γ flux of up to 1 W/gr.

Within these limits, the temperature depends on the adjustment of the mass to be irradiated and on the γ flux.

Precision of temperature control -

Within the above limits, the temperature can be maintained within $\pm 1\%$ of the setting. In optimum conditions, longitudinal temperature variation can be limited to 1% in spite of the flux gradient.

Pressure -

The furnace is filled with gas (Helium, Argon or, for temperatures below 500°C, CO₂). When this serves only for thermal isolation, a pressure of 2 kg/cm² is sufficient. However, with a view to catering for the majority of cases that may be found in practice, the "Hébé" furnace is designed to withstand an operating pressure of 60 kg/cm².

Useful volume -

The "Hébé" furnace is designed to enable maximum use to be made of the space available in an MTR type fuel element. The useful volume is 53 mm in diameter and 400 mm long.

Note -

- (1) The useful length of 400 mm can be increased to 500 mm if longitudinal uniformity of temperature is not essential.
- (2) In certain cases of irradiation at high temperatures, a heat screen can be fitted to the furnace. This reduces the useful diameter to 49 mm.

Heat flow -

Graph n° 1 shows the temperature reached on the outside of the heating duct as a function of the calorific power evacuated, with and without heat screen and for both helium and argon filled furnace. It will be seen that the "Hébé" furnace can evacuate 6000 W for the total volume, i.e. a heat flow of 10 W/cm^2 approx.

4 - DESCRIPTION OF "HEBE" RIG

The figure represents an outline view of the rig, showing:

- the furnace itself, or lower, active part
- the biological protection baffle. This also serves as a housing for the connections between the upper and lower parts of the rig.
- the down tube. The rig head, with connections to the control bay, is fixed at the top of the down tube.
- the control bay and electrical connections (not shown on figure).

Description of lower part -

Note -

The stainless steel used for the "Hébé" furnace is a low-carbon 18/8 steel to American AISI standard 304 L and French AFNOR standard Z3 CN 18/10. This is referred to below by the Ugine steelworks reference NS 22 S.

The lower part of the furnace is in the form of a tube made of NS 22 S. The size of this tube depends on the handling facilities available at the pile where it is to be used.

The tube is closed by a leak-tight NS 22 S plug welded on the bottom. A milled NS 22 S fitting on the top of the tube comprises:

- on the vertical centreline of the tube: a connection flange and an opening through which the sample holder is introduced.
- on the side:
 - . a leak-tight passage for heating resistor power circuits and furnace thermocouple leads
 - . an opening for the furnace atmosphere control pipe.

The lower part of the tube (I.D. 66 mm, O.D. 70 mm) containing the furnace itself can be fitted into a water box having the outside dimensions of the MTR type fuel element.

The furnace is cooled in the pile by water circulating in the annular space between the 66/70 mm dia. pipe and the water box.

Description of the furnace itself -

The figure represents an exploded view of the furnace and shows:

- the outer furnace tube (66/70 mm dia., NS 22 S) closed by a conical plug which serves to centre the furnace in the water box.
- the heating tube (I.D. 53 mm, O.D. 54 mm, length 707 mm up to upper connection level, NS 22 S)

The latter comprises, from the bottom upwards:

- a machined fitting which serves as a lateral guide to locate the heating tube on the lower plug. (The heating tube is free to move axially to allow for expansion.)
- holes to ensure free circulation of furnace gas.
- four successive stages of Sodern type 1 NC1 20 hairpin shaped heating resistors. These are held in mechanical and thermal contact with the tube by a thin layer of sprayed nickel-chrome approx. 5/10 mm thick.
- four thermocouples
- the junctions between the Sodern heating conductor and the Sodern type 1 CZ AC 20 copper power supply conductors; these are located at a distance of approx. 200 mm from the top resistor, i.e. in an area away from the influence of electric and nuclear heat. These junctions are bedded between two layers of sprayed alumina.

The heating duct is "suspended" inside the outer furnace tube at this level from a collar fitted to the latter. This direct metal contact ensures evacuation of the heat produced at the level of the junctions and provides good operating conditions for these.

Note -

On one model of the "Hébé" furnace, thermal exchange is limited by a cylindrical screen (I.D. 62 mm, O.D. 62.8 mm, NS 22 S) fitted between the heating duct and the outer furnace tube. In this case the dimensions of the heating tube are reduced to I.D. 49 mm, O.D. 50 mm, consequently reducing the volume available for irradiation.

Biological protection baffle -

The main purpose of the baffle is to prevent radiation in a straight line from the core to the top of the pile. The dimensions of the baffle are standardised, but can be adapted in special cases if necessary.

The baffle comprises the cylindrical connection box containing the plug-in connectors between the conductors from the furnace (heating resistors and furnace thermocouples) and the connecting leads to the rig head.

The Sodern type 2 AB-1 15 sample thermocouple connections are lead out through the top of the furnace tube. A maximum of 18 thermocouples can be connected to the rig head by means of a PVC sheathed 36-conductor cable.

Down tube -

The down tube forms the rigid connection between the furnace and the platform above the surface of water in the pile. (In some special cases the down tube can be replaced by a flexible tube, provided that this is leak-tight and will withstand the furnace pressure. In this case a separate arrangement must be provided to maintain the furnace in the irradiation position.)

On the standard rig, the down tube consists of two pipes:

- rigidity is provided by an NS 22 S pipe, I.D. 45 mm, O.D. 50 mm which serves for furnace atmosphere control.
- an NS 22 S pipe, I.D. 25, O.D. 28 mm, houses the resistor power circuits and the control thermocouples leads.

The two pipes are connected by a number of 40 x 100 mm NS 22 S gussets welded between them to ensure perfect rigidity. The PVC sheath, I.D. 18 mm, O.D. 23 mm, containing the 36-conductor cable for the sample thermocouples, is fixed to the down tube.

Rig head -

The rig head occupies a cylindrical space dia. 120 mm, height 810 mm overall. From the bottom upwards it comprises:

- 1 39-pin plug (Socapex type EF 439 R) for the PVC sheathed cable which goes up the down tube.
- 1 39-pin plug (Socapex type EM 439 R) for connection of sample thermocouple leads to control bay.
- 1 19-pin plug (Socapex type EM 319 L) for connection of furnace thermocouple and pressure switch leads to control bay.
- 4 2-pole plugs (Socapex type EM 22 B) for heater resistor power circuits.
- 1 pressure gauge
- 2 pressure switches. On the standard rig:
 - . one of these trips a remote alarm if furnace pressure falls below 1.5 kg/cm^2 .
 - . the other cuts the heater supply if pressure falls below 0.65 kg/cm^2 .
- 1 valve which is leak-tight at 60 kg/cm^2

5 - FURNACE AND SAMPLE HANDLING

Provided sufficient space is available close to the core of the pile, the "Hébé" furnace can be lowered into place and raised while the pile is operating. This presupposes the following precautions before starting the pile:

- installation, in the irradiation position, of a water box fitted with an orifice calibrated for normal water flow.
- a good estimate of the antireactivity of the furnace.

The furnace is suspended from the rig head for handling. A hook is provided at the top of the lower part for a guide wire.

Introduction and withdrawal of samples -

Non-active samples can be introduced very easily, while the lower part of the furnace is held so that the upper plug is clear of the water.

The "Hébé" rig is designed so that samples can be withdrawn either in the hot cell or by means of a withdrawal shield. In the latter case, the furnace is held at the surface, as for sample introduction.

6 - FEATURES WHICH CAN BE MODIFIED TO CATER FOR SPECIFIC REQUIREMENTS

Thermal features -

- Gas with which furnace is filled (helium, argon or CO₂) - problem of chemical compatibility at operating temperature.
- Whether screen is required.
- Shape of sample holder. This can also be fitted with a screen if required to increase temperature gradient between samples and heating tube wall surface.
- If necessary, number of heating elements.

Mechanical features (adaptation for different piles)

- length of active part
- shape and dimensions of upper part

7 - BASIC DESIGN INFORMATION

The following information is required to design a "Hébé" furnace:

Thermal features -

- Gamma, fast neutron and thermal flux required
- Degree of irradiation or integrated flux fixing duration of irradiation

- Type of samples:

- . compatibility with furnace gas and with materials used for construction of furnace and sample holder
- . shape and quantity
- . specific gravity

- type and pressure of gas

- temperature required and precision of temperature measurement

Mechanical features

The following three main dimensions must be precisely defined, taking into account the size of the pile and the means available for handling the furnace and withdrawing samples (overhead crane, hot cell, etc.) (see outline drawing).

- length of down tube from connection box to rig head (L)
- length of active part (M)
- length of baffle (N)
- operating pressure

Electrical features

- supply voltage for control bay
- connections available with control room for safety and alarm circuits.

8 - SOME TYPICAL APPLICATIONS

Type of sample	Glucine BeO	Refractory materials	Graphite	Organic liquids	Stainless metals
Sp. gravity	2.5	0.1 to 0.5	1.7	< 1	8
Weight	160 gr	total 241.6 gr	250 gr	60 gr	120 gr max. at 350°C
Sample holder	glucine turret	silicon capsule	quartz capsule	quartz capsule	alum. or st. steel acc. to temp.
Flow	n. th. $1.03 \cdot 10^{14}$ n. r. $1.58 \cdot 10^{13}$ 1.62 W/g	0,9 W/gr	0,2 W/g	0,5 W/g	0,6 W/g
Gas used and pressure	Helium 2kg/cm ²	Helium 2 kg/cm ²	CO ²	Helium	Helium
Temperature	1.230° ± 20°	400° ± 10°	350°	300°C 450°C ± 20°C	350°C 800°C
Number of hours operation in pile	3.000 h	> 3.000 h	Variable > 3.000 h		

CONTROL CABINET FOR "HEBE" FURNACE

The control cabinet described below is designed for use with a furnace fitted with four heating elements.

The control cabinet is rectangular in form; its dimensions are as follows:

a) outside dimensions:

- height 1960 mm
- width 625 mm
- depth 665 mm

b) inside dimensions:

- height 1770 mm
- width 575 mm
- depth 615 mm

The cabinet is mounted on four castors, the front two of which are of the pivoting type.

The cabinet comprises five sub-assemblies:

- 1) power supply
- 2) power control and recording
- 3) safety devices
- 4) temperature regulation and recording
- 5) control panel

Sub-assemblies nos. 2, 3 and 4 are mounted on 3 mm thick aluminium decks measuring 520 x 520 mm.

All electrical connections to these decks can be unplugged. Each deck is completely independent as regards electrical distribution, so that checking or repair work can be carried out quickly outside the cabinet. Component layout has been designed with a view to maximum accessibility and to enable speedy dismantling without damage to circuitry. Most of the circuits are in the form of plug-in plates.

Side and rear cabinet panels can be removed.

The cabinet is connected by means of a 15 m long, three 5.5 mm² conductor type 750 CMC cable to a 220 V 3-phase or 380 V 3-phase 10 kVA supply (depending on type of transformer connection.) A 3-pole main switch is provided on the control cabinet input.

1 - POWER SUPPLY

The power stage comprises (see drawing n° EA 149 391 1/5):

- Four 2.5 kVA isolating transformers protected by four 15 A porcelaine fuses. The secondary side is of the multiple plug type to give the following voltages: 80, 90, 100, 110, 120, 130, 140, 150 V.
- Four pairs of SESCO type 2 N 1848 silicon thyratrons, mounted head to tail on vaned radiators
- Four Cehess type A4 - UR 16 quick melting safety fuses to protect the thyratrons
- Four thermocouple shunts for flow measurement in the various parts (Metrology laboratory type, rating 0.15 A)
- Two Télémécanique type CN 1 EB 141 four-pole contactors to energise heating elements. These are controlled by the safety chain.

Component supply outlets are located on a female Socapex type EF 413 AB plate.

The power stage is located at the bottom of the control cabinet; the thyratrons and various transformer secondaries are accessible from the side after the side panels have been removed.

2 - POWER CONTROL AND RECORDING

Principle -

The thyratrons mounted head to tail in the circuit of each resistor are controlled in such a way that they actually conduct for only a variable part of their "conducting period" (the period during which the thyatron anode-cathode voltage is positive).

Description -

The assembly forms the "trip deck" (diagram EA 149 391 2/5)

It consists of four identical arrangements each comprising:

- A relaxing oscillator UJT type 2 N 1671 (see drawing EA 149 392)

which supplies impulses to trip the thyratrons.

The frequency of oscillation is directly related to the time constant of a self capacity circuit.

A potentiometer controlled by a miniature motor forms the resistance of this circuit, which varies as a function of the difference between the temperature setting and the actual temperature in the furnace.

- A synchronising stage (see drawing EA 149 392) consisting of a transformer connected to the same phase as the pair of thyratrons, and a Zener diode form circuit. This serves to synchronise the oscillator impulses with the thyatron supply phase.
- A Fontaine type stabilised 24 V 0C-500 mA supply for the above equipment.

3 - SAFETY DEVICES

The safety deck comprises:

- a mains relay with 3-second time lag on opening
- a set of Logimors type semi-conductor static relays for fault indication (contacts close in case of fault.)

The functions of this assembly are as follows:

	ORDER	ACTION
Pressure	Contact closes when pressure is slightly below normal operating pressure - 1.5 kg/cm^2 for op. pressure of 2 kg/cm^2	Lamp on control cabinet Remote transmission to control room possible
	Contact closes when pressure falls to minimum - 0.65 kg/cm^2 for op. pressure of 2 kg/cm^2	Lamp on control cabinet Cuts heater supply Prevents re-starting Remote transmission possible
Temperature	Adjustable cam on MECI recorder closes contact when predetermined temperature is exceeded	Lamp on control cabinet Cuts heater supply Prevents re-starting Remote transmission possible
Mains	Fault on supply phase	Lamp on control cabinet Cuts heater supply Prevents re-starting Remote transmission possible

- A Merlin Gerin type 18 V d.c. supply for the various circuits
- A 6 V d.c. supply for circuit interlock, in the form of a plate
- two Siemens type relays at the end of the chain, one for connections to the control room, the other to control the two heating circuit contactors.
- Three tumbler switches to short-circuit the safety devices.

Remote transmission of safety device signals to the control room is by means of a type 10 EM 13 U connection plate.

4 - TEMPERATURE REGULATION AND RECORDING

The regulation deck comprises:

- a safety relay which is energised when the heater circuit contactors are "off" and brings the potentiometers back to zero.
- an 18 V a.c. supply for indicating circuits
- a 24 V d.c. supply for the relays, arranged on boards
- four identical assemblies each comprising:
 - . a potentiometer to adjust the angle of passage of a pair of thyratrons
 - . a Crouzet type reversible motor which controls the potentiometer
 - . two relays which control the motor and reverse its direction of rotation
- the 16-channel, 9-cam MECI potentiometer recorder, with two contacts for range of regulation and contacts for safety devices

The furnace temperature thermocouples and safety device leads are connected to the control cabinet by a Socapex type EF 439 R female plate.

The 18 sample thermocouples are connected by a Socapex type EM 439 R male plate.

A distributor is provided to connect the recorder channels to the thermocouples and to select the regulation thermocouples.

5 - CONTROL PANEL

The control panel comprises three sections:

Measurement -

The measurement section gives voltage and current readings for the four heating elements.

It comprises:

- a 4-way switch. As safety measure, one terminal is isolated between each of the four channels.

- a thermocouple type ammeter graduated from 0 to 15 A
- a voltmeter graduated from 0 to 150 V
- a push button reversing switch which enables voltage measurements to be taken above and below the thyratrons

Heater control -

The heater control section comprises:

- a double push-button switch with two lamps, controlling the two heater contactors
- a set of switches and lamps to control and regulate the four heater elements separately

Safety devices -

The safety device section comprises:

- four fault indicating lamps (pressure, temperature, mains supply)
- a fault indicating lamp which lights up when a safety device is short-circuited
- a push-button safety device reset switch -

- 1 - Rig head with protection cover
- 2 - Samples thermocouples connector
- 3 - Vinyl thermocouple tube
- 4 - Heater element power conductor tube
- 5 - Furnaces atmosphere control tube
- 6 - Samples-thermocouples connections
- 7 - Samples introduction flange
- 8 - Connection box
- 9 - Furnace
- 10 - Connection of heater-coil conductors and power conductors
- 11 - Alumina metallizing insulation
- 12 - Outer furnace tube
- 13 - Water circulation
- 14 - Gas gap
- 15 - Sample holder
- 16 - samples
- 17 - Heat screen
- 18 - Furnace thermocouples
- 19 - Heater coils
- 20 - Water-box

1.2.52

"CYRANO" FISSILE MATERIAL IRRADIATION RIGS

1 - INTRODUCTION:

The CYRANO device is designed for the irradiation of fissile materials at core temperature and power levels equivalent to nuclear power plant conditions..

A series of this rig is being produced by industry.

Leading features are:

- standardisation of the rig
- the upper part can be uncoupled from the lower, active part
- high useful volume
- operating pressures up to 60 kg/cm^2
- thermal barrier can be changed according to requirements (metal powder or wool, gas)
- possibility of gas-flushing (He in the case of fission gas emission rate measurements)
- high linear flow rate (up to 600 W/cm)
- uninterrupted measurement of specific power

2 - PRINCIPLE OF OPERATION:

2-1 - Irradiation parameters are:

- specific power of fuel
- surface temperature of specimen

2.1.1 - Specific power:

This is obtained by selecting for the enriching and the fuel specimen diameter values compatible with the thermal neutron flux in which the irradiation takes place.

These values can be obtained by:

- theoretical or semi-empirical calculations

- measurements made of models,
- extrapolating results obtained from existing systems

The required specific power can be adjusted by varying the distance between reactor core and irradiation capsule by the use of a special irradiation container.

Two methods are used:

- a system of containers which makes it possible to select between 4 irradiation positions. Neutron fluxes vary by about 5% between one position and another.
- up and down movement of the capsule. Should the experiment require it, the capsule can be moved while the reactor is operating.

2.1.2 - Surface temperature of specimen:

Two variable parameters are used to bring it to the value required by the experiment:

- auxiliary electric heating
- type of gas used in thermal barrier annulus (e.g. iron-wool in helium or helium-nitrogen mixture)

2-2 - Measurements -

2.2.1 - Measurement of the heat-flow by calorimetric methods:

The heat-flow released in the specimen is measured by two methods:

- The total heat-flow is measured through variations of the resistance of a heater-coil wound round the irradiation capsule.
- The heat-flow is measured by area-unit by means of four thermocouples placed between specimen carrier and thermal barrier.

Principle of heat-flow measurements:

The rig is placed in irradiation condition and the following measurements are taken:

1st measurement:

It is taken while the reactor is not operating and the primary element of pile-core cooling-circuit operating. Calibration curves are plotted:

$$\Delta R = f(P_e)$$
$$\frac{\theta_1 + \theta_2 + \theta_3 + \theta_4}{4} = g(P_e)$$

where:

P_e = electric power supplied by the auxiliary heater coil

ΔR = variation in resistance of the monitoring resistor when a known electric power is applied.

$\theta_1, \theta_2, \theta_3$ and θ_4 are the readings given by the 4 thermocouples placed between specimen carrier and thermal barrier.

2nd measurement:

After reactor divergence, measurements are taken at various reactor power levels. The values of ΔR and θ are observed when $P_e = 0$.

(N.B. In order to avoid thermal shocks in the specimen, the supply of electric power is reduced during reactor divergence.)

These step-by-step measurements taken at various reactor power levels make it possible to follow the increase of the specific power of the specimen as well as the evolution of temperatures.

The nuclear thermal energy supplied to the rig is plotted on the calibration curve.

2.2.2 - Thermal conductivity in measurements:

Thermocouples placed inside the fissile specimen supply specimen-core and specimen-surface temperatures. By combining these values with those obtained by heat-flow measurements, the thermal conductivity integral of the specimen is calculated.

2.2.3 - Uninterrupted measurement of fission gas release rate:

This is achieved by flushing the fission gases that have accumulated in the space above the fuel by means of a pure helium flow.

The quantity of radioactive isotopes is determined by gamma spectrometry and that of stable isotopes by chromatography.

3 - MAIN CHARACTERISTICS:

3.1 - Temperature range:

The CYRANO rig is designed to cover a range of temperatures up to 700°C at specimen surface with a high linear flow rate up to 600 W/cm.

This upper limit of the temperature range depends on:

- the specific power of the fuel
- the fluid bathing the specimen responsible for the heat-transfer
- the type of thermal barrier

3.2 - Pressure:

The inner shell containing the specimen can be pressurized to up to 60 kg/cm².

3.3 - Useful volume:

The standard rig is designed for fuel specimen up to 16 mm in diameter and 100 mm in length. Various adaptations can be made.

For instance: specimens : dia. 30 mm length 150 mm
 dia. 15 mm length 550 mm

4 - DESCRIPTION

(See sectional view)

The rig is composed of two parts:

- A consumable, lower part (which is replaced after each irradiation) consisting of the irradiation capsule and suspension tube that extends up to 3 meters above the pile-core.
- A "Cold" upper part consisting of a connecting-box between the two parts, inlet pipes for pressurizing the inner shell or capsule (60 kg/cm^2) and the outer counterpressure annulus (50 kg/cm^2); it also contains fission product outlet lines a heater and thermocouples leads for the connection with the control console.

Irradiation Capsule:

The fuel specimen is placed in a 347 stainless steel (I.D. 20 mm) shell on the standard rig.

On this shell are successively arranged:

- a SODERN I NC I 15 thermocoax heater coil,
- a plurality of hair-pin shaped I NCA 10 (nickel element) resistors constituting a temperature probe whose resistors are distributed all over the capsule surface.

A Molybdenum layer is metallized over the whole assembly and honed to 18 mm diameter.

A stainless steel sheath is then applied by a magnetic-shaping process.

This irradiation capsule is then placed inside an outer shell or thimble (I.D. 30, 4 mm, O.D. 34, 4 mm) cooled by circulating reactor pool water.

The thermal barrier between the capsule and the outer shell consists of a lagging of iron-wool or metal sintering in helium atmosphere or mixture of helium-nitrogen at a pressure of 50 kg/cm^2 .

Because of high heat flow rates existing during irradiation (up to 600 W/cm) the hot part of the outer shell around the specimen is made of zircalloy 2 (good resistance to thermal stresses, low neutron absorption). The rest of the outer shell is made of stainless steel. The junction between the two metals is achieved by intermetallic diffusion.

Four Chromel/Alumel thermocouples (O.D. 1 mm) inbedded in the molybdenum layer are placed at 90° diameters between the hair-pin elements of the shaped monitoring resistors.

At the moment, this rig is being used for the irradiation of a stainless-steel cased UO₂ rod (length 100 mm, dia. 11 mm). This rod is placed in the 347 stainless steel inner shell containing NaK or Na pressurized to 60 kg/cm².

The casing of the fuel rod extends into an expansion chamber for fissile products. Two stainless steel pipes, (diameter 2 x 4) connect this chamber to a fission gas recovery chamber located outside the pile. Two NaK traps fitted on these pipes, above the expansion chamber prevent surface pool water accidental contamination by radioactive liquid metal in case of a leak in the casing.

Five thermocouples are placed on the specimen and its casing:

- one Tungsten/Tungsten Rhenium Tantalum-sheathed thermocouple located at the specimen center
- one Chromel/Alumel, 5 mm dia. thermocouple on the surface on the fuel specimen
- three Chromel/Alumel, 1 mm dia. thermocouples on the specimen casing.

V - FEATURES THAT CAN BE MODIFIED TO CATER FOR SPECIFIC REQUIREMENTS

Thermal features:

- Medium in direct contact with the specimen (NaK, Na, He, etc.)

- Type of outer thermal barrier (heat transfer medium):
iron-wool, sintered metal, gas, etc.

Mechanical features (use in other reactors):

- Length of active portion
- Shape and dimensions of upper part

6 - BASIC DESIGN INFORMATION TO BE SUPPLIED FOR AN IRRADIATION PROJECT IN A "CYRANO" TYPE FURNACE

- Specific power of fuel
- Required combustion rate
- Flux at experiment location (gamma, thermal and fast neutrons)
- Type and dimensions of specimen and casing
- Compatibility of specimens and surrounding medium (Na, NaK, gas)

For instance: He flushing for determining the fission gas release rate -
Determination of the integral of conductivity in the case of UO₂, etc.

CONTROL CONSOLE AND SAFETY DEVICES

The control console includes the following items:

Power supply -

Auxiliary heating is controlled by means of a Variac type rototransformer. Voltages and intensities are measured by class 0.5 precision meters in order to get good thermal measurements.

Measurements: (measuring devices)

Thermocouple readings are continuously recorded on a MECI type recorder. The various readings provided by the resistor which constitutes a temperature probe are also recorded, a recorder being fitted to this circuit.

Safety devices:

Safety devices are connected to the rig as it is being used now for the irradiation of a sheathed UO₂ rod, viz.:

- safety devices on the fission product recovery system
- safety devices on the pressurisation system.

CONTROL CONSOLE AND SAFETY DEVICES

The control console includes the following items:

Power supply -

Auxiliary heating is controlled by means of a Variac type rototransformer. Voltages and intensities are measured by class 0.5 precision meters in order to get good thermal measurements.

Measurements: (measuring devices)

Thermocouple readings are continuously recorded on a MECI type recorder. The various readings provided by the resistor which constitutes a temperature probe are also recorded, a recorder being fitted to this circuit.

Safety devices:

Safety devices are connected to the rig as it is being used now for the irradiation of a sheathed UO₂ rod, viz.:

- safety devices on the fission product recovery system
- safety devices on the pressurisation system.

- 1 - NaK traps
- 2 - Bursting sheet design
- 3 - Helium flushing and recovery of fission products
- 4 - NaK pressure line
- 5 - Impervious pass-throughs for piping and thermocouples
- 6 - Conductor-to-coil connection
- 7 - Molybdenum metallizing
- 8 - Heat monitor wire
- 9 - Central and peripheral thermocouples in UO₂
- 10 - Heater coil
- 11 - Sample
- 12 - Four 90° peripheral thermocouples on cylinder
- 13 - Magnetically restricted sleeve
- 14 - NaK
- 15 - Iron wool filling
- 16 - Spacer

H.F.3 FURNACE

1 - INTRODUCTION

The HF 3 device is designed for the irradiation of material specimens at very high temperatures (up to 2,000 °C) in a thermal neutron flux of 5.10^{13} n/cm²/sec.

This rig is manufactured by private companies.

Its leading characteristics are :

- R.F. heating of specimens.
- Specimen temperature can be maintained after reactor shut-down.
- The lower part can be uncoupled from the upper part.
- Helium flushing.
- Small volume of gas-circuit piping between the specimen and the furnace outlet which reduces the response-time in case of continuous analysis.
- Wide temperature range (1,000-2,000°C)
- Temperatures are measured by means of W/W-Rh thermocouples.
- It is possible to move the specimen inside the inductor while the reactor is operating.
- It is possible to replace worn parts in a hot cell.

2 - OPERATION

The induced electric power in the susceptor is about 1 Kw.
In special cases, the susceptor may be the specimen itself.

The specimen is placed at once :

- In a region where there is a low gradient of thermal neutron flux (in the core median plane)
- In a region with a low thermal gradient (the specimen can be positioned in the center of the inductor).
- In an argon circuit affording accurate and continuous dosimetry.

The specimen may also be flushed by a gas : the return path is in grey on figure 1.

3.- MAIN CHARACTERISTICS

- Temperature range

The H.F.3 rig is designed to cover a temperature range from 1,000°C to 2,000°C. The lowest attainable temperature is limited by heating of nuclear origin.

- Temperature adjustment accuracy

It is that of the W/W-Rh thermocouples. The use of specially calibrated thermocouples brings the error down to 1%. The calibration of these thermocouples is usually carried out by the I.N.S.A. in Lyons. Temperature stability: deviations from posted temperatures are less than one °C (CAT regulation system).

Temperature homogeneity is achieved through sliding the specimen up and down the inductor.

- Usual operating pressure :

Helium at 0,8 atm.

- Useful volume

The in-flux volume is 10mm in diameter and 40 mm in length.

- Irridiation

Specimen irradiation can be either static or helium-flushed.

An argon circuit affords continuous rating of thermal neutron flux.

4 - DESCRIPTION (see figure 1)

The rig consists of :

- the furnace proper or lower 'active' part.
- the upper part containing the connection devices to the furnace.

Lower part :

It comprises :

- 1) A bell which constitutes an impervious chamber supporting consumable parts 11 and 16.
- 2) The lower portion of line (9) leading to furnace conductors (17) and (18) which support inductor (15). A bellows mechanism (7) positioned on top of the bell permits a sliding movement of the thermocouple inside the inductor in order to check the isothermal condition of the furnace.
- 3) Susceptor(16) and specimen-carrier (11) both consumable and the only parts that are subjected to very high temperatures.

The specimen-carrier is capped by a connection-box.

Upper part :

The whole of the upper furnace part has been conceived like a "meccano" construction so that any faulty or inadequate part can easily be changed.

Components:

- 1 R.F. line (12)
- 1 cluster of gas-pipes (8)
- 1 cluster of electric-lines (2)

The gas- and electric clusters can be adapted to the user's needs. Two argon pipes (8) are connected to the lower part of the furnace.

RIG HEAD

It includes :

- the connection with the measuring devices.
Helium inlet and outlet for measuring purposes.
- the connections for the flux-measuring device
Inlet and outlet of argon gas used for the continuous rating of the neutron flux.
- the connection with the R.F. generator includes a flexible conductor for transmitting the R.F. energy.
- the connection with the control-console includes
a coaxial conductor for voltage-rating and oscillogram inlet.

- . an inlet for a 2,000°C pyrometer.
- . a contact pressure-gauge.
- a device for raising and lowering the specimen inside the inductor.

6.- FEATURES WHICH CAN BE MODIFIED TO CATER FOR SPECIFIC REQUIREMENTS

Thermal and electrical features

- . Diameters of inductor, susceptor and, eventually, of specimen-carrier.
- . Metal constituting the susceptor.
- . Height of the inductor.

Mechanical features (adaptation to other types of reactor)

- Length of active part.
- Shape and size of upper part.

7 - BASIC DESIGN INFORMATION FOR AN IRRIDIATION PROJECT IN A H.F.3. TYPE FURNACE.

- Required extreme temperature of the irradiation cycle.
- Required fluxes: gamma thermal or fast neutrons.
- Type and size of specimens.
- Compatibility between specimen-carrier and specimens.
- Unloading frequency of specimens.

8 - TYPICAL APPLICATIONS

Three such furnaces have been used in the MELUSINE reactor; two more are being built. Some UO₂ specimens have been in the course of irradiation since June 1963. Twenty-two irradiation programmes of from 100 to 120 hours each were carried out until June 1965. A 500-600 hour irradiation duration without any unloading corresponds to the most rational utilisation of the furnace.

Irridiation programmes have recently been doubled.

The possibility of moving the specimen inside the furnace while the heating is on makes it possible to carry out continuous studies of the temperature given the specimen position, as well as very quickly submit the specimen to various temperatures (with reactor and furnace both operating).

ELECTRIC CIRCUITRY, CONTROL & REGULATION

R.F. transmission :

- a tuning box.
- a R.F. flexible power line, diameter 25mm.

R.F. Generator :

It includes :

- A 6 Kw oscillator.
- A step-up filter-rectifier.
- A 50 c/sec. upstream voltage regulator.

REGULATION & CONTROL CONSOLE

It includes :

- One recorder.
- One recorder-regulator.
- Alarm and safety devices.
- A place is provided for the control oscillator.

GENERAL STRUCTURE OF H.F. 3 (Feb. 1966)

- 1 - Pin-assembly for quick connecting and standard HF3, 5 and 6 tuning-box.
- 2 - Thermocouple leads
- 3 - Terminal strip
- 4 - Pivoting connecting member
- 5 - Connection-box
- 6-7 - Bellows mechanism (for vertical adjustment of specimen position, with furnace and reactor operating).
- 8 - Gas piping (fission product analysis and continuous dose-metering by means of argon)
- 9 - 10 x 18 R.F. power line
- 10 - Bell (can be uncoupled in a hot cell)
- 11 - Specimen-carrier (consumable)
- 12 - R.F. 16 x 26 power line (1965 type)
- 13 - Standard pivoting bends
- 14 - Adaptor (16 x 26 on 10 x 18)
- 15 - Inductor (consumable)
- 16 - Susceptor (consumable)
- 17-18 - Furnace conductors
- 19 - Water-box

Figure 1

1.2.68

IRRADIATION RIG HF5

I - INTRODUCTION

The HF5 rig is designed for the irradiation of structural materials or fissile materials at controlled very high temperature and a high burn-up.

Its leading characteristics are:

- Irradiation temperature is obtained by means of high frequency induced currents.
- Very high electric and dissipative powers
- Possibility of keeping the specimen at the required temperature even after reactor shut-down
- Specimen can be loaded and unloaded easily
- The lower part is interchangeable
- Experiments carried out in gaseous atmosphere
- The thermal neutron flux can be measured continuously

II - PRINCIPLE OF OPERATION

The heating portion or coupling unit is separated from the heated portion (specimens and specimen-carrier). Induced currents are passed through the specimens or the specimen-carrier (which is conductor) which are thus heated, without electrical or physical contact, by Joule effect. This arrangement makes it possible to keep at a low temperature all the heating parts whereas only the specimens and the specimen-carrier are raised to a very high temperature.

Cooling of the heating parts is achieved by a helium layer about 2 mm thick. A high electric energy, corresponding to a 40 W/cm^2 dissipation, can be inductively applied to the specimen-carrier. This energy is carried away by water.

The attainable irradiation temperature depends upon the following parameters:

- nature of the gas layer (helium, argon or helium/argon mixture)
- clearance between the specimen-carrier and the cooling coupling unit.

III - MAIN CHARACTERISTICS

Temperature:

Irradiation temperature is limited by the mechanical strength of the structural material of the specimen-carrier (tungsten, molybdenum, niobium or carbon) and the maximum amount of energy that can be dissipated at the coupling-unit level. The amount increases with the electric resistivity of the specimen-carrier. 40 W/cm^2 can be achieved a molybdenum helium-flushed specimen-carrier.

Above a minimum temperature, dependent on the heating inside the samples of nuclear origin, with this rig, it is possible to obtain a thermal cycle independent from the reactor power.

The device makes it possible to maintain the required specimen-temperature when the reactor is shut-down.

Pressure:

The normal operating pressure is 0.8 atm. in Helium, He/Argon, Argon, etc. ...

IV - DESCRIPTION

(See sectional view which shows:

- A cold part or upper-furnace built around the H.F. power line.

It includes also:

- A connection-box for connecting the lower and upper parts as well as biological protecting devices.
- An extension of the thermocouples leads
- A cooling water pipe
- Connecting members

- A hot part or lower furnace that can be uncoupled from the upper furnace. It is built in A5 aluminium.

At reactor core level, the furnace comprises:

- A coaxial H.F. input line,
- two cavities bored in a low electric resistivity material, cooled by circulating water; the two cavities are connected by a vertical slot.

The larger cavity constitutes the secondary of a step-down transformer; the smaller one couples the specimen-carrier to the larger cavity.

The specimen-carrier is made of refractory metal. Its shape and nature depends on the irradiation parameters.

It includes:

- A specimen-carrier receiving the induced current (susceptor)
- an extension tube containing the thermocouples
- a fluid-proof connection-box casing
- water-cooled cavities arranged near the specimen-carrier and in which dosimeters are lodged.

V - FEATURES THAT CAN BE MODIFIED TO CATER FOR SPECIFIC REQUIREMENTS

Electrical and thermal features

- Nature and shape of the specimen-carrier
- Clearance between specimen-carrier and coupling-unit
- Type of heat-transferring gas

Mechanical features

(For adapting HF5 to other reactors)

- length of active part
- shape and size of upper part

VI - BASIC DESIGN INFORMATION TO BE SUPPLIED FOR AN IRRADIATION PROJECT IN A HF5 TYPE FURNACE

- Lower and higher temperatures of the required irradiation cycle
- Required flux (gamma, fast or thermal neutrons)
- Type and size of specimens
- Compatibility between specimen-carrier and specimens
- Unloading frequency of specimens

ELECTRIC CIRCUITRY, CONTROL AND GOVERNING DEVICES

H.F. Transmission

It includes:

- a tuning box
- a flexible H.F. power line, dia. 25 mm.
- a variable self-induction coil integrated in the generator

H.F. Generator

It includes:

- a 6 KW oscillators
- a step-up filter rectifier
- a 50 Hz upstream voltage governor

THE CONTROL & REGULATION CONSOLE includes:

- 1 recorder
- 1 recording governor
- alarm and safety devices
- a place provided for the control oscilloscope

- 1 - FR power line, inner conductor
- 2 - FR power line, outer conductor
- 3 - Aluminium shell
- 4 - Water inlet pipe
- 5 - Dosimeter tube
- 6 - Water outlet
- 7 - Induction coil
- 8 - Sample-holder position in coupling unit
- 9 - Detail of independent cooling system in coupling unit
- 10 - Insulating support of RF power line
- 11 - Water-box

UNIQUE PROBLEMS AND SOLUTIONS
ON GETR MATERIALS IRRADIATION
CAPSULES - STRAIN CYCLE #10 & 11

by

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Work Performed Under
U. S. Atomic Energy Commission
Contract AT(04-3)-189
Project Agreement 13

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UNIQUE PROBLEMS AND SOLUTIONS
ON GETR MATERIALS IRRADIATION
CAPSULES - STRAIN CYCLE #10 & 11

The designs of material irradiation capsules continue to be highly complex in order to meet the needs of the experiment. Unique problems must be solved because of the specific critical requirements.

This paper will discuss the design of Strain Cycle #10 and #11. Specifically, I would like to discuss three design problems on Strain Cycle #10 which had novel solutions. I will conclude the paper with a brief description of a proposed capsule design, Strain Cycle #11.

Strain Cycle Capsule #10 was designed for operation in the GETR to irradiate candidate superheat fuel-rod cladding materials at elevated temperatures. The materials included Incoloy-800, Inconel-625, Incoloy-825 and Hastelloy-X.

The purpose of the experiment was to irradiate these materials as tubing and as subsize tensile specimens. Post-irradiation testing of the specimens would provide data for the studies on (a) the mechanisms causing the loss in elevated temperature ductility in these alloys; (b) the effect of a carbide agglomeration heat treatment on improving the post-irradiation ductility; and (c) the ductility of the materials tested under biaxial stress conditions.

The irradiation objective was to accumulate approximately 1.5×10^{21} nvt fast neutron ($E_n \geq 1$ mev) exposure at the peak flux position. The capsule was inserted in the GETR on August 25, 1965 and removed on April 24, 1966. The capsule was at temperature for a total of approximately 4300 hours.

1. GAS GAP DESIGN

The Strain Cycle #10 capsule required a specimen temperature of 1300°F in subcapsules #1 and #2 and 1000°F in subcapsule #3. These temperatures were reached with gamma heating of the specimen material and a Zircaloy-II filler piece which supported the specimen. A gas gap separated the filler piece from the capsule containment.

The capsule had to be designed so that the temperature along the five-inch-long tubular specimen would be uniform. Figure #1 shows the flux shape over the specimen length in subcapsule #1. The flux shape is converted to gamma heat rate and a radial heat transfer analysis is calculated at both ends. For a constant gas gap, a difference of 300°F is apparent between the ends of the specimen. At the desired specimen temperature, a variation of 100°F was considered the maximum allowable. A variable gas gap design was used in order to bring the variation within the allowable 100°F.

Figure #2 shows the temperature difference if the gas gap takes on the three different shapes of constant, continuously variable, and stepped. The continuously variable gas gap would give best uniformity in temperature. This was compromised in the actual capsule to a stepped design. Close tolerances are required with gas gaps and the stepped design was chosen for ease of machining.

The actual temperature variations in-pile were 125°F. The variation seemed to narrow as each cycle progressed, being maximum at the start of cycle. End effects, which were not considered in the analysis, seem to be contributing to the temperature spread. The end effects, increase in number of stepped gaps and increase in the depth of the steps, will be considered in the next proposed capsule, SC-11.

2. BURST SPECIMEN DESIGN

The burst specimen is the payload of the experiment. The requirements were (a) maximum number of specimens per subcapsule; (b) utilization of the space within the empty burst tube for irradiation of tensile specimens of the same material; (c) design the burst specimen for remote handling; and (d) testing in a "hot" lab.

Figure #3 shows the final design of the burst tube specimen. A modified Autoclave Engineering high pressure fitting was used at one end. Welding could not be done on the front end of this fitting since it has a polished tapered sealing surface. Therefore, in order to seal this end, a tube plug welded on the front end was inserted through the fitting and welded on the other end. After irradiation, the plug weld is accurately cut off remotely. An

FIGURE 1. SC-10 CAPSULE IN RELATION TO FLUX

FIGURE 2. TEMPERATURE VARIATION ACROSS VARIOUS GAS GAPS

FIGURE 3. BURST SPECIMEN

FIGURE 4. HIGH PRESSURE FITTING
1.3.5

Autoclave Engineering gland nut and collar will be screwed on the fitting. The entire assembly will then be connected to an Autoclave Engineering high pressure fitting which is connected to a high pressure system. Figure #4 shows a photo of the end fitting.

The loading of tensile specimens within the burst tubes also presented a problem. The shape of the specimens would produce non-uniform gamma heating along the tube length. Figure #3 shows the spacers used around both the round and flat tensile specimens. This simulated a solid rod of material inside each tube. Thermal expansion of the parts was considered and each tube contained a helium atmosphere for best heat transfer across any remaining gaps.

3. ALUMINUM COATINGS

This is not so much a design problem as it is a fabrication problem. Its solution has applications on many capsules.

The gas and thermocouple leads were to be brazed at the capsule penetration joints using a high nickel brazing alloy. Because a vacuum induction brazing furnace was not available for this job, silver soldering with Easy-Flo #3 was substituted.

Silver soldering of capsule leads had been prohibited on capsules irradiated in the GETR because of the bare silver in contact with the reactor coolant water. An aluminum coating applied to soldered joints of a preceding capsule showed no deterioration of the coating. Post-irradiation inspection showed it had remained intact under conditions of vibrations, heat, water flow and high radiation for 5200 hours. With this successful experience, the coating was also used on SC-10.

The coating used was SermeTel, Type W, and is manufactured by Teleflex Incorporated. It is an inorganically (ceramically) bonded aluminum coating.

Note: Post-Irradiation Examination of SC #10

The SC-10 capsule was removed following the presentation of this paper at the "Developments in Irradiation Capsule Technology" meeting in May 1966. Examination in the "hot" laboratory showed excessive flaking of the Sermetel "W" coating which indicated a lack of bonding of the coating to the capsule end caps. SermeTel was still partially present around the silver soldered penetrations and the seals were still intact.

The conclusion that has to be made at this time is that aluminum coating of soldered joints is not recommended until a better bonding material can be found and satisfactorily tested.

INSTALLATION DWG. FOR STRAIN CYCLE NO. 11
 SK-56100-295
 J. B. Miller MAR 31, 1966 IPO-VAL

FIGURE 5. STRAIN CYCLE NO. 11

FIGURE 6. PROPOSED SC-11 SUB-CAPSULE

4. STRAIN CYCLE #11 (PROPOSED)

Capsule SC-11 is a design proposed for a series of in-reactor strain cycling experiments. The basic capsule design is very similar to the in-reactor fatigue test described by Dr. M. B. Reynolds at the October, 1963 capsule meeting. A copy of his report will be found in TID-7697, section 2.20.1.

The objective of this proposed experiment is to establish strain cycle behavior of superheat cladding materials. This has to be done at elevated temperatures and after exposure to neutron dosages equivalent to those accumulated by cladding in service near the end of its desired fuel service life. The peak exposure is estimated to be about 1.6×10^{21} nvt fast neutron in a thermal superheat reactor and greater than 10^{22} nvt in a fast reactor.

The scope of the experiment includes the testing of both Incoloy-800 and Inconel-625 tubing materials in the GETR. Dr. Reynolds' capsule tested three 1.125-inch diameter by .016-inch wall tubes per capsule. The tubes were strained with up to 2000 psi inert gas pressures. Neutron exposure before cycling was 1×10^{20} nvt. SC-11 has a proposed design for simultaneously irradiating and testing twenty-one 0.450-inch diameter by .012-inch wall tubes. Straining will be done with pressure up to 6000 psi.

Figures #5 and #6 show the proposed capsule design. Each capsule has three subcapsules similar to previous Strain Cycle capsules. Each subcapsule has seven test samples. The test sample contains the specimen tube and the inner and outer mandrels which are used to control the total strain. The pressure system will be similar in operation to that used for Dr. Reynolds' capsule.

The new design problems involved in this capsule are: (a) the design of a small reliable test sample; (b) the design of lead-out system capable of handling up to 60 individual gas and thermocouple leads; (c) the design of the high pressure system for combined operation of twenty-one independent systems. All components will be out-of-pile tested prior to the fabrication and irradiation of the actual in-reactor capsule.

Steel Irradiation Capsule and Rig

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I. Introduction

The Steel Irradiation Joint Committee of Japan (The Iron & Steel Institute of Japan, Japan Society for Promotion of Science, Japan Welding Engineering Society and Japan Atomic Energy Research Institute) started the steel irradiation program in 1963 with a partial subsidy from the Japanese Government. The first irradiation rig consisted of 4 assemblies containing 192 steel specimens. Mainly, Charpy test specimens and a small amount of tensile and basic research test specimens had been irradiated in October to December, 1965, at BR-2 in Belgium. The second irradiation rig having almost same configuration with 225 specimens had also finished the irradiation in March, 1966, at the same position in BR-2. The initially planned irradiation conditions are as follows in both cases.

- a. Fast neutron dose (over 1 MeV) for most specimens : 3×10^{19}
 $\pm 10\%$ nvt and for some specimens : $1 \times 10^{19} \pm 10\%$ nvt
- b. Specimen in-pile temperature for high temperature irradiation:
 $260^\circ\text{C} \pm 50^\circ\text{C}$ and for low temperature irradiation: under 150°C
- c. Specimen clad : Aluminum alloy

The requirements for fast neutron dose were altered later, considering actual fast neutron flux distribution in BR-2. The final ones were that the aimed value must be in the range of $\pm 15\%$ of 3×10^{19} nvt and the variation in one group of specimens located in one assembly must be in $\pm 10\%$ of the mean value.

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Because the irradiation channel in BR-2 had a heat flux as high as 6 W/gr, any heater or gas flow temperature regulating system could not be applied to get the planned high temperature. A new design for high temperature capsule having a temperature regulating system had been tested at out of pile successfully, and then the irradiation capsules having the system were fabricated by Mitsubishi Atomic Power Industries, Inc. (MAPI).

To realize the planned fast neutron dose of over 1 MeV, the irradiation rigs were rotated and moved axially out according to the movement of peak plane of fast neutron flux during the irradiation. The irradiation rigs were designed and fabricated by the Societe d'Etudes, de Recherches et d'Applications pour l'Industries (S.E.R.A.I.). The irradiation has been performed successfully at BR-2 under S.E.R.A.I.'s supervision.

II. Capsules for Low and High Temperature Irradiation

Photo. 1 shows several capsules of different length for low temperature irradiation and 2 capsules for high temperature irradiation. The specimens for low temperature irradiation were simply covered by an aluminum alloy clad. The longest one can contain 6 standard V-notch Charpy test specimens. Six of these capsules were located in the same assembly to be able to get the 12 Charpy test specimens in one test group having the fast neutron dose variation within $\pm 10\%$ of the mean value. That is, the 12 Charpy test specimens were arrayed at the same plane in the irradiation rig even they belonged to the indifferent capsules. Figure 1 shows general configuration of the rig. Dosimetry wires of Co, Fe, and Ni, or Co and Ni were put at each position corresponding to V-notch plane. The fast neutron dose was measured by the reactions $\text{Fe}^{54}(n,p)\text{Mn}^{54}$ and $\text{Ni}^{58}(n,p)\text{Co}^{58}$, both reactions presenting approximately the same effective threshold energy (~ 3 MeV) and being used to check each other. The Co wires were used to correct Co^{58} burn-up by thermal neutrons in the Ni wires. As a conclusion, both the measured doses for Fe and Ni wires showed a good agreement. The temperatures during the irradiation were measured by sheathed thermocouples put in dummy specimens. The temperature records showed less than 90°C . The variations remained within $\pm 7^\circ\text{C}$.

For the high temperature irradiation capsule, the γ -heating in the irradiation channel was utilized to get the planned specimen temperature. The capsule consisted of an aluminum body made by extrusion into which the specimen container was inserted. Heat flux between aluminum body and specimen container was reduced by milled grooves. Additionally, the specimen container had flat chambers containing a low melting point alloy. The temperature regulating action of this system is based upon the fact that heat transfer across the chamber increases when the alloy is molten. See Figure 1 Section A - A. The calculation and the out-of-pile thermal experiment showed the specimen temperature could be kept in the planned

temperature range for the γ -heating of 6 w/gr $\pm$ 2 w/gr which was an anticipated γ -heating in BR-2 irradiation channel E 30. The low melting point metal layer was designed to act temperature flattening in case γ -heating was beyond this range to some extent. This layer provides a new concept of high temperature irradiation capsule.

Actually the γ -heating in the irradiation channel were reported as 4.6n~5.7 w/gr (Peak value on the channel axis) for the first irradiation rig and somewhat lower than it for the second irradiation rig. The temperature records for the first and second irradiation showed the mean specimen temperature was in the planned range and its variation during the irradiation was lower than $\pm 20^\circ \text{C}$ in the worst case as shown in Figure 2. The differences of temperature indicated during the irradiation were caused by the position moving of thermocouple due to the rig rotation.

III. Rig and its Irradiation

The irradiation rig consisted of following parts as shown in Figure 3.

- the retractable and rotatable rig, about 8 m long, whose main component is the stainless steel cylindrical housing, 2415 mm long, 78 mm external diameter, containing the 4 groups of specimens with 8 thermocouples and the sets of flux monitors (Co, Fe, Ni).
- the channel plug, to be locked in the reactor pressure vessel top cover.
- the pneumatic blocking mechanism, enabling either to block the rig in position or to free it for axial displacements or rotations.

Total displacement length of the rig from the lowest irradiation position to the position where the rig was fully removed out of the reactor core region was 2.7 m.

In the cylindrical housing containing the capsule, the length of the individual specimen groups was maximum 280 mm and the spacing between adjacent specimen groups had a length of about 400 mm, in order to respect the required close grouping of specimen dosages.

Reactor primary cooling water flew freely inside the cylindrical housing, through the numerous holes provided in the outer casing and in the different intermediated flanges to which the capsules were attached.

One particular feature of the cylindrical part containing the capsules was the type of centering device made from 6 balls located between movable conical flanges. This device allowed easy translation and rotation of the rig and adapted itself to the different internal diameters of the irradiation channel.

Photo 2 shows the rig insertion into the irradiation channel.

During the irradiation, a direct measurement of the thermal neutron flux distribution was made daily, using a calorimetric device, along the axis of another channel (A 330). This distribution could be directly converted into the fast (over 1 MeV) neutron flux distribution in the specimen region in E 30, since it had been established before the rig irradiation that there was a nearly constant ratio between thermal flux in A 330 and fast flux in E 30 region of interest. By this method, the average fast neutron flux along the specimen length was calculated daily and consequently, the "remaining" period of time for obtaining the required dosage could be determined.

The upward displacement of the peak flux plane could be simultaneously followed and the rig position was corrected by steps of about 5 mm every 8 hours (Figure 4). The rotation of rig over 120° at regular time interval were required to eliminate the variation of neutron dose due to the radial fast neutron distribution as shown in Figure 5.

It was also experienced that axial motion and rotation of the rig in pile produced more pronounced reactions from BR-2 automatic control system than had been anticipated. Consequently, it was decided that, for subsequent rigs, pneumatic blocking action should be replaced by a more reliable mechanical device in order to avoid any possibility of reactor trip due to accidental rig falling. For the same reason, the originally designed continuous rig rotation method had to be abandoned and replaced by rig rotations over 120° at regular time intervals, generally every 8 hours, using the displacement tool.

The total fast neutron dosage curve calculated for one particular specimen group by multiplying each day the fast neutron flux values by the corresponding irradiation time and adding the products obtained in this way. The obtained dosage results indicate that the $\pm 10\%$ maximum permissible deviation requirement in one group of specimens located in one assembly seems to have been actually met.

IV. Conclusion

Two rigs have already been irradiated successfully in BR-2 and the post irradiation tests on most specimens were performed in Belgium. The specimens for basic research are being tested in Japan Atomic Energy Research Institute. The aforementioned precautions concerning capsule and rig design, fabrication and irradiation must give an essential contribution to the good performance of the steel irradiation tests.

Acknowledgement:

The authors wish to express their sincere thanks to the Steel Irradiation Joint Committee of Japan for their permission to publish the report and to Japan Atomic Energy Bureau for their support to achieve the work.

Photo 1 High Temperature Capsules and Low Temperature Capsules

Fig.1 General configuration of rig

RIG: MSR-A-CHANNEL: E30
SPECIMEN GROUP: A3
TEMPERATURES INDICATED BY THERMOCOUPLES
N°5 AND 6
FIG. 2

IRRADIATION DEVICE

SCHEMATIC

FIG. 3

Photo 2 Rig Insertion in BR-2

FAST NEUTRON FLUX (>1 MeV)

AXIAL DISTRIBUTION

FIG. 4

MAXIMUM RADIAL DISTORSION OF

FAST NEUTRON FLUX (>1 MeV)

FIG. 5

LIMITED TEMPERATURE AND FATIGUE TEST CAPSULE EXPERIMENTS*

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Prepared for presentation at the Capsule Irradiation Experiments Meeting, May 3-5, 1966, at Pleasanton, California.

ABSTRACT

Two unique capsule designs and associated problems are discussed.

A splined type aluminum tube was used to encapsulate ATR surveillance beryllium samples. A shrink fit was employed to insure a good metal contact between the samples and the tube. It is shown how this design met the requirements of maintaining a sample temperature of less than 120°C, allowed void space for gas release, and did not excessively constrain the growth of the sample.

A comparison is made with a more standard designed stainless steel capsule which indicates the effectiveness of the splined tube.

A capsule for evaluating the irradiation effects on the fatigue properties of steels has been designed which permits cycling a specimen through an axial tension and compression cycle while in-reactor. The cycling can be controlled by the magnitude of stress or strain on the sample. Provisions have been made to measure and record the stress, strain, hysteresis energy, and the number of cycles to failure.

The capsule is single containment which allows the more massive structural components to be cooled by reactor process water. The specimen train is

*Work performed under Contract AT(10-1)-205 to the Atomic Energy Commission.

**Formerly with Phillips Petroleum Company, Atomic Energy Division.

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temperature controlled and cooled with high purity helium. The system has been designed to hold to within 6°C of test temperature during reactor transients.

The capsule or pressure tube, the sample, the cycling components, the controlling and data measuring components, and the means of controlling temperature are discussed.

LIMITED TEMPERATURE CAPSULE

Background:

These capsules, a total of 64, were designed for the purpose of encapsulating ATR surveillance beryllium samples. The capsules were inserted in the ETR to build-up "lead time" on the ATR beryllium. After the ATR becomes operational, they will be transferred to the ATR then removed in a programmed sequence for surveillance purposes.

Design Criteria:

The capsules had to be designed to meet the following requirements:

- (1) The samples should not exceed 120°C.
- (2) There should be a void space sufficiently large so that the gas released during irradiation would not build-up an excessive pressure.
- (3) Growth and swelling measurements of the beryllium were required, so the stress on the samples had to be relatively small so there wouldn't be any constraint on growth.
- (4) The atmosphere around the sample was to be an inert gas.
- (5) The capsules were to be stacked in a train, four capsules to the train. The capability of separating one or more of the capsules from the train had to be included in the design so that some of the capsules could be removed from the reactor for inspection.

Capsule Design and Operation:

The capsule design for 48 of the capsules was a splined type aluminum tube, as shown in Figure 1, employing a shrink fit on the beryllium samples. The tubes were sized so there would be an interference fit of about 0.0005 inches at room temperature between the samples and tube. Calculations indicated that

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as the beryllium grows with exposure, the aluminum will yield slightly, and that the stress exerted on the samples would be less than 2,000 psi. This was considered to be satisfactory.

The splined tube resulted in good metal-to-metal contact over one third of the sample surface for good heat transfer. A heat transfer analysis using a two-dimensional IBM-7090 computer program indicated a maximum sample centerline temperature of 69°C . It should be noted that both beryllium and aluminum have a high thermal conductivity, which allow the heat to be transferred through the splined areas without high thermal gradients in the sample. These gradients are less than 6°C . The void space in the flutes plus a void section at the top of the capsule allowed for gas release. The total void space amounted to 0.43 cubic inches. It was determined that the total gas release from the beryllium, at standard conditions, would be in the order of 9.15 cubic inches. Assuming the perfect gas law is valid for both the gas released and the argon cover gas, the final pressure at operating temperature would then be about 400 psia. When the reactor is pressurized, the differential pressure across the tube would be about 240 psi. The resulting stress based on a thin wall pressure vessel approach was 2800 psi.

In addition, seven of the capsules had one aluminum clad thermocouple each penetrating the upper head and making contact with the beryllium sample. This contact was made by pressing the thermocouple into a drilled hole in the top sample and then maintaining pressure on the TC while it was welded to the upper end piece.

In addition to the 48 aluminum capsules, 16 stainless steel capsules (four with thermocouples) were made by contract with Battelle Memorial Institute. The beryllium samples in these capsules were submersed in NaK to act as a thermal bond. Heat transfer calculations indicated that both designs would

1.5.5

Crossectional View Of Splined Aluminum Capsule

Figure 1
Crossectional View of Splined Aluminum Capsules.

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limit the temperature as required. The amount of corrosion for long periods of time between the NaK and the beryllium was unknown, which produced an uncertainty in obtaining growth measurements. However, there was a question of the constraining effect from the aluminum capsule, so that both types of capsules were built to evaluate these effects.

The capsules have now been in the ETR over a year. Table I shows the temperature history of the monitored capsules. Two of the stainless steel capsules show a rather wide variation in temperature and have changed substantially since insertion. It is not known whether this is the thermocouples malfunctioning or a NaK-beryllium surface "wetting" problem⁽¹⁾.

TABLE I
Capsule Temperature History

Capsule Temperature °C

Date	Stainless Steel				Splined Tube Aluminum						
	1	2	3	4	1	2	3	4	5	6	7
April, 1965	70	50	125	54	54	56	71	60	77	66	77
May, 1965	93	54	143	60	63	63	82	66	82	71	82
March, 1966	101	56	149	59	38	66	77	64	82	69	77

Assembly Procedure

The aluminum capsules were assembled as follows: (refer to Figure 2)

1. Lower end piece welded to tube.
2. Tubes heated to about 320°C and five samples inserted.
3. Upper end piece welded on.

¹To prevent voids in the NaK and to insure a wetted surface, these capsules were heated to 110°C and rotated at 300 rpm for about five minutes to allow the NaK to settle.

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Assembly Procedure (Cont'd)

4. Thermocouples welded to upper end piece of seven capsules.
5. Capsule evacuated and refilled with argon and seal plug screwed tight into lower end piece (all done in an argon atmosphere).
6. Seal plug welded.
7. Final machining to maintain tolerances and concentricity.
8. Drill holes so capsule trains can be connected together.

Comments on Assembly:

Welding these thin aluminum tubes proved to be very difficult. As a result of this, after each weld was made, the excess material was machined off and the weld pressure-tested before proceeding to the next step.

Due to the fact that aluminum is quite "soft" at the temperature noted above, and that the samples had sharp edges, the samples had to be in good alignment with the capsule before inserting them or they would become "cocked" and complete insertion would become impossible. A special jig (Figure 3) was made to align the samples with the capsules so they could be inserted quickly and easily after the capsule had been heated.

The final seal was made with the seal plug. The welding of the seal plug was a backup measure.

IN-PILE FATIGUE CAPSULE

Purpose:

The purpose of this project is to evaluate the elevated temperature fatigue properties of steels for pressure vessel and structural components principally by in-pile tests to provide design and service life data.

Figure 2 Components of Splined Aluminum Capsule

Figure 3 Splined Capsule Loading Fixture

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To accomplish the objectives, it is necessary to measure the fatigue properties (stress, strain, hysteresis energy, and number of cycles to failure) in an irradiation field.

Design Criteria:

The system was to be designed so that the sample could be cycled in a pure helium atmosphere at any temperature between 200°C and 540°C through both tension and compression. The fatigue cycling can be controlled by either stress or strain. It was required that the temperature be controlled to within 6°C during reactor transients and that the strain could be measured to within 0.0001 inch per inch.

Capsule Components:

To fulfill the requirements, the following basic components were used:

(see Figure 4)

1. Fatigue specimen.
2. Three linear variable differential transformers (LVDT).
3. Load cell.
4. Secondary load transmitting link.
5. Primary load transmitting link (includes radiation shield plug).
6. Electrohydraulic servoactuator, servovalve, hydraulic pump, and controls.
7. Pressure or load tube.
8. Gas heaters and heat exchangers.
9. Source of helium for cooling and temperature control.

The fatigue specimen used was hollow with a 0.235 inch O.D. and 0.125 inch I.D. The hollow specimen with a higher section modulus was selected over the solid one so that buckling or bending stress due to misalignment would not be as significant.

Figure 4
Cutaway View of Fatigue Experiment Components

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There are three LVDT's incorporated in the system. Two of them are mounted on the specimen diametrically opposite each other. These two LVDT's counterbalance each other and one acts as a back up for the other one in measuring the strain on the specimen. In addition, any undesirable bowing in the plane of the highest gamma heat gradient should be detected by them. The signal from these LVDT's goes to the servo controller which gives the capability of controlling the cycling on strain.

The third LVDT is attached to the load cell, which measures the strain on the load cell. Since this strain is well within the elastic limit, the applied force can be determined from which the stress on the specimen can be calculated.

Located just above each LVDT there is a recalibration actuator. This is a pneumatic device which induces a prescribed displacement on the LVDT core, which makes it possible to detect any drifting as a result of radiation damage.

The load cell is anchored to the pressure tube, this being the lower anchor point of the specimen train.

The secondary load transmitting link is a tube about eight inches long that transmits the force to the specimen. There are two diaphragms attached to this link which hold the alignment of the sample train to prevent buckling. Through this section of the sample train, the manufacturing tolerances were held very close because of the alignment problem.

The primary load transmitting link is a tube about 75 inches long which transmits the load between the servoactuator and the secondary load transmitting link. This link is relatively long so the servoactuator can be located above the high radiation area, leaving it in a field of 1×10^{-3} watts/g(C). The primary load link contains a radiation shield plug to shield the servoactuator from streaming.

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The servoactuator has the capability of cycling the specimen ± 0.250 inches at 0.009 to 1 cps. The servovalve is located outside of the reactor about 35 feet from the actuator. As a result of this, the response is flow velocity limited at about 3 to 4 cps.

The corner position E-14 of the ETR core was assigned to this experiment. This location has a gamma flux variation of more than 50% between diagonal corners, which results in a substantially higher gamma heating rate on one side of the pressure tube than on the other. This could result in more thermal expansion on the one side and possible bowing of the tube. A stress analysis was made to determine what effect this would have on the specimen so that a satisfactory design could be made.

The specimen train is rigidly positioned in the pressure tube with two diaphragms on the secondary load transmitting link above the specimen and with the load cell bulkhead below the specimen (see Figure 4). Therefore, any bowing of the pressure tube will cause a small S bend in the specimen inducing an unwanted stress in the specimen. In addition, the specimen cycling load would not be coaxial with the specimen centerline which would result in a moment on the specimen and thereby induce an additional unwanted stress.

In order to keep these unwanted stresses to a minimum, the pressure tube was made out of "Invar" steel, a high nickel content steel with a coefficient of thermal expansion of about 0.7×10^{-6} inch/inch $^{\circ}\text{F}$ at ambient temperature. With this material, the above mentioned stresses are about 800 psi and 700 psi respectively for a total of 1500 psi. As a comparison, if the pressure tube were made of type 347 stainless steel, these stresses would be approximately 14,000 psi and 23,000 psi respectively for a total of 37,000 psi.

Using Invar resulted in an additional benefit in as much as the combined shear stress in the pressure tube due to pressure stress, thermal stress, and axial stress due to working the specimen was only 2665 psi, a very nominal value compared to most materials located in a high gamma heating region.

Invar is not much more expensive than Type 347 stainless steel but is harder to machine. In the region above the core where the gamma heating is negligible, the pressure tube was made out of 347 stainless steel. The stainless steel section was welded to the Invar section using a special filler material called Invarod. Figure 5 shows the sample train assembled ready for insertion into the pressure tube.

Temperature Control:

The capsule is a single containment structure which makes it possible to cool the more massive containment vessel with reactor process water, leaving only the sample train to be temperature controlled and cooled with high purity helium, the primary cooling medium.

There are four separate helium gas streams going to the sample train components. One stream goes to the specimen, one stream to the load cell, one stream to the secondary load transmitting link, and one stream to the LVDT's. The LVDT stream splits and a third of it goes to each LVDT. The specimen, the load cell, and the LVDT's are shrouded to direct the gas stream and to maintain a sufficiently high gas velocity by the components so as to obtain an optimum heat transfer coefficient. After passing the particular component, all four gas streams unite into a common return stream which flows up through an annulus between the primary load transmitting link and the pressure tube thus cooling the primary link. Most of the latent heat from the gas stream is transferred to the reactor water

1.5.15

Figure 5 Fatigue Experiment Sample Train Ready for Insertion into the Pressure Tube

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as it passes up the annulus.

Three of the gas streams, specimen, load cell, and LVDT, have a dual function of cooling and controlling the temperature of the particular component. To accomplish this control, there are two electrical heaters incorporated in each of the three gas streams. The first one, referred to as the line heater, is a slow response heater located on the gas stream outside of the reactor. The second one, referred to as the trim heater, is a very fast response heater located on the gas stream in the in-pile tube just above the primary load transmitting link. During a reactor transient, the trim heater picks up the heating load immediately and maintains the temperature until the line heater can take over, at which time, the trim heater shuts off. This makes it possible to have a quick response system and a long life system.

The specimen is cycled by means of an electrohydraulic servoactuator. The signal to the servovalve comes from the controller which consists of the necessary electronics to apply a predetermined low frequency sinusoidal load on the specimen which approximately determines the cycle stress or strain developed in the specimen. The actual stress and strain developed in the specimen is detected by the LVDT's and recorded on magnetic tape. In addition, circuitry is provided for detecting specimen failure by generating an error signal whenever the load changes by more than a predetermined fraction. The data on the magnetic tape is plotted at a later date with the aid of an x-y plotter which gives the hysteresis curves from which the hysteresis energy can be determined.

SUMMARY

Two unique capsule designs and associated problems are reported.

A splined type aluminum tube was used to encapsulate ATR surveillance beryllium samples. A shrink fit was employed to insure a good metal contact between the samples and the tube to insure that the sample temperature did not exceed 120°C. The design allowed void space for gas release, and did not excessively constrain the growth of the sample.

A comparison is made with a more standard designed stainless steel capsule which indicates the effectiveness of the splined tube.

A capsule for evaluating the irradiation effects on the fatigue properties of steels has been designed which permits cycling a specimen through an axial tension and compression cycle while in-reactor. The cycling can be controlled by the magnitude of stress or strain on the sample. Provisions have been made to measure and record the stress, strain, hysteresis energy, and the number of cycles to failure.

The capsule is single containment which allows the more massive structural components to be cooled by reactor process water. The specimen train is temperature controlled and cooled with high purity helium. The system has been designed to hold to within 6°C of test temperature during reactor transients.

IN-REACTOR MECHANICAL PROPERTIES EXPERIMENTS

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ABSTRACT

The design and development of materials and techniques used in conducting mechanical properties experiments in the irradiation facilities of the Oak Ridge Research Reactor are discussed and illustrated.

Detailed descriptions of fabrication and operation of instrumented capsules used for stress-rupture testing, creep testing, and postirradiation studies are presented. A variety of materials tested include stainless steels, beryllium, Inconel, Hastelloy N, and graphite.

Temperatures of the tests described cover a wide range: stress-rupture testing from 300 to 800°C, creep testing from 300 to 650°C, and postirradiation studies from 250 to 900°C.

Auxiliary equipment such as heaters, loading systems, access tubes, junction boxes, and instrumentation is discussed.

*Research sponsored by the U. S. Atomic Energy Commission under contract with Union Carbide Corporation

INTRODUCTION

In the selection of structural materials for the construction of nuclear reactors or component parts, the effects of irradiation on the mechanical properties are of prime importance. The only way to determine these effects is to conduct tests in radiation fields similar to those in which the material will be used. This paper describes the techniques developed at the Oak Ridge National Laboratory for conducting mechanical property tests in the experimental facilities of the Oak Ridge Research Reactor.

EXPERIMENTAL FACILITIES

Figure 1 is a view of the ORR showing the experimental facilities, access tubes, and junction boxes. The usual experiments conducted in a pool-type reactor must be placed in a container that is sealed to prevent inleakage of water.

Figure 2 shows an experiment ready to be installed in a pool-side facility. Aluminum tubing attached to the container and extending above the core of the reactor provides access to the container for heater leads, thermocouples, and other sensing devices. To this tube a 14-ft length of 1 1/4-in.-diam polyethylene plastic pipe is attached with clamps and, in turn, is connected to an 18-in. length of aluminum tubing that terminates with a flange. The experiment container is fabricated by welding plates of type 6061 aluminum to form a box or can 2 7/8 in. wide x 24 in. high x 12 in. deep. Top and bottom plates are 3/4 in. thick, side plates are 1/4 in. thick, and the front and back plates are 1/2 in. thick.

Control instruments for the experiments are located in a room approximately 70 ft from the top of the pool. As leads of 100 ft would be impractical to handle, a junction box is placed on the poolside above the water level. The cylindrical-shaped box is divided into two compartments. Permanently installed electrical-, thermocouple-, and gas-leads from the control instruments terminate in the lower part of the junction box. Communication to the upper compartment is provided by amphenol

Photo 65785

1.6.3

Fig. 1. Oak Ridge Research Reactor and Experimental Facilities. Original reduced 17.5%.

1.6.4

Fig. 2. Poolside Experiment Ready for Insertion in ORR. Original reduced 12.5%.

connectors and pipe fittings attached to the dividing plate. Figure 3 shows the upper half of a junction box. Leads from the experiment enter the junction box through a round hole in the top and are plugged to the appropriate connector in the dividing plate. The flange on the end of the plastic tube seals the hole.

Fig. 3. Oak Ridge Research Reactor Poolside Junction Box. Original reduced 38.5%.

EXPERIMENT DESIGN AND CONSTRUCTION

Because of the complexity of in-reactor testing, the tube-burst test was selected as the most convenient test from which to obtain stress-rupture data. Following experiments of this type, a more sophisticated creep test was designed and will be discussed later.

Figure 4 is a schematic drawing of a tube-burst specimen. These specimens are fabricated by just cutting to length seamless tubing of the desired diameter and wall thickness or by honing the inside surface or grinding the outside surface of tubing to produce the desired dimensions. The specimens are sealed with slip-fit end plugs welded with an edge fusion weld. A pressurizing line of 1/16-in. OD stainless steel tubing is connected to the specimen as shown in the drawing.

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Fig. 4. Schematic Drawing of a Tube-Burst Specimen.

Regardless of the type of test one is conducting in reactors of high-power density, such as the ORR, gamma-ray heating is an important factor. Nuclear heating may be so great that cooling of the specimens will be required to maintain the test temperature. However, if the test temperature is above that temperature produced by gamma heat, auxiliary heating will be required. A maximum of 1 w/g is used as the nuclear heating value in the design of ORR poolside experiments. This varies considerably in an experiment depending on the distance above or below the centerline of the reactor as well as the distance from the reactor face. Temperatures of from 340 to 600°C are produced by gamma heating in specimens of type 304 stainless steel. It was necessary, then, to add auxiliary heat as the tests were to be conducted from 650 to 871°C.

Several types of heater designs have been used but the most successful and only type used now, is fabricated from MgO insulated Inconel sheathed heater cable. The diameter of the cable is 0.040 in. and contains either a 28-gage Nichrome V wire with a resistance of 4.5 ohms/ft or a 30-gage wire with 7 ohms/ft. Figure 5 shows the

Fig. 5. Metallic-Sheathed Three Section Heater. Original reduced 41%.

various stages in the construction of a heater. This heater is for a tube-burst specimen and is made in three sections, each section being wound on a grooved copper mandrel to give the desired diameter and spacing between the coils. The overall length of this heater is $3 \frac{3}{8}$ -in. with an inside diameter of $\frac{3}{4}$ in. Spacing between the coils is $\frac{1}{16}$ in. The resistance of each section is 15 ohms and contains $3 \frac{1}{2}$ ft of cable. The three sections are made into one unit by spot welding Nichrome ribbon to each coil over the entire length. One of the most important and most used tools in the construction of these experiments is a portable stored-energy type spot welder equipped with a tweezer handpiece. This demonstrates only one of the many times it is used. Flat rings, punched from 0.020-in. stainless steel sheet, then are fastened to each end of the heater to facilitate mounting in the experiment container. The versatility obtained in heater design using the sheathed cable is shown in Figs. 6, 7, and 8.

It is necessary to strip the sheathing material from the Nichrome wire in order to connect the heater leads. Terminals for use with this type of cable are available commercially, but they proved to be bulky

Fig. 6. Metallic-Sheathed One-Section Heater. Original reduced 37%.

Fig. 7. Metallic-Sheathed Rectangular Heater. Original reduced 47%.

Fig. 8. Metallic-Sheathed Flat Heater. Original reduced 43%.

spacewise as well as from the standpoint of creating hot spots as the results of gamma heating. Figure 9 is a schematic drawing illustrating the manner in which the heater terminals are fabricated. Using a hand-operated cutting tool, the sheath is removed so as to expose a 3/4-in. length of wire. A ceramic insulating bead (0.035-in. OD $\times$ 1/8 in long) is slipped over the wire to insulate the heater wire from the sheath. A 3/4-in. length of stainless steel tubing (0.035-in. OD $\times$ 0.010-in.-thick wall) is placed over the wire next and crimped to it several times. Another ceramic insulating tube (5/8 in. long $\times$ 1/16-in. OD $\times$ 0.040-in.-thick wall) slides over the stainless steel and ceramic tubing to provide rigidity and butts against the sheath. A final protection tube of stainless steel (0.080-in. OD, 0.005-in. thick wall, and 1 1/16 in. long) is added, crimped, and spot welded to the sheath. The power lead can now be spot welded to the stainless steel tube protruding from the ceramic insulator.

The power leads are 32 ft long and are made up in two parts: 26-ft lengths of 22-gage asbestos insulated copper wire joined to 6-ft lengths of 22-gage ceramic insulated nickel wire. Because of the

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Fig. 9. Schematic Drawing of Heater Terminals.

embrittlement and thermal fatigue problems, it is not desirable to connect copper leads directly to the heater terminals. The nickel-copper junction is made by silver soldering the wires together inside a short length of stainless steel tubing. A ceramic tube slips over this junction to insulate it. The nickel-lead section is then spot welded to the heater terminal.

The thermocouple leads also are 32 ft long and are of 30-gage Chromel-P-Alumel glass insulated wire. Six feet of insulation is removed from each lead and replaced with ceramic insulation. Joining of the leads to the short thermocouples, previously attached to the specimens, is made by twisting the matching wires together and welding with a capacitance discharge welder. These joints are not insulated and care must be exercised so that a short will not develop.

Figure 10 is a photograph of a completely wired tube-burst experiment. The specimens and heaters are positioned on a stainless steel framework so that the longitudinal axis of the specimen is normal

Fig. 10. Tube-Burst Experiment. Original reduced 39%.

to the face of the reactor core. The heaters are attached by spot welding the end rings to the mounting plate. The specimens, each with its attached 30-ft length of pressurizing line, are positioned inside the heaters and held in place by spot welding the pressurizing lines to the supporting arm. Another invaluable aid in this assembly process is Nichrome ribbon. Besides its use in the construction of heaters, it is handy for tie-down straps for the pressurizing lines, electrical leads, thermocouples, and as small support arms. It is available in widths of from 1/16 to 1/4 in. and thicknesses of from 0.002 to 0.010 in.

In the wiring process it is extremely important that each lead wire be looped such that expansion or contraction of the whole wire bundle will not force undue strain on the heater or thermocouple connections. Figure 11 is a closeup view showing the tube-burst specimen, with its three attached thermocouples inside a heater. In Fig. 12 it can be seen

Fig. 11. Tube-Burst Specimen and Heater. Original reduced 54.5%.

Fig. 12. Tube-Burst Experiment Positioned in Container.
Original reduced 38.5%.

that the bundle of wires, along with an 1/8-in. thin-wall stainless steel tubing serving as an atmosphere line, has been pulled through a partially fabricated can and access tube until the experiment is positioned in the can. The framework is attached to the top of the can with screws. After a check is made of all heaters and thermocouples, the side plate will be welded to the can. Amphenol connectors are attached to the junction box end of the heater and thermocouple leads and the whole system pressurized with helium and checked with a helium-leak detector.

After the experiment is installed in the poolside facility, the junction box connections made, and the junction box sealed, the power and thermocouple leads are connected from a patch panel in the instrument room, as shown in Fig. 13, to the desired control instrument. The output of each thermocouple is fed to a proportioning on-off controller (Fig. 14), which adjusts the power input to one section of a heater.

A schematic flow diagram of the high- and low-pressure system is shown in Fig. 15. The atmosphere to the can, or the low-pressure gas, is supplied from tanks of various gases located outside the building. From

Fig. 13. Electrical - and Thermocouple-Patch Panel. Original reduced 45%.

Fig. 14. Control Instruments for In-Reactor Experiments. Original reduced 45%.

Fig. 15. Schematic Flow Diagram of High- and Low-Pressure Systems.

each supply tank a gas line passes through three regulators in series and terminates in the instrument room at a patch panel. The panel is equipped with quick disconnecting fittings so the experimenter can select the desired gas and patch to the experiment atmosphere line with a short length of flexible tubing. The atmosphere inside the experiment can is maintained at 12 psig so that, with the head produced by the poolwater, the differential pressure across the can wall is reduced to +1 psi. Overpressurization of the can is prevented by three pressure relief valves; one located in the gas supply line and two connected directly to the junction box. These valves are set to function slightly above operational pressure and vent any overpressure to the off-gas system. An additional safety feature is a pressure switch, placed in the supply line, and set to give an alarm at 25 psig.

High pressure for the specimen loading is obtained from a compressor with a small accumulator. The maximum output is 5000 psig. The gas passes from the accumulator, through a high-pressure regulator, into a pressurizing header equipped with pressure gages. Valved pressure lines run from the header to another patch panel. A bourdon-tube, Computran pressure transducer monitors the pressure in each pressurizing line. As the specimen pressure lines also terminate in the patch panel, any pressurizing line may be selected to pressurize a particular specimen by connecting the two terminals with flexible tubing. After each specimen is pressurized to the desired stress, the specimen valve is closed and remains closed for the duration of the test. Signals from the Computran pressure transducers are fed to a multipoint indicator and recorded continuously. Rupture of a specimen is immediately detectable from the multipoint chart.

The design and construction of an in-reactor creep test presents many more problems than the previously described stress-rupture test. The most important problem is that of obtaining an extensometer with the desired sensitivity and at the same time being able to withstand the reactor environment.¹ Figure 16 is a schematic drawing of a creep testing system using a cantilever parabolic specimen. With this type of specimen a large ratio of displacement to strain is obtained as well as a constant bending moment in the specimen. The load in each specimen results from a differential pressure applied across a stainless steel bellows mounted in a lever arm frame. The differential pressure is obtained by evacuating the bellows while maintaining a constant can pressure. Variation in specimen loading is achieved by the selection of various sizes of bellows and the lever arm ratios.

The extensometer, affixed to the experiment can and located along the pullrod between the specimen and lever arm, is made from a small 100-ohm trimmer potentiometer. The frame is of ceramic with a resistance ribbon fritted on the surface. A metal sliding contact is attached to the pullrod.

¹W. W. Davis and C. R. Kennedy, Stress-Strain Measuring System for In-Reactor Use in High-Neutron Flux, ORNL-3747 (January 1965).

Fig. 16. Graphite Creep In-Reactor Extensometer for Use in an ORR Poolside Facility.

The extensometer is placed in a null-balance circuit as shown in Fig. 17. The potentiometer in the control room is balanced and read by an operator at designated intervals, and a creep curve is plotted from the data. Figure 18 shows a creep experiment under construction. It contains nine graphite specimens with stresses ranging from 500 to 5000 psi. Another graphite creep test is shown in Fig. 19. Rectangular heaters have been placed around each specimen so that the test may be conducted at a temperature of 700°C. With the addition of the heaters, the number of specimens was reduced to six.

Stress-rupture testing of Hastelloy N is illustrated in Fig. 20. The lever arm arrangement has been modified in order that a tensile specimen can be tested. Five of the specimens are stressed between 30,000 and 19,000 psi at 650°C. The other five specimens are to be used in postirradiation studies. Figure 21 is a closeup showing the extensometer just before being placed in position.

Two experimental facilities located in the reactor core are available for mechanical properties tests. Both of these facilities are used mainly to obtain specimens for postirradiation studies. Figure 22

Fig. 17. Nul-Balance Potentiometer Circuit.

Fig. 18. Graphite Creep Experiment. Original reduced 41%.

Fig. 19. High-Temperature Graphite Creep Experiment. Original reduced 40%.

Fig. 20. Poolside Stress-Rupture Experiment. Original reduced 40%.

Fig. 21. Extensometer and Loading System of Stress-Rupture Experiment. Original reduced 46.5%.

Fig. 22. In-Core Water-Cooled Experiment. Original reduced 44%.

shows an uninstrumented, water-cooled experiment. It is possible to irradiate 264 tensile specimens ($2\frac{7}{8}$ in. long $\times$ $\frac{1}{4}$ -in. OD) in an experiment of this type. An instrumented experiment to be placed in the core facility is shown in Fig. 23. Thirty-two specimens can be irradiated at temperatures from 150 to 900°C.

Another uninstrumented experiment can be seen in Fig. 24. This experiment is attached to the rear of a poolside can and is known as a piggy-back experiment. The 47 tensile specimens for postirradiation studies are exposed to the pool water.

Fig. 23. Instrumented In-Core Experiment. Original reduced 36.5%.

Fig. 24. Poolside Piggy-Back Experiment. Original reduced 36%.

CONCLUSION

The various experiments that have been described here illustrate the fact that planning and design are extremely important for a successful in-reactor experiment. Although miniaturization is necessary, it is still possible to test a large number of specimens in a relatively small facility and obtain important data not available from other types of tests.

ACKNOWLEDGMENTS

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TECHNIQUES FOR IRRADIATING HIGH TEMPERATURE MATERIALS IN A STEEP FLUX
GRADIENT*

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Abstract

The J-12 core position of the Engineering Test Reactor is used to irradiate high temperature materials under a variety of temperature and high neutron flux conditions. Specimen types include ceramic oxides, refractory metal carbides and nitrides, tensile test specimens, and coated fuel particles. A maximum fast neutron dose of 6×10^{21} neutrons/cm² can be achieved in one calendar year.

Present techniques have evolved from design and development based on operating experience and the need to accommodate new specimen materials and irradiation condition requirements. This paper describes several versatile multiple-capsule assemblies and the facility in which they are operated.

Introduction

The Reactor Chemistry Division of Oak Ridge National Laboratory (ORNL) participates in the High Temperature Materials Program sponsored by the USAEC (1). The behavior of high temperature materials which are of interest for use in advanced nuclear reactor or thermionic converter applications is being investigated through a series of irradiation experiments. The specimens are installed in single or multiple-capsule assemblies, which are designed to achieve the desired conditions of temperature, neutron dose, and environment. After irradiation in the Engineering Test Reactor (ETR) in Idaho, the assemblies are returned to ORNL for determining and evaluating the effects of irradiation on the specimens. This paper discusses the design of the capsule assemblies and the capabilities of the ETR facility in which they are operated.

The J-12 core position is used to irradiate high temperature materials under a variety of temperature and high neutron flux conditions. Neutron fluxes and gamma heating rates vary through a

factor of five and reach maximum levels of 3×10^{14} neutrons/cm² sec fast (> 1 Mev), 5×10^{14} neutrons/cm² sec thermal, and 15 watts/gram of graphite gamma heat generation. The maximum fast neutron dose that can be achieved in one calendar year is 6×10^{21} neutrons/cm².

In past experiments, extensive investigations have been completed and reported for beryllium oxide (2, 3). More recent investigations have centered on other oxides and carbides. Nitrides, coated fuel particles, and tensile test specimens are being irradiated currently, along with final work on the oxides and carbides. An effort has been made to increase the capacity and versatility of the facility by making use of operating experience gained in the preceding investigations.

Four independent assemblies, each approximately 1-1/4 in. in diameter and 36 in. long, can be irradiated at the same time. This results in an irradiation capacity of 200 to 250 cylindrical specimens nominally 1/2 in. diameter by 1/2 in. long. Materials which have been irradiated in this specimen form include the oxides of beryllium, aluminum, magnesium, and yttrium; the carbides of tungsten, tantalum, niobium, titanium, and zirconium; and the nitrides of tantalum, niobium, titanium, and zirconium. Other materials include tensile test specimens and coated fuel particles.

Uninstrumented low temperature assemblies are designed to operate as close to cooling water temperature as possible, resulting in maximum specimen temperatures of about 300°C. Instrumented high temperature assemblies, each containing several different specimen materials in independent capsules, utilize gamma heating in the specimens and the cladding to achieve controlled specimen temperatures up to 1100°C. The thickness of a gas annulus and the thermal conductivity of a helium-argon

mixture flowing through the annulus control the rate of heat flow from each capsule. Conduction across these precisely established annuli determines the relative temperatures of individual capsules, and thermocouple-actuated variation of the composition of the gas mixture determines the operating temperature of a control capsule. An automatic data transmission system relays operating temperature information from the ETR to ORNL.

Description of Irradiation Facility

Figure 1 shows the lead tube of a high temperature capsule assembly entering one of four 1-5/16 in. diameter holes in the J-12 core filler piece. The top piece of an uninstrumented low temperature capsule assembly is shown protruding from another hole and the remaining two holes are vacant. A total of four lead tube and/or uninstrumented capsule assemblies can be installed in any combination.

The lead tube rises vertically through a hanger clamp, bends, and runs horizontally through another hanger clamp into a junction box mounted on the flanged nozzle. Thermocouple leads and gas inlet lines are indicated by the dotted line passing through a hole in the nozzle trench floor and running to the instrument console. Off-gas lines are not shown.

The aluminum core filler piece accommodates the four capsule assemblies in holes nominally 1-5/16 in. in diameter with approximately 36 in. of useful irradiation length. Four small aluminum tubes, mounted vertically in slots at the center of each face, accommodate flux monitor wires. The wires are routinely withdrawn at the end of each cycle for scanning. The J-12 lattice position is surrounded by four fuel elements adjacent to the sides and by three control rods and an experiment facility on the corners.

ORNL DWG. 66-3163

Figure 1

Figure 2 illustrates the ranges of neutron flux and gamma heating considering variations between the four capsule assembly holes and variations caused by core loading changes and rod movements. Gamma heating values range from a minimum of 2 to 4 watts/gram to a maximum of 10 to 16 watts/gram. Thermal neutron flux values range from a minimum of 0.9 to 1.8×10^{14} neutrons/cm² sec to a maximum of 3 to 7×10^{14} neutrons/cm² sec. Fast neutron flux (> 1 Mev) values range from a minimum of 0.5 to 0.8×10^{14} neutrons/cm² to a maximum of 2.0 to 4.5×10^{14} neutrons/cm² sec.

Routine neutron flux measurements are made with Co-Al wire and with Ni wire, via the reactions $^{59}\text{Co}(n, \gamma)^{60}\text{Co}$ (thermal) and $^{58}\text{Ni}(n, p)^{58}\text{Co}$ (fast). Resonance corrections are determined from periodically measured cadmium ratios. In addition, perturbed neutron flux values are obtained from monitors placed inside each capsule assembly. Such materials as high nickel steel, stainless steel, pure iron, and $\text{Al}_2\text{O}_3\text{-Co}$ have been analyzed for the results of the reaction $^{54}\text{Fe}(n, p)^{54}\text{Mn}$, plus those listed above. A computer program is used to determine the flux values, based on the radiochemical data and detailed reactor operating history. Gamma heating rates in the reactor at low power levels are measured by the reactor operator with a graphite/ CO_2 ionization chamber. After extrapolation to full power, these values and data accumulated from other experiments are used to design new irradiation assemblies.

Gamma heating rates can be measured only at the beginning of a reactor cycle. The effect of rod movements, fuel burnup, and reactor power variations on gamma heating can be measured with stainless steel calorimeter capsules included as part of a normal instrumented capsule assembly. The relative temperatures of wall thermocouples in individual

Figure 2

capsules also gives some information regarding changes in the gamma heat distribution during operation. This operating temperature information is automatically recorded on a data logger tape and subsequently transmitted from the ETR to ORNL on a daily basis.

A current irradiation schedule for capsule assemblies is shown in Table 1 to illustrate the variety of specimens and temperatures, and the magnitude of the accumulated fast neutron doses.

Table 1. Summary of High Temperature Materials Irradiation Program

Experiment Number	Specimen Materials	Max. Irrad. Temp. (°C)	Fast Neutron Dose Range (nvt > 1 Mev x 10 ²¹)	Irradiation Status
41-23	Hot pressed carbides; oxides	400	2-7	Completed Jan. 1966
41-24	Hot pressed and slip cast carbides	400	1-4	Completed Dec. 1965
41-25	Explosion pressed carbides	400	1-4	In reactor
41-27	Hot pressed and slip cast carbides; oxides	1100	2-7	In reactor
41-26	Nitrides; explosion pressed carbides	400	2-7	In reactor
41-29	Nitrides; explosion pressed carbides	1100	1-4	Design stage
41-28	PyC coated fuel particles	1400	0.5 - 1.0	Assembled
41-30	Hastelloy N tensile test specimens	150	0.1 - 0.3	Completed Oct. 1965
41-31	Hastelloy N tensile test specimens	700	0.1 - 0.4	Completed Feb. 1965

Low Temperature Capsule Assemblies

Figure 3 is a pre-assembly photograph of the hardware used for low temperature irradiations of carbide and oxide specimens. Cylindrical stainless steel holders are bored and honed to accept the centerless

Figure 3

ground specimens, and machined externally for a close fit with the outer containment tube. The holders are then segmented in three sections to facilitate hot cell disassembly. The exterior of the capsule assembly is water-cooled in the same manner as the high temperature assemblies.

Nitride specimens are being irradiated in a similar capsule design utilizing a two-piece aluminum holder. Tensile test specimens have also been irradiated in a capsule design utilizing aluminum sleeves and holders.

High Temperature Capsule Assemblies

Figure 4 illustrates the temperature control system employed with high temperature capsule assemblies. Helium cover gas provides a non-reactive static environment for the specimens and internal capsule components. The constant pressure supply bleeds past the top of the capsule column to off-gas to prevent activated material from diffusing back to the console. The He/Ar control gas mixture is introduced at the base of the capsule column, and flows upward through the annulus between the surfaces of the capsules and the outer containment tube. The amount of Ar in the mixture is automatically regulated as necessary to keep a thermocouple in the wall of one of the capsules at a constant operating temperature.

Figure 5 is a pictorial section illustrating high temperature capsule assembly components used for carbide specimens (upper capsule) and oxide specimens (lower capsule). Specimens are centerless ground and precisely fitted into sleeves affording chemical compatibility and dimensional stability. This sleeve is then inserted into a standard stainless steel

SIMPLIFIED FLOW DIAGRAM FOR HIGH TEMPERATURE ASSEMBLY

Figure 4

HIGH TEMPERATURE CAPSULE ASSEMBLY COMPONENTS

Figure 5

capsule, which is seal welded to the adjoining capsule. A thin flange centers the capsule in the outer containment tube. The outer cylindrical surface adjacent to the specimens is machined to a diameter which will produce the proper gas annulus to achieve a desired specimen center line temperature.

In the case of the graphite sleeves used for carbide specimens, differential thermal expansion causes a significant gap at the graphite/steel interface at operating temperatures. This expansion gap requires provisions for keeping the graphite sleeves centered. A Nb-1% Zr alloy disc and pin were developed to meet the requirements for high temperature strength, compatibility with graphite, and thermal expansion similar to stainless steel. The disc is installed in a counterbore at the junction between capsules, and also serves as a support for the sleeve and specimens. A center line flux monitor material in the form of a thin washer may be placed over the centering pin on top of the disc.

Twelve axial holes running through the capsule walls accommodate thermocouples, gas lines, and flux monitor wire. Ten thermocouples, one control gas inlet tube, and one flux wire are normally used.

The complete high temperature capsule assembly installed in the core filler piece is shown in Figure 6. The gamma heat generated in the specimens and capsule components is removed by conduction through the control gas and transferred to the reactor cooling water which flows through the annular space between the outer containment tube and the core filler piece. Control gas inlet line, cover gas lines, and thermocouples are shown at the top of the assembly. A sintered stainless steel filter in the cover gas entrance tube is provided to allow an epoxy seal of the capsule column when the assembly is removed after irradiation.

A standard capsule assembly consists of twenty-six capsules. The Inconel "X" spring below the lower capsule allows for thermal expansion of the column and provides support at operating temperature. In addition to the geometry described above for carbide and oxide specimens, there are capsule designs to accommodate coated fuel particles and gamma heating calorimeters. Center line thermocouple instrumentation is possible for all types of specimens. Chromel-alumel is used for wall thermocouples, and tungsten-rhenium or chromel-alumel for center line applications.

A typical temperature profile for a capsule containing a carbide specimen is shown in Figure 7. Heat transfer calculations (4) are accomplished with a radial geometry digital computer code, which has required frequent revision to accommodate new geometries and materials. It is now being rewritten in a sufficiently general and versatile form to allow future use without major modification or supplemental hand calculations (5).

Summary

Features which make this irradiation facility particularly useful are the efficient utilization of space, the high fast neutron flux, and the demonstrated reliability and versatility. Since more than 50% of the volume of the core position may be utilized, a large number and variety of specimens may be irradiated simultaneously in the four independent capsule assemblies. Broad ranges of specimen temperature and neutron dose are attainable, up to about 6×10^{21} nvt fast (> 1 Mev) per year. The heat source, the absorption of reactor gamma radiation, is inherently dependable; it is not subject to complete failure (such as loss of electrical heat) or to large decreases with time (such as

HIGH TEMPERATURE CAPSULE ASSEMBLY-INSTALLATION

Figure 6

THERMAL PROFILE FOR HIGH TEMPERATURE CAPSULE — CARBIDE SPECIMEN

Figure 7

burnup of fissile material), which are frequent problems in long irradiations. The basic design is readily adaptable to different specimen materials, and the reliability of the temperature sensing and control system has been demonstrated by some six years of relatively trouble-free operation.

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IRRADIATION SYSTEM FOR CRYOGENIC EXPERIMENTS
IN THE PLUM BROOK REACTOR

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INTRODUCTION

Nuclear rockets will require many components to function while exposed to an environment of high-intensity nuclear radiation, cryogenic temperature, and hard vacuum. Shielding may be used to reduce the radiation problem, but shield weight must be optimized with payload and vehicle thrust. This optimization requires that shielding be minimized and components be qualified or developed for maximum tolerance to radiation.

Much work has been done on the effects of radiation on materials and components and a considerable amount is in progress dealing with combined environments. The Radiation Effects Information Center of Battelle Memorial Institute reports available results of radiation effects work. Reference 1 lists the latest compilation of Battelle reports. There is, however, a continuing need for research wherein materials and components are functionally tested in a combined environment of nuclear radiation, cryogenic temperature, and vacuum. To provide a test environment for this type of effort, three cryogenic systems exist at the NASA Plum Brook Reactor Facility (PBRF) in Sandusky, Ohio. One closed-loop system, which has been in use for several years, has a 1.1-kW capability and can deliver helium gas at 30° R and 50 psig to an experiment (ref. 2). The second system is a 20-kW closed-loop system capable of delivering helium gas at 50° R and 100 psig (ref. 3). The third system, the subject of this paper, is being installed in support of the Lewis Research Center's radiation effects program. This system is a once-through system and is also unique in that helium can be supplied to large experiments in the 11 $\frac{3}{4}$ -inch i.d. HT-2 test hole at 100° R and 1000 psig as compared with small experiments and low pressures for the other two systems.

The irradiation system discussed in this paper consists of an experiment capsule assembly for holding experiment packages, an insertion system for placing of the experiment capsule in the PBR, a data acquisition system for recording experiment data and controlling the experiment, and a cryogenic helium system for cooling and actuating experiments. In addition to the irradiation system, a brief description of the PBR is included.

REACTOR FACILITY

The PBR is a 60 MW test reactor with MTR-type fuel elements. It is light-water cooled and moderated with primary beryllium and secondary water reflectors. The reactor complex has been described briefly in a paper presented at the previous conference in this series (ref. 4) and in references 5 and 6.

Figure 1 is a cutaway perspective of the PBR tank assembly, showing the basic configuration and layout of test holes. A number of beam holes and rabbit tubes and two horizontal through holes are used for experiments. Figure 2 shows the orientation of beam holes and through holes around the PBR. The test system described in this paper utilizes horizontal through hole 2 (HT-2), which is located adjacent to the north side of the reactor and traverses the reactor tank from quadrant A to quadrant C. The equipment described herein utilizes the quadrant C end of HT-2.

HT-2 is an aluminum cylinder of $11\frac{3}{4}$ -inch inside diameter by about 14 feet long. Small guide rods ($1/4$ -inch aluminum) are installed along the inner surface to aid capsule alignment and to protect the tube from abrasize wear. The tube is cooled inside and outside by circulating reactor primary water.

The fast neutron flux and gamma heating rate along the axial centerline of HT-2 are shown in figure 3. The sketch on this figure indicates the tube's proximity to the fueled region. The curves are calculated for the tube in a voided condition. Fast neutrons of the order of 5×10^{12} n/(cm²)(sec) ($E > 1$ MeV) and a gamma heating rate of approximately 1.4 W/g are available, based on full reactor power. This gamma rate corresponds to approximately 4.6×10^8 rads/hr. Exposure rates on an experiment are variable by adjustment of the insertion depth into the tube.

IRRADIATION TESTING SYSTEM

The system being installed will provide for the irradiation of experiments functioning in a cryogenic-vacuum environment and for the pre- and post-irradiation handling of the experiments. Figure 4 is a plan sketch of the arrangement of equipment in the quadrant C area. Equipment shown includes the experiment capsule and the insertion and handling system. Except for the hot cave and parts of the service frame, the equipment is located and operated under 20 to 25 feet of water.

Experiment Capsule

The experiment capsule positions the experiment in the HT-2 test hole and provides the following additional capabilities:

- (1) Control of the cryogenic coolant and actuation gas flow to the experiment

- (2) Connection of the experiment to the data system
- (3) A vacuum environment of approximately 2 microns for the experiment
- (4) Indexing of the experiment in 60° increments around the axial centerline of HT-2; The indexing is performed remotely in a special hot cave.

Description. - Figure 5 is a cutaway sketch of the experiment capsule. The capsule is constructed from four basic subassemblies, which are the equipment box, the capsule snout, the experiment cap, and the porting plate bulkhead. The equipment box has outside dimensions of approximately 5 by 2 by 4 feet (lwh) and is constructed with two compartments. The front compartment (1) contains the surge chambers and the piping that directs gas to and from the porting plate bulkhead. A vacuum of approximately 130 microns is held in this compartment to reduce heat transfer from the capsule surroundings to the gas piping. The rear compartment (2) contains service equipment for the experiment such as control valves, vacuum pumps, pressure transducers, thermocouple reference junctions, and power supplies. This compartment is purged with clean dry air to remove heat generated by the vacuum pumps and other electrical components.

The capsule snout is constructed from a tube with an 11.25-inch outside diameter, a 10.25-inch inside diameter, and 9 feet 6 inches long. The 9-foot length of the snout allows the experiment package to be positioned close to the fueled region of the reactor. The 11.25-inch outside diameter is a basic limitation for HT-2 experiments. Eighteen vacuum-jacketed gas flow lines traverse the snout from compartment 1 of the equipment box and terminate at a bulkhead located at the front end of the snout. Six of the lines are 1.0-inch outside diameter and contain the electrical wiring for experiment command signals and data. Of the remaining twelve gas flow lines, six are 1/2-inch outside diameter and six are 1/4-inch outside diameter. In addition, one 3-inch vacuum line, two 1/4-inch seal vacuum lines, and one 5/8-inch water supply line are located in the snout. The lines are woven in the snout to reduce radiation streaming.

The experiment cap is 11.25 inches in outside diameter, 10.6 inches in inside diameter, and 40.0 inches long. This cap is constructed of Ti-6Al-4V alloy and weighs 69.5 pounds. It provides the primary containment between the HT-2 coolant water and the vacuum surrounding the experiment. The experiment package contained in the cap includes the experiment, the necessary support tubing, and a special flange that mates with the snout bulkhead. The experiment package is limited to 10 inches in outside diameter and a mass of about 220 pounds. The experiment cap is screwed onto the snout after connection of the experiment to the snout bulkhead. During irradiation, water must be circulated through HT-2 around the cap at a flow rate of at least 24.8 gpm to prevent nucleate boiling. Over 100 gpm is available with the present capsule geometry.

The porting plate bulkhead, while not a major subassembly, is described because of its unique use. The bulkhead is located on top of the equipment box and, when used with a porting plate, directs gas to the proper lines in the snout. When an experiment is indexed, the rotation of this porting plate redirects the gas flow as desired. In addition, the bulkhead provides a convenient area for mounting experiment pressure transducers that are not radiation tolerant.

Figure 6 is a photograph of the completed experiment capsule assembly. The overall assembly is 18 feet long and weighs approximately 4000 pounds.

Capsule flow. - Figure 7 indicates the flow path within the experiment capsule. Two warm and two cold gas streams penetrate the purged region of the equipment box and merge to form two independent pressure, temperature, and flow streams. The two flow streams are passed through final pressure-control valves and then penetrate the wall separating the purged compartment from the vacuum compartment. The two streams then pass through surge chambers, where the pressure and temperature feedback signals are obtained, and enter the porting plate bulkhead. From the porting plate bulkhead, the gas streams are routed through the porting plate, back down through the vacuum compartment, and through the gas lines in the capsule snout to the experiment. Gas exhaust follows a reverse flow path through separate tubes and is exhausted outside of the reactor building. Lines are also included in the capsule for snout coolant water (supply and exhaust), for purge air supply and exhaust for the equipment in the box, and for vacuum exhaust.

Insertion and Handling System

Reference is made to figure 4 for the location of the insertion handling system. The criteria for the design of this system were as follows:

- (1) Insertion of the experiment capsule snout into the HT-2 test hole, during reactor operation, against hydraulic forces and friction in excess of 15,000 pounds
- (2) Rate of reactivity change less than $0.02 \Delta K/(K)(\text{sec})$ because of experiment insertion
- (3) Separation of the coolant water in the HT-2 test hole from the water in the quadrant
- (4) Redundant interlocks and thrust-limiting devices to prevent damage to the HT-2 test hole or the experiment capsule
- (5) Operation and handling of the experiment capsule during reactor operation
- (6) Removal and replacement of the experiment without removing the experiment capsule from the quadrant.

Insertion table. - The insertion table (fig. 4) is approximately 17 feet long and is constructed of schedule-40 304 stainless steel pipe. A ball screw, center mounted and end fixed, drives a carriage along rails located on the top surface of the table. The experiment capsule is held to the table carriage by means of a rotating C clamp, which engages with a concentric ring on the capsule snout. The table frame and ball screw were tested against thrust loads in excess of 20,000 pounds after fabrication. Limit switches suitable for operation under water for periods in excess of 1 year are mounted in pairs along the side of the table in four places. Additional limit switches, which indicate four approximate capsule insertion depths in HT-2, are located on one side of the table. All limit switches are actuated by a cam arrangement attached to the underside of the table carriage. Located beside the insertion table is the drive motor which, through a gear reducer and chain arrangement, rotates the ball screw to provide capsule movement. The table inserts the capsule at 20 in./min. This rate may be changed to 30 in./min by changing the sprockets on the output of the drive unit.

Gate valve and seal assembly. - The gate valve and seal assembly are attached to the HT-2 test hole nozzle and provide a means of isolating the water in HT-2, which is at 150 psig, from the quadrant water. The seal assembly performs the isolation function during experiment insertion, and the gate valve is used when the capsule snout is not in HT-2.

The seal assembly consists of two independent sealing units. Triple chevron seals made of natural rubber provide a primary seal during capsule movement. Opposed, natural rubber U-cup seals, held in place by a spacer ring and water pressurized, provide a static or secondary seal when the capsule is not in motion. The seal assembly is separated from the gate valve by a bellows unit that provides for $\pm 0.5^\circ$ of misalignment and allows the seal assembly to float on the outside diameter of the capsule snout. Because of radiation damage, the seals used in this equipment require replacement, at 9-month intervals based on a 70% operating time for the reactor.

Capsule service frame. - The service frame is located directly behind and above the insertion table and is used when indexing of the porting plate or routine maintenance of the equipment in the experiment capsule is required. The frame provides a means for tilting the capsule snout down at angles of 0° to 40° from horizontal and for use of the reactor building crane to raise the equipment box portion of the capsule above water level. When the capsule equipment box is above water, the experiment is shielded with 6 feet of water.

Hot cave subsystem. - The hot cave subsystem consists of an insertion table located in quadrant C and a hot cave located outside the quadrant. The hot cave contains two 12-inch penetrations for capsule snout insertion and experiment transfer, a disassembly table and related tools, two master-slave manipulators, a viewing window, and an access door for maintenance of disassembly equipment. The cave is suitable to hold an experiment

equivalent to a point source of the order of 20 Kilocuries located in the center of the cave. This source will result in a radiation level of about 2.5 mr/hr at the outside surface of the cave wall.

Reactivity

The reactivity worth of a typical experiment was measured in the PBR mockup reactor. The results of this testing and the corresponding PBR limits are listed in table I.

TABLE I. - REACTIVITY EFFECTS

Reactor reactivity effect	PBR limit for one experiment	Typical HT-2 experiment
Maximum decrease	0.75% $\Delta K/K$	0.13% $\Delta K/K$
Maximum step increase by experiment failure	.15% $\Delta K/K$	.04% $\Delta K/K$
Maximum planned change rate	.02% $\Delta K/(K)(\text{sec})$	.0015% $\Delta K/(K)(\text{sec})$
Maximum planned change with reactor at power	.25% $\Delta K/K$	.13% $\Delta K/K$

Instrumentation

The instrumentation complex is located outside the containment vessel, in the experiment control room, approximately 100 feet from the experiment package. The complex was designed primarily for support of dynamic testing of control components. Equipment is available, however, for recording thermocouple or other related signals such as flow or pressure.

For dynamic data acquisition, the console contains 44 channels. The 44 channels consist of an 8-channel direct writing oscillograph, a 12-channel recording oscillograph, a 14-channel FM tape recorder, and a 10-channel strip chart recorder.

For data conditioning, the console includes a 15-amplifier analog computer, a transfer function analyzer, a low-frequency oscillator, a frequency changer, and other equipment such as amplifiers, power supplies, etc.

The experiment safety equipment consists of 48 channels for monitoring critical parameters. A total of 24 channels are reserved for experiment support equipment and 24 channels for thermocouples. Each channel

of the system provides both visual and audible alarms and can also be used to initiate an automatic withdrawal of the experiment if the particular parameter approaches an unsafe condition.

A control panel to operate devices in the experiment capsule and a control console, which will contain operating equipment required for an individual experiment, are included in the complex.

The entire complex is tied together through a master plugboard. Transition from one experiment to another requires programing the master plugboard, connecting the appropriate safety channels, and placing in the control console the operating equipment required for the particular experiment.

CRYOGENIC HELIUM SYSTEM

Some experiments planned for this irradiation system require helium gas in two separate flow streams called coolant gas and drive gas. Additional design criteria for this system include the following:

- (1) Each of the two gas streams is variable in temperature from about 100° to 600° R.
- (2) The coolant gas stream is variable in pressure from 600 to 1,000 psig and is capable of delivering 0.5 pound of helium gas per second.
- (3) The drive gas stream is variable in pressure from 150 to 250 psig and is capable of delivering 0.04 pound of helium gas per second.
- (4) The experiment exhaust back pressure is controllable to about 35 psig.

Flow and Control

Figure 9 is a simplified flow diagram of the cryogenic helium system. Helium gas flows from a high pressure storage bank through a main pressure control valve (PCV). The flow is split into two streams, which are called hot and cold. The hot line is routed through a heater where the temperature is controllable. The cold line is routed through a liquid-nitrogen bath-type heat exchanger and through a similar liquid-hydrogen heat exchanger.

The hot and cold lines are each split into two flow lines and routed through flow control valves called temperature control valves (TCV). One cold and one hot stream are blended and routed through a final pressure control valve to form the coolant supply for the experiment. The temperature of this coolant gas is sensed near the experiment and controlled by cycling the two appropriate temperature control valves. The other hot and cold lines are likewise blended and controlled to form a second gas supply for driving the experiment equipment.

As the coolant and drive gas exit from the experiment, they combine to form a common exhaust. Exhaust gas is routed through a final pressure control valve to maintain the desired back pressure on the experiment.

The helium exhaust gas may be treated in several ways. Clean gas may be routed directly to a vent stack or reheated and stored in a low-pressure bag. Gas of unknown activity may be reheated, passed through a radiation monitor and then either vented or stored. Stored helium will later be re-compressed, purified, and returned to the high-pressure storage bank.

Helium flow duration through this system is related to total flow rate and the final source pressure head available from the helium supply. As indicated in figure 10, an experiment that requires 0.1 pound of helium per second and a source pressure of 1000 psi can be operated continuously for 3 hours. Likewise, if a high flow rate such as 0.5 pound per second and a 1250 psi head are required, the experiment can be operated for about 1/2 hour. (At a flow rate of 0.5 pound per second, the cold helium gas will experience a pressure drop between the gas storage area and the reactor building of about 240 psi.)

Description

The physical layout of the cryogenic helium system is shown in figure 11. Major pieces of equipment are grouped in the functional areas that will now be discussed.

Gas and cryogen storage. - This area is about 500 feet north of the reactor and includes the following equipment:

- (1) The high-pressure helium storage bank has a capacity of 200,000 SCF of gas at 2200 psi; this is equivalent to about 2100 pounds of helium gas at 60° F.
- (2) A nitrogen gas cylinder bank stores purge gas and gas for valve operators; this bank stores about 39,000 SCF of nitrogen at 2200 psi.
- (3) A liquid-nitrogen bath-type heat exchanger consists of a 4000-gallon Dewar containing 3/4-inch stainless steel cooling coils equivalent to about 2000 lineal feet. This heat exchanger is capable of cooling helium gas at 0.5 pound per second from 520° R down to about 150° R.
- (4) A 4000-gallon liquid-hydrogen heat exchanger is similar to the nitrogen Dewar, with 3/4-inch stainless steel cooling coils equivalent to 1500 lineal feet. A 25-foot vent stack is provided to dissipate hydrogen boiloff gas above ground level. Under full flow conditions, hydrogen boiloff rate will be about 0.5 pound per second. This heat exchanger will be capable of cooling the 150° R gas further to below 100° R. Hazards associated with the storage and use of this liquid hydrogen are primarily responsible for the remote location of this area.

(5) A sand-filled blast barrier protects the reactor complex from line-of-sight blast damage in the event of a hydrogen explosion or deflagration.

(6) Control and miscellaneous valving, piping, transducers, and other equipment provide for the safety and functional control of this area.

(7) A 4-inch stainless steel helium vent stack, 61 feet high, discharges used or purge gas. The radioactivity of the effluent from this stack is limited to a continuous gaseous release of 4.6 microcuries per minute. Higher short-time release rates are permissible.

Piping trench. - A concrete walled trench extends from the gas storage area to the north basement wall of the reactor building. The necessary piping and conduit connecting the main functional areas are located in this trench.

Reactor building basement. - The helium gas lines run from the piping trench across the reactor building basement to an equipment area just outside the containment vessel. The heater for the hot gas stream is located here. Just prior to penetrating the containment vessel, each gas line passes through an emergency shutoff valve. These valves are remotely operable from the helium system control console and the reactor operation console and serve to preserve containment vessel integrity when necessary.

Containment vessel. - Located on the inside containment vessel wall is an assembly of valves, surge chambers, and associated equipment called the valve package. Principal components of this package are the temperature and back pressure control valves, a purge vacuum pump, and an air compressor that supplies operator air to all valves inside the containment vessel.

From the valve package, the helium lines are routed through the quadrant C wall and via flex lines to the experiment capsule. Final supply-pressure control valves and pressure and temperature transducers are located inside the capsule. Exhaust gas from the capsule is routed back along the same route as the supply gas.

Low-pressure storage and recompression area. - Helium exhaust gas is routed first to the low-pressure storage area. From here, it may be routed directly to the vent stack or stored and later recompressed and returned to the helium storage bank. The equipment for low-pressure storage and reprocessing is located about 200 feet from the reactor building adjacent to the piping trench. Major items of equipment are described as follows:

(1) The pebble bed heater consists of a steel cylinder, 6 feet in diameter and 14 feet high, which contains 20 tons of quartz pebbles. This heater is designed to warm helium gas from 150° to about 510° R (suitable for storage in the low-pressure bag) at a maximum flow rate of 0.5 pound per second.

(2) The hemispherical helium storage bag is 92 feet in diameter and 46 feet high and is capable of storing 200,000 SCF of helium at a gage pressure of about 2 inches of water. The bag is a double envelope type with the inner bag containing the helium and the outer bag pressurized by air blowers to maintain its shape and rigidity. The fabric material for the inner envelope is a laminate of hypalon-coated nylon bonded to a Mylar and aluminum foil laminate. The outer bag is a vinyl-coated nylon fabric. The ground cloth is of a material similar to the inner bag.

(3) A compressor and purification system are located in the compressor building adjacent to the storage bag. The compressor is a four-stage unit with a rated delivery of 71 SCFM at 2400 psi. The purification system consists of an oil and moisture adsorber bank and a liquid-nitrogen-cooled charcoal adsorber box for the removal of gaseous contaminants. This system is designed to recompress and purify helium gas at a rate greater than 100,000 SCF per day.

Control room. - All remotely operable equipment in the helium supply portion of the system is controlled from a central console located in the experiment control room. This console is separate from the experiment control console described previously and comprises three standard racks of equipment. The temperature and pressure of the coolant and drive gas streams, the exhaust pressure, the hot heat exchanger temperature, and associated normal flow valves and equipment are all controlled from this console.

All operations dealing with the low-pressure storage and recompression of the used gas are controlled from separate equipment located in the Compressor Building.

CONCLUDING REMARKS

An irradiation system has been described that is suitable for certain experiments planned for the Plum Brook Reactor in support of the nuclear rocket effort. This system consists of an experiment capsule, an insertion assembly for horizontal through hole 2 in the NASA Plum Brook Reactor, and a cryogenic helium system to service the test items. The testing system was designed primarily for the dynamic testing of components in a combined environment. This testing is a determination of how well a component follows a given command under various environmental conditions. The command signal may be either electric or pneumatic. Standard dynamic tests will include frequency response, threshold response, linearity, step response, and dynamic resolution. Both command and feedback signals are measured and recorded for evaluation of the component performance. Some of the basic capabilities of this system are restated as follows:

(1) A fast neutron flux of the order of $5.6 \times 10^{12} \text{ n}/(\text{cm}^2)(\text{sec})$ ($E > 1 \text{ MeV}$) and a gamma heating rate of the order of 1.4 W/g may be obtained.

(2) Helium gas flow to an experiment is controllable to about 0.5 pound per second, up to 1,000 psi, and at temperatures from 100° to 600° R. Run time is dependent on helium flow rate and the pressure required for the particular experiment.

(3) A vacuum environment may be provided for the experiment. The vacuum is estimated at approximately 2 microns without cryopumping.

(4) Experiments up to 10 inches in diameter and 31 inches long with a mass of approximately 220 pounds may be irradiated.

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Figure 1 Cutaway Perspective Drawing of Reactor Tank Assembly

CD-8467

Figure 2. - Plum Brook Reactor.

Figure 3. - Radiation along centerline of voided HT-2 for full reactor power.

Figure 4. - Capsule insertion and handling system.

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Figure 5 Experimental Capsule Assembly

Figure 6 Experiment Capsule

Figure 7. - Capsule flow schematic.

Figure 8 Instrumentation Complex

Figure 9. - Simplified flow schematic of cryogenic helium system.

Figure 10. - Flow duration of cryogenic helium system. Initial source, 2100 pounds helium at 2200 pounds per square inch.

Figure 11 Cryogenic Helium System

II. ANALYTICAL APPROACHES TO CAPSULE DESIGN

COMPUTER PROGRAM FOR DETERMINING SPECIMEN AND IRRADIATION
CAPSULE COMPONENT TEMPERATURE DISTRIBUTION*

by

Charles F. Reinke
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Before an experiment can be inserted into a test reactor, some form of hazards analysis is usually required. One section of this analysis consists of the specimen and capsule component temperature distributions under varying conditions of heat generation that may be encountered during the experiment. A computer program has been set up to handle these temperature calculations and thus eliminate the tedious and error prone hand calculations.

HEAT TRANSFER EQUATIONS

There are four basic heat transfer equations which the program solves to determine the appropriate Δt 's. The equations are given below.

$$\Delta t = \frac{Q}{AH} = \frac{Q}{2\pi RLH} \quad (1)$$

$$\Delta t = \frac{Q}{4\pi K} \quad (2)$$

*This work was performed under the auspices of the U.S. Atomic Energy Commission.

$$\Delta t = \frac{Q}{2\pi K} \left[\frac{1}{2} - \frac{(RI)^2}{(RO)^2 - (RI)^2} \ln \left(\frac{RO}{RI} \right) \right] \quad (3)$$

$$\Delta t = \frac{Q}{2\pi K} \ln \left(\frac{RO}{RI} \right) \quad (4)$$

where

$\Delta t = ^\circ\text{C}$

$Q =$ heat generation - cal/sec-cm

$K =$ thermal conductivity - cal/sec-cm- $^\circ\text{C}$

$H =$ contact resistance - cal/sec-cm²- $^\circ\text{C}$

$A =$ area of contact resistance - cm²/cm

$RI =$ inner radius

$RO =$ outer radius

$R =$ radius, cm

$L =$ length equal to 1 cm

Equation (1) gives the Δt due to a contact resistance or film drop.

Equation (2) gives the Δt across a cylindrical specimen in which there is uniform internal generation of heat with dissipation at the outer surface. Equation (3) gives the Δt across an annular section in which there is uniform internal generation of heat with dissipation only at the outer surface. Equation (4) gives the Δt due to steady state conduction in an annular section.

The specimen and irradiation capsule are divided into a series of concentric annular regions, see Figure 1, in which the Δt through each region can be divided into two components, one component being

$$\text{FUEL} = Q1 (\text{FISSION} + \text{GAMMA})$$

$$\text{ANNULUS} = Q1$$

$$\text{FUEL JACKET} = Q1 + Q2 (\text{GAMMA}) = Q3$$

$$\text{COOLANT} = Q3 + Q4 (\text{GAMMA}) = Q5$$

$$\text{CAPSULE} = Q5 + Q6 (\text{GAMMA}) = Q7$$

$$\text{FILM DROP} = Q7$$

SCHEMATIC LAYOUT OF IRRADIATION CAPSULE
SHOWING PARAMETERS USED IN COMPUTER CODE

FIG 1

due to steady state conduction and the other due to uniform internal heat generation. For instance, the Δt across the fuel jacket is given by

$$\Delta t = \overset{\text{Eq. (4)}}{\frac{Q_1}{2\pi K} \ln \left(\frac{R_O}{R_I} \right)} + \overset{\text{Eq. (3)}}{\frac{Q_2}{2\pi K} \left[\frac{1}{2} - \frac{(R_I)^2}{(R_O)^2 - (R_I)^2} \ln \left(\frac{R_O}{R_I} \right) \right]}$$

where Q_1 is the sum of the heat generation due to fission and gamma heat in the fuel and Q_2 is equal to the gamma heating in the jacket. This process is repeated for each annular region with appropriate values of Q , R_O , R_I , and K . Δt 's due to any contact resistances or film coefficients are obtained by applying equation (1). Equation (2) gives the Δt across a fuel specimen. The temperature at any point of interest is obtained by adding the appropriate Δt 's onto the bulk reactor coolant temperature which is known.

COMPUTER PROGRAM

The computer program, written in Fortran, contains three subroutines (H, COND, FUK) which can be called into operation in any order or combination necessary to solve the temperature distribution for a specific capsule layout. Subroutine H, which solves equation (1), gives the Δt due to a film drop or contact resistance. Subroutine COND, which solves equations (3) and (4), gives the Δt through an annular region. Subroutine FUK, which solves equation (2), gives the Δt across a cylindrical test specimen. The arguments or variables (Q , R , T_Y , T_X , H, COND, I) used in the subroutines are specified when a subroutine is called. Values for the radii, R , and heat generation, Q , are input

data to the computer. The initial value of TX, the bulk reactor coolant temperature, is also specified. The other values of TX are calculated by the computer. I is a number which specifies which contact resistance coefficient, H, or thermal conductivity expression, COND, to use. The computer selects the appropriate expression, assumes a Δt and calculates an average value for the thermal conductivity. It then uses this value to calculate the Δt . The assumed and calculated Δt 's are compared and if they differ by more than some specified amount, in this case 2°C, the computer adjusts its assumption and repeats the process until satisfactory agreement is reached.

THERMAL CONDUCTIVITY VALUES

Values for the thermal conductivity versus temperature for materials of interest were obtained from the open literature. A least squares fit was obtained over the temperature range covered by the data. The expressions obtained in this manner are the ones used by the computer in calculating average thermal conductivities. Table I lists the materials for which expressions are available and the appropriate I value to use in specifying the arguments when a subroutine is called. It should be emphasized that the list can be expanded or modified simply by adding additional expressions or modifying the existing ones.

Table I

<u>Subroutine</u>	<u>Material</u>	<u>I Designation</u>
COND	304 SS	1
COND	347 SS	2
COND	Na-78NK	3
COND	Tantalum	4
COND	Niobium	5
COND	Sodium	6
COND	Vanadium	7
COND	Hastelloy-X	8
COND	Vanadium-20Titanium	9
FUK	U-15Pu-15Zr	1
FUK	U-5Fs	2
H	MTR film coeff.	1
H	Na or NaK interface resistance	2

EXAMPLE

A simple example will help clarify the procedure involved in using the program. The capsule arrangement shown in Figure 1 will be used in the illustration. Capsule inner and outer surface, jacket inner and outer surface, and fuel surface and center temperatures are to be determined. The only film drops present are assumed to be at the capsule outer surface and jacket inner surface. Figure 2 is a listing of the program with the input data and call statements needed to define the temperatures shown in Figure 1. Figure 3 lists the call

statements and the corresponding Δt 's obtained. For instance, the temperature drop through the film is obtained by calling subroutine H where $TX = T1$, $Q = Q7$, $R = R5$, and H is determined by the value 1 assigned to I. $T2$ is calculated by the computer. The temperature drop through the capsule wall is obtained by calling subroutine COND where $TX = T2$, $QA = Q6$, $QB = Q5$, $R1 = R4$, $RO = R5$, and the thermal conductivity is determined by the value 1 assigned to I. The drops through the capsule coolant, fuel jacket, annulus surrounding the fuel, and specimen are obtained in similar fashion.

C RADIAL TEMPERATURE PROFILE - SPECIMEN AND IRRADIATION CAPSULE COMPONENTS

```

1 FORMAT(7F6.1)
2 FORMAT(5F6.3)
3 FORMAT(5X,A5,5X,A5,F10.1,I5////)
4 FORMAT(1X,A5,3X,A5,2X,7(F7.1,2X))
50FORMAT(18X,3H T1,6X,3H T2,6X,3H T3,6X,3H T4,6X,3H T5,6X,3H T6,6X,
13H T7)
5 FORMAT(5X,A5,5X,A5,F10.1,I5)
10 READ 1,Q1,Q2,Q3,Q4,Q5,Q6,Q7
PRINT 1,Q1,Q2,Q3,Q4,Q5,Q6,Q7
READ 2,R1,R2,R3,R4,R5
PRINT 2,R1,R2,R3,R4,R5
READ 6,CAP,SPEC,T1,K
PRINT 3,CAP,SPEC,T1,K
CALL H(T1,Q7,H,R5,1,TY)
T2=TY
CALL COND(T2,Q6,Q5,R4,R5,COND,1,TY)
T3=TY
CALL COND(T3,Q4,Q3,R3,R4,COND,3,TY)
T4=TY
CALL COND(T4,Q2,Q1,R2,R3,COND,1,TY)
T5=TY
CALL H(T5,Q1,H,R2,3,TY)
T6=TY
CALL FUK(T6,Q1,FUK,3,TY)
T7=TY
PRINT 5
PRINT 4,CAP,SPEC,T1,T2,T3,T4,T5,T6,T7
IF(K) 10,10,11
11 END

SUBROUTINE H(TX,Q,H,R,I,TY)
50 GO TO(51,52,53),I
51 H=1.030 MTR-H
GO TO 61
52 H=13.6 NAK
GO TO 61
53 H=.1017
GO TO 61
61 TY=TX+(Q/(6.283*R*H))
RETURN
END

SUBROUTINE COND(TX,QA,QB,RI,RO,COND,I,TY)
TY=TX+25.
TA=(TX+TY)/2.
75 GO TO(31,52,53,54,55,56,57,58,59),I
51 IF(TA-1400.) 1,2,2
1 COND=.03576+(.31E-04*TA) 304SS
GO TO 61
2 COND=.079
GO TO 61
52 IF(TA-1400.) 3,4,4
3 COND=.03483+(.3613E-04*TA) 347SS
GO TO 61
4 COND=.025
GO TO 61
53 IF(TA-784.) 5,6,6
50COND=.04460+(.1149E-03*TA)-(.2572E-06*(TA**2))+(.2432E-09*(TA**3)) NA-78K
X -(.3954E-13*(TA**4))
GO TO 61
6 COND=.060
GO TO 61
54 IF(TA-3000.) 7,8,3
7 COND=.1283+(.175E-04*TA) TA
GO TO 61
8 COND=.131
GO TO 61
55 IF(TA-1000.) 9,9,10
90COND=.1084+(.2934E-04*TA)+(.1086E-08*(TA**2))+(.1313E-11*(TA**3)) NB
X +(.1978E-14*(TA**4))
GO TO 61
10 COND=.1066+(.3431E-04*TA)
GO TO 61
56 IF(TA-883.) 11,12,12
110COND=.2321-(.2331E-03*TA)+(.2690E-06*(TA**2))-(.2108E-09*(TA**3)) NA
X +(.5112E-13*(TA**4))
GO TO 61
12 COND=.1220
GO TO 61

```

Figure 2. Fortran Program Listing

```

57 IF(TA-1900.) 13,14,14
13 COND=.0705+(.35E-04*TA)
GO TO 61
14 COND=.137
GO TO 61
53 IF(TA-1290.) 15,16,16
150 COND=.02241+(.3896E-04*TA)+( .1360E-07*(TA**2))-(.1100E-10*(TA**3))HAST-X
X +(.7584E-14*(TA**4))
GO TO 61
16 COND=.092
GO TO 61
59 IF(TA-1750.) 17,18,18
170 COND=.02653+(.1172E-03*TA)-( .1281E-06*(TA**2))+(.7664E-10*(TA**3))V-20TI
X -(.1655E-13*(TA**4))
GO TO 61
18 COND=.095
GO TO 61
61 C=(RI**2/(RO**2-RI**2))*LOGF(RO/RI)
A=(QA/(6.283*COND))*(.5-C)
B=(QB*LOGF(RO/RI))/(6.283*COND)
TY=TX+A+B
TA1=(TY+TX)/2.
IF(ABSF(TA1-TA)-2.) 71,71,72
72 IF(TA1-TA) 73,71,74
73 TA=TA-(ABSF(TA1-TA)/2.)
GO TO 75
74 TA=TA+(ABSF(TA1-TA)/2.)
GO TO 75
71 RETURN
END

SUBROUTINE FUK(TX,Q,FUK,I,TY)
TY=TX+150.
TA=(TX+TY)/2.
75 GO TO(51,52,53),I
51 IF(TA-1150.) 1,1,2
10 FUK=.19596E-01+(.55928E-04*TA)+( .11903E-06*(TA**2))-(.19932E-09* 15PU15Z
X (TA**3))+(.77177E-13*(TA**4))
GO TO 61
2 FUK=.07310
GO TO 61
52 IF(TA-1000.) 3,3,4
3 FUK=.0435+(.30483E-04*TA)+( .58673E-07*(TA**2))
GO TO 61
4 FUK=.1326
GO TO 61
53 FUK=.1600
GO TO 61
61 TY=TX+(Q/(12.566*FUK))
TA1=(TY+TX)/2.
IF(ABSF(TA1-TA)-2.) 71,71,72
72 IF(TA1-TA) 73,71,74
73 TA=TA-(ABSF(TA1-TA)/2.)
GO TO 75
74 TA=TA+(ABSF(TA1-TA)/2.)
GO TO 75
71 RETURN
END

```

VA

U-5FS

Figure 2. (Contd.)

<u>ΔT Through</u>	<u>Call Statement</u>
Film	Call H(T1, Q7, H, R5, 1, TY) T2 = TY
Capsule	Call COND (T2, Q6, Q5, R4, R5, COND, 1, TY) T3 = TY
Coolant	Call COND (T3, Q4, Q3, R3, R4, COND, 3, TY) T4 = TY
Fuel Jacket	Call COND (T4, Q2, Q1, R2, R3, COND, 1, TY) T5 = TY
Annulus	Call H(T5, Q1, H, R2, 3, TY) T6 = TY
Specimen	Call FUK (T6, Q1, FUK, 3, TY) T7 = TY

Figure 3. Call Statements Necessary to Solve for ΔT 's Shown.

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IRRADIATION CAPSULE TEMPERATURE CONTROL*

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IRRADIATION CAPSULE TEMPERATURE CONTROL

One important phase of irradiation capsule design is that of attainment and control of temperature. I would like to present a method of looking at the problem that I have found helpful. Many methods have been employed for temperature control; however, two of the most common are the use of a radial gas gap and or supplemental electrical heat. Both methods have their limits and it is the limits and methods of extending some of the limits that I would like to discuss.

Figure 1 is an idealized plot showing the range of usefulness for two methods of control: gas gaps and electrical heat. These limits are not exact, will vary from design to design, and give only an approximate relative picture for electrical heat vs gas gaps. Even for a given design it may be hard to draw an exact curve; the limits will depend in some degree on the optimism of the designer in predicting performance. The abscissa is the test specimen heat generation rate per unit length. One could use heat flux per unit area of the specimen but the curves would be more sensitive to a given capsule configuration. The ordinate is the surface temperature of the test element assuming water as the coolant of the outer capsule wall.

For any capsule configuration there is a minimum specimen temperature for any given specimen heat rate. This results from the thermal resistance of the materials of the capsule's construction. This limit is represented by the lower line which puts a floor on the use range of any means of temperature control.

The electrical heater is most useful in the range of low nuclear power and moderate temperature and may be used in conjunction with gas gaps. It has an upper temperature limit which depends on the heater design and materials. Heaters have been developed to exceed 1200°C for special cases. However, high performance heater development can be costly and if you are not careful the irradiation program degenerates into a series of electrical heater performance tests. The electrical heater has a practical upper limit for temperature control based on the heat input per unit length that can be crowded into a capsule. Unless the heater is equal to some appreciable fraction, perhaps 25% of the nuclear heat, its contribution to temperature

control is not worth its complication. Since most heaters have a higher heat capability at lower temperatures, I have shown the temperature limit for electric heat as decreasing with increasing power.

The second set of curves applies to gas gaps. The gaps may employ a fixed gas composition, usually He, Neon or argon, or may use an adjustable mixture of two of these gases. The former requires that the designer be more sure of performance or have the option of adjusting power. The latter requires more complicated controls. However, by varying the ratio of two gases with different thermal conductivities in the gap one can change the thermal resistance across the gap and thereby provide a means of adjusting temperature during capsule operation.

At low heat fluxes the upper gas gap boundary is limited by how large a gas gap one can use. If the heat flux is too low it may be necessary to supplement it with electric or other heat sources to drive the effective operating point into the region of gas gap control. As the temperature increases the upper limit levels off as radiant heat transfer across the gap becomes predominant. Once in this region changing the gap or the gas composition has little effect on the temperature required to drive the heat across the gap.

The lower gas gap limit is somewhat above the limit imposed by the thermal resistance of the capsule structure. There is some finite gas gap below which it is not practical to go. The difference between the two lines represents the resistance of this minimum gap. Even if the gap were reduced to a shrink fit some resistance would be present, but this could hardly be called a temperature control gap. The lower limit for gas gaps will tend to approach the line for the structure at very high temperatures because here radiant heat transfer again predominates.

I have plotted some representative points on the curve. These, with the exception of point "A" where we only investigated the problem, are capsules we have run. Note these are only representative curves and may vary considerably for different capsule configurations. I do not consider it impossible to build capsule A, however it would require different designs than we have employed.

The region to the left and above the upper gas gap curve can be handled by devices such as thermal radiation baffles which will raise the limit or structural thermal resistance which in effect drive the design point down into the region of gas gap control.

The 10 kw/ft 820°C capsule was designed to operate well within the range of gas gap control. It consisted of a fuel pin in a NaK bath, a stainless steel primary wall, a gas gap and then an outer water-cooled wall. The 20 kw/ft 760°C case is on the edge of the lower gas gap control line. The design of this capsule was similar to the 10 kw/ft case. Safeguards dictated double containment; hence, we were forced to use two walls. Initially these were a shrink fit with no gas gap and we attained the necessary overall thermal resistance by adding a Zircaloy sleeve of the required thickness in the NaK annulus.

The 3.2 kw/ft 1500°C capsule was a materials compatibility test achieving its temperature from γ heat only. It operated to the left and above the gas gap control curve. Two 3-mil thick molybdenum foil reflector baffles were inserted in a large gas gap to shift the curve into a region of control. Without the baffles the maximum temperature we could achieve with the same gap was 1170°C.

Table 1 shows some of the requirements of the capsule for point A, which was to the right and below the lower limit from the thermal resistance of the capsule structure, that is high heat flux and moderate temperature. In order not to exceed a given heat flux to the water the capsule

Table 1. Proposed Capsule - Thermal Design Evaluation

Requirements	
Specimen surface temperature, °C	650
Heat rate, kw/ft	30
Design - stagnant pool cooling	
Allowable heat flux to water, Btu/hr.°F.ft ²	190,000
Minimum capsule OD, in.	2.2
Required minimum mean capsule structure conductivity, Btu/hr.°F.ft	35
Design - forced convection cooling	
Capsule OD, in.	1.5
Required minimum mean capsule structure conductivity, Btu/hr.°F.ft	26

wetted surface area must not be less than some minimum. Without fins this fixes the minimum diameter and hence minimum distance from the fuel specimen to the heat sink. The requirements were 30 kw/ft at 650°C. We assumed simple cylindrical geometry, and a maximum heat flux to the water which was a stagnant pool of 190,000 Btu/hr·ft². This gave an overall diameter of 2.2 in. with a required mean conductivity of all the intervening material of 35 Btu/hr·°F·ft. As an alternate we considered forced convection with an outside diameter of 1.5 in. In this case the required mean thermal conductivity of the structure was 26 Btu/hr·°F·ft. We have not carried the design further; however, the double wall stainless steel capsule will not work. It is necessary to lower the line representing the limit imposed by structural thermal resistance. The outer wall could be thick and be made of aluminum

Where practical, as in poolside capsules, a carefully designed, fixed thermal resistance with final temperature control attained by adjusting the capsule position relative to the core is probably best for capsules at or below the gas gap control line. If one must have on stream temperature adjustment for such capsules without changing the flux, some other technique than gas gaps or electric heaters must be resorted to. It may be possible initially to have a sodium annulus, thermal conductivity about 38 Btu/hr·°F·ft at 900°F, to which small amounts of potassium are added at operating power to increase the overall thermal resistance until design temperature is reached. This would by no means be a simple technique.

In summary, electric heaters are most useful for low power moderate temperature capsules. Gas gaps do not function well at low heat rates or at very high temperatures. The use of supplemental heat with gas gaps will alleviate the former difficulty, and devices such as radiation baffles or structural thermal resistance as pyrolytic carbon will help in the latter case. The low to medium temperature, high power case may be the most difficult to accommodate and appears to be best handled by designing for a fixed structural thermal resistance, and if possible obtain temperature control by adjusting the nuclear heat.

ORNL-DWG 66-3680

Limits of Capsule Temperature Control vs Heat Generation.

ASSESSMENT AND CORRELATION OF ERRORS IN PRE- AND POSTIRRADIATION MEASUREMENTS

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In general, the basic goal of most irradiation testing programs is to determine the effects of a particular radiation, temperature, and atmospheric environment on the physical properties of potential reactor materials and components. In support of this goal, the experimental objective is to evaluate the material or component stability for reactor service conditions, where this evaluation seeks the correlation of physical property changes over a range of the environmental parameters (radiation, temperature, burnup, pressure, etc.).

The determination of the material's physical properties changes requires careful measurements, before and after irradiation. The environmental parameters must also be carefully measured during the irradiation. Inherent in these measurements are errors which propagate into the correlations; and, if not checked or minimized, they can lead to apparently inconclusive results. A detailed study of the interaction of small errors in the measurement systems that provide data for correlations has been conducted at Atomics International. The following will be a general discussion of the techniques and procedures developed for measurement system analysis and for assessment of measurement accuracy requirements, as well as some of the equipment which resulted from that study.

Engineers are generally cognizant of the arithmetic rules of error combination. The classical analytical approaches are widely described in the literature. At Atomics International, the general analytic tools and procedures have been specifically applied to assist the engineer in determining the steps required for minimizing error in a particular measurement system. Too often, engineers faced with setting accuracy requirements on a set of measurements will do so strictly from tradition, without regard to the effect the error induced by these accuracies will have on the final data, whether this accuracy is necessary and feasible, or even if this accuracy is later maintained.

Work performed under AEC Contract AT(11-1)-GEN-8

The procedure for setting accuracy requirements at Atomics International is effectively a reverse process, starting with the answer and working backward, and is illustrated in Figure 1:

- 1) The desired accuracy for a particular correlation is defined.
- 2) The measurement system or systems that must be employed to generate data required for correlation are thoroughly analyzed for sources of error.
- 3) Expressions for the interaction of sources of error in these systems and in propagation to a maximum error in the correlation are derived by standard techniques.
- 4) The required accuracy for each measurement is then back-calculated from the expressions derived and the quoted desired accuracy.

The sequel to this procedure is to examine the measurement systems to determine whether they are capable of the required accuracies. If these systems are not adequate, either new ones must be developed, or (if the requirements are beyond the existing state of the art for that measurement) the assigned accuracy for the correlation must be relaxed. The end result, in either event, is that the engineer has a feel for the quality of the data to be gained from the experiment. He also has a knowledge of the significance of the data, enabling him to differentiate between fluctuations caused by inherent error in the measurements and real changes in the material property as an experimental result.

The irradiation testing program at Atomics International has been highly influenced by the requirements of the SNAP reactor systems. These compact systems use a fuel material which exhibits high stability during irradiation. Any measured effects, therefore, are generally very small changes. To maintain a sufficiently high accuracy in the measured changes, a very high degree of accuracy is required in the individual pre- and postirradiation measurements.

For example, correlation of dimensional stability with temperature and burnup is very important in evaluating the capabilities of a reactor fuel system. Two methods are generally used for measurements of dimensional stability. The simplest is by comparison of the fuel specimen's physical dimensions (length, diameter, width, thickness, etc.), before and after irradiation. The other is by comparison of the fuel's bulk density, before and after irradiation. A typical fuel rod might be of rod form, $\sim 1/2$ to 1 in. in diameter. Measured dimensional changes might well be in the mil (0.001 in.) range, and the density changes would then probably be in the tens of milligrams per cubic centimeter range. In order to achieve a sufficient accuracy in the calculation of these changes, the measurements under comparison must be an order of magnitude more accurate. This required accuracy calls for some fairly sophisticated measuring devices and systems.

To illustrate the methods utilized to set these "order of magnitude" accuracy requirements, consider an analysis of the density measurement systems. The expected tens of milligram per cubic centimeter changes represent a fuel volume increase of $\sim 3\%$, which we want accurate to 10 or 15% (engineering accuracy). This assigns the required accuracy of the comparison or correlation. Archimedes' principle of liquid displacement is used for the density determinations. Four separate measurements are required for each determination, and these measurements are combined with the parameters of the system, as shown in Figure 2. The maximum error for each determination in this system is a weighted combination of the error in each of these factors, as shown in Figure 3. Each of the partial differentials are arithmetic combinations of known specimen and system characteristics. The total differentials represent the error in each measurement (mass in air, mass in liquid, and densities of the air and the liquid). The errors in the liquid and air densities are known (as characteristics of the system), and the error in the mass-in-liquid measurement can be set equal to twice that of the mass-in-air (since it requires two measurements to determine). Since the assigned error in the density is known, the required accuracy for the mass measurements can now be calculated. If these accuracy requirements are too stringent, either the acceptable error in the correlation must be increased or the error in the liquid and air densities must be reduced. The systems used for pre- and postirradiation density

determinations at Atomics International are designed to accomplish the latter by environmental control.

The pre-irradiation density measurement system follows practices for these measurements established by the National Bureau of Standards. All measurements are taken in an isolated, reasonably constant-temperature room (air density known to 0.0001 g/cc); the in-liquid measurements are taken in constant-temperature, high-purity distilled water (density known to 0.001 g/cc); temperature monitoring is by mercury-filled glass thermometers (to 0.1°C); and weights are recorded from a standard 1000-g Mettler balance (to 0.0003 g). The accuracy of this system has been demonstrated to be at least 0.1%.

The procedures required for the postirradiation measurements in a hot laboratory are more sophisticated, because of the enforced remote nature of the system and the hostile environment of the cells. Figure 4 is a schematic of the system developed at the Atomics International Hot Laboratory (AIHL) for density measurements of nonpyrophoric materials. The basic instrument is an Ainsworth SCI, 400-g capacity, remote-readout balance with weigh below (accuracy, 0.0003 g). The weighing liquid is monobromobenzene, selected for its high density, stability, and compatibility with the fuel. The liquid bath is maintained at a constant temperature (generally, the temperature of the cell atmosphere) by insulation and recirculating constant temperature oil. The liquid temperature is monitored by five thermistors, placed at different radial and axial locations within the liquid (0.1°C); these also check on temperature stratification and thermal convection. Complete wetting of the specimen surface and liquid degassing is accomplished through the use of an ultrasonic piezoelectric crystal attached to the weighing bath, energized only during raising of the bath. Constant liquid level between successive measurements is insured by a pneumatic jack, controlled by level switches which seat the bath with the bottom of the balance support pedestal; this also eliminates hindrance by convection currents within the cell during the in-liquid weighings. The entire weighing system is enclosed within a "wind shield," which also eliminates the hindrance of in-cell air convection.

The mechanics of this system are designed to eliminate the obvious environmental errors, and the measurement techniques are designed to

eliminate systematic error. A restriction in the in-liquid weighing sling requires that its weight be, at the absolute maximum, one-tenth that of the smallest specimen, preferably one-hundredth. An NBS-calibrated, Type 304 stainless steel standard, of a mass comparable to that of the specimen, is used to determine the liquid density between each specimen measurement (accuracy to 0.001 g/cc). The density of each specimen is measured at least three times between these standard measurements to give an indication of reproducibility, and the liquid density is adjusted for any temperature change between measurements.

The quoted accuracy of this system is a 0.1% uncertainty in a given density at a 95% confidence level. This is adequate for the fuels now under investigation; but other techniques and equipment are being studied; because, as more stable fuels are developed, smaller changes will be measured, requiring higher accuracy. Use of mercury as the weighing liquid is being considered; taking both dry and wet weights with the sling constantly immersed (to reduce the sling subtraction error) is being explored; and more accurate balances with finer remote readout systems are being sought. Computational procedures for compensation of temperature differences between the specimens and liquid due to decay heat have also been studied; but, at the present time, these effects have negligible bearing.

The density system is illustrative of combining generally accepted techniques into a well coordinated measurements system. The system developed for measuring the diameters of the SNAP rods is illustrative of utilizing a fairly unique approach to the complex problem of standardized pre- and postirradiation dimensional measurements to a 0.0001-in. accuracy.

A device, called a profilometer (not the surface roughness measuring instrument), has been used with excellent results for dimensional measurements. Figure 5 is a photograph of the one used for measurement of the pre-irradiation fuel rod and element diameters. The main body of the instrument is simply a machine lathe, and is used for positioning and rotating the specimen. The carriage of the lathe is fitted with two opposed inductive linear voltage displacement transducers. The transducers are spaced to just touch the outer surface and to lie on a diameter of the specimen. The lathe carriage then moves along

the lathe bed, and the voltage outputs from the transducers are recorded as they monitor any variations in diameter along the length of the specimen. A profile (hence the name) of the specimen is thereby recorded. Diameters of the rod are computed from this profile by comparison with the outputs of the transducers when in contact with a standard of known diameter and over steps of known heights. For convenience and continuity, the standard diameter and steps are usually a part of the holding fixtures at the ends of the specimen. Both measurements are then made at one time, and the transducers and recording systems are calibrated for each specimen. This particular system, if carefully used in a controlled environment, has a rated precision of 50 millionths of an inch. The quoted accuracy for "production" measurements, however, is 1/10 mil (0.0001 in.). This is adequate, and is comparable to the accuracy available in postirradiation measurements.

Figure 6 shows (out of cell) the postirradiation diameter measurements system used at the Atomics International Hot Laboratory. The electronics of this system are identical to those of the system used for the pre-irradiation measurements. This eliminates any source of error due to differences in the two systems. The mechanical portion of the system is designed for easy specimen handling in cell, and for accommodation of a wide variety of element or specimen sizes. This instrument also has a quoted accuracy of 1/10 mil, even in the relatively hostile in-cell environment.

There are a number of features about the mechanics and electronics of these systems that contribute to the maintenance of their high accuracy, but description of them is beyond the scope of this paper.

These systems have effectively eliminated the source of measurement error in point-by-point dimensional change comparisons. However, the error associated in interpretation of these data comparisons has always been a problem, because the sampling points may or may not be representative of the total rod configuration. For this reason, these systems are also equipped to automatically record data on IBM punch paper tape. With this capability, a large number of diameters can be measured which completely blanket the rod; the data pre- and post-irradiation can then be fed into a computer program which reduces the data and performs statistical comparisons, giving average

and peak rod change, ovalities at any desired axial locations, and profile scans at a number of circumferential orientations. Such a computer program is now in the development stage, and the pre-irradiation data for some experiments currently in operation have been taken accordingly.

These are only two of the measurement systems that are used at Atomics International for the characterization of experimental fuels and materials. The development of these systems are good examples of the general procedures and care used in development of all the required systems, including those used for environmental control and recording during irradiation. Use of this procedure, as well as others developed for statistical experiment design and complete project experimental integration, lays a good foundation to justify the sophisticated means utilized for the construction and the in-pile control of these experiments.

1. DETERMINE THE DESIRED ACCURACY OF THE REQUIRED CORRELATION
2. ANALYZE FOR ALL SOURCES OF ERROR THE MEASUREMENT SYSTEMS EMPLOYED TO GENERATE THE DATA FOR THE CORRELATION
3. DERIVE EXPRESSIONS FOR THE INTERACTION OF ALL SOURCES OF ERROR IN THESE SYSTEMS AND IN PROPAGATION TO A MAXIMUM ERROR IN THE CORRELATION
4. BACK-CALCULATE THE REQUIRED ACCURACY FOR EACH MEASUREMENT FROM THE EXPRESSIONS DERIVED (3) AND THE QUOTED DESIRED ACCURACY (1) FOR THE CORRELATION

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Figure 1. Procedure for Setting Measurement Accuracy Requirements

$$\rho_{\text{SPECIMEN}} = \frac{M_a \rho_l - M_l \rho_a}{M_a - M_l}$$

M_a = MASS OF THE SPECIMEN IN AIR

M_l = MASS OF THE SPECIMEN IN LIQUID

ρ_l = DENSITY OF THE LIQUID

ρ_a = DENSITY OF THE AIR

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Figure 2. Formula for Density Determinations

$$d\rho_s = \frac{\partial \rho_s}{\partial m_a} dm_a + \frac{\partial \rho_s}{\partial m_l} dm_l + \frac{\partial \rho_s}{\partial \rho_l} d\rho_l + \frac{\partial \rho_s}{\partial \rho_a} d\rho_a$$

$$\rho_s (\text{quoted}) = \overline{\rho_s} (\text{measured}) \pm d\rho_s$$

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Figure 3. Maximum Error for a Density Determination

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Figure 4. Atomics International Hot Laboratory System for Density Measurements

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Figure 5. Profilometer Used for Pre-irradiation Diameter Measurements

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Figure 6. Profilometer Used for Post-irradiation Diameter Measurements

DIGITAL COMPUTER TECHNIQUES FOR THE HEAT TRANSFER ANALYSIS OF IRRADIATION ASSEMBLIES

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Atomics International

Heat transfer studies of irradiation assemblies can be arbitrarily classified in two groups: design studies and analytical studies. Design studies involve the establishment of an irradiation assembly configuration, such that a given specimen, under given heat generation conditions, would operate at a specified temperature. The heat generation conditions could be given explicitly, or they could be given implicitly in the form of a specified neutron flux or burnup rate. When the configuration is given, then the analytical studies are made to determine in detail the temperatures, temperature gradients, and heat fluxes in the assembly. This paper will describe calculational techniques which are currently being employed at Atomics International for solving both design and analytical heat transfer problems.

Perhaps the most frequently encountered geometrical configuration for irradiation assemblies consists of a system of concentric right circular cylinders, with internal heating, cooled at the outside surface. Frequently, the heat flow is predominately one dimensional in the radial direction. Such cases may be treated simply; and, in general, a sophisticated approach is not justified, due to the lack of precision in the reported values for the thermo-physical properties of engineering materials. However, when large temperature gradients exist and, in particular, when gas bonding is employed, the temperature variation of materials properties must be taken into account.

The materials properties of interest are thermal conductivity and thermal expansion. For most materials, the temperature variation of these properties can be adequately expressed as a truncated power series of the form, $K = A + BT + CT^2 + DT^3$. Figure 1 illustrates a general cylindrical heat transfer region, one of several such regions in a typical irradiation assembly. The region may have both heat flowing through it and heat generated within it, and can be characterized by an average temperature and an average radius. The coefficient of thermal expansion (α) and the thermal conductivity (K)

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are determined for the average temperature in the region. Thermal expansion is computed at the average radius. The temperature difference across the region can then be calculated, using Equation 1 (Figure 1). If the temperature at one surface is known, the problem can be solved iteratively, using the equations in Figure 1. Initially, the average temperature is assumed to be the same as the known temperature. The solution is obtained when the unknown temperature, calculated in two successive iterations, does not change significantly [i. e., $|T_i - T_{(i-1)}| \leq \epsilon$,] where ϵ denotes a convergence criterion which can be made as small as desired.

This example has assumed that the region consists of a solid material. When a fluid region is encountered (for example, a liquid-metal or gas bond), it must be bounded by two solid regions. In this case, two regions must be treated simultaneously. For solution, the temperature and hot radius of one surface of the fluid region must be known. The radius of the other surface is determined by the thermal expansion of the adjoining solid region. Starting with either the centerline or surface temperature of an irradiation assembly, the temperatures and hot dimensions of adjoining regions can be determined in succession, until the entire assembly has been analyzed.

In binary gas controlled irradiation experiments, both the centerline and coolant temperatures are specified; and the gas control annulus size is to be determined. Figure 2 presents a method for sizing the gas annulus. It is assumed that all the temperatures and hot dimensions interior to the gas annulus region have been determined previously, including the temperature and size of the inside radius of the gas annulus itself. The outer radius of the gas annulus (R_2) will be sized to meet the required conditions. Equations 1a and 2 (Figure 2) give the temperature difference across the outer capsule wall and the gas annulus respectively. These equations cannot be solved directly. Equation 3 is obtained from Equations 1a and 2 by eliminating the logarithm of the unknown radius (R_2). Equation 2 can be rearranged to obtain Equation 2a, which can be solved iteratively together with Equation 3. This calculation will converge if an initial value for R_2 is chosen which approximates the solution. The calculation will not converge if, on any iteration, an inner radius becomes larger than an outer radius. Heat transfer to the coolant is considered in Equation 4.

These equations have been programmed for solution, using the IBM 7094 computing system. The computer code, GASP (Gas Annulus Sizing Program), was written in Fortran II and has been converted to Fortran IV computer language. GASP will treat irradiation assemblies containing up to 30 regions, and will use temperature-dependent materials properties for up to 50 materials. The code was designed to perform multiple calculations for parametric studies.

In addition to sizing the gas annulus and computing temperatures and hot dimensions, the code will determine the range of power levels over which the design temperature can be maintained by changing the gas mixture in the control annulus. Figure 3 illustrates a typical design study. The gas annulus size has been calculated for a range of gas mixtures. The maximum and minimum curves define the range of allowable power levels for the corresponding gas annulus size. Since there is an equal probability for the experiment power level to be greater or less than the desired condition, the design point should provide equal overpower and underpower compensating ability.

GASP was developed primarily for the analysis of SNAP irradiation experiments. Typically, these experiments operate with the gas annulus at temperatures in the neighborhood of 1000°F, where radiative heat transfer is insignificant. However, if a high-temperature experiment is to be analyzed (for example, a UO_2 test), the effects of radiative heat transfer must be considered. Another danger to be avoided is the use of a materials property function outside its applicable temperature range. For example, the thermal conductivity function for NaK will yield a negative value at temperatures about 2500°F.

A detailed analytical study of heat transfer of an irradiation assembly requires the use of a multidimensional computer code. These codes use an iterative method to obtain a solution for the temperature distribution in a heat transfer model which consists of a lattice of mesh points. Computer time, for these codes, is proportional to the number of mesh points and the number of iterations needed for convergence. Whenever the thermal resistance between one node and two adjacent nodes differs by more than an order of magnitude, convergence will tend to be slow. It is therefore desirable to assign more mesh points to a low conductivity region than to an adjacent high conductivity region. Figure 4 shows a hypothetical heat transfer model with a low

conductivity central region. In order to increase the mesh point density in this region, unwanted mesh points must be accepted outside the region. This situation is undesirable, and should be avoided if possible. As a practical example, an irradiation program currently being conducted by Atomics International makes use of a horizontal test facility in one of the Hanford production reactors. The fuel rods in this experiment rest on the cladding in the highly asymmetric configuration shown in Figure 5. A possible heat transfer model would have the fuel radius (R_i) change as a function of the angular position. In this case, increasing the mesh point density in the gas bond region (low thermal conductivity) would produce unwanted mesh points in the fuel regions. A superior approach would be to assign a constant fuel radius and vary the thermal conductivity in the gas as a function of the angular position, such that the thermal resistance from fuel to cladding in the symmetric model simulates the asymmetric condition. If the conductivities are chosen properly, then the temperature difference from fuel surface to cladding will be the same in both models at any angular position. Therefore,

$$\Delta T = \frac{Q}{2\pi K_o} \ln \left(\frac{r}{R_i} \right) = \frac{Q}{2\pi K_i} \ln \left(\frac{r}{R_o} \right)$$

or

$$K_i = \frac{\ln \left(\frac{r}{R_o} \right)}{\ln \left(\frac{r}{R_i} \right)} K_o$$

where

Q = heat throughput

r = outer radius (constant).

A graph of this equation (Figure 6) can be used to assign thermal conductivity values to an annular region to simulate changes in the inner radius of the region. If changes in the outer radius were to be simulated, the equation would be:

$$K = \frac{\ln \left(\frac{R_o}{r} \right)}{\ln \left(\frac{R_i}{r} \right)} K_o$$

where

r = inner radius (constant).

Similar relationships can be derived for other geometries. The asymmetric analysis described here has been verified, using electrical analog techniques.

Figure 7 shows a schematic of an irradiation experiment currently being designed at Atomics International. This experiment will be a full-scale fuel element test. A shaped flux depressor will produce the required cosine flux distribution, and a shaped gas annulus will impose a reactor temperature profile on the fuel element. A preliminary axial shape and size for the gas annulus was obtained, using the GASP code. These calculations were made, using a feature of the code which allows an axial centerline temperature profile and a corresponding heat generation rate profile to be specified. GASP will perform its one-dimensional analysis at given intervals along the length of the assembly, using the specified heat generation and temperature functions to obtain the gas annulus size at each axial location. These dimensions were used to establish modified thermal conductivities which simulated the shaped gas annulus. This model was then analyzed, using the HEATING code (Heat Transfer in Nine Geometries — a multidimensional nodal heat transfer code). Several iterations, using this code, were needed to obtain the optimum gas annulus profile. The results of this study are shown in Figure 8. The temperature fall-off at either end of the fuel element is due to axial end losses, which are difficult to avoid in a static NaK system. A method for eliminating end losses, by using nuclear heaters, has been developed for another irradiation experiment; and application of this method is being considered for the full-scale fuel element test.

In conclusion, these calculational techniques have been successfully used to design and analyze many irradiation experiments conducted by Atomics International, including experiments which will be described in other papers given at this meeting (i. e., NAA 82-1, NAA 115-1, and NAA 117-1). Typically, these experiments have operated in pile at well within $\pm 5\%$ of their design temperatures.

$$T_{AVE} = \frac{1}{2} (T_1 + T_2)$$

$$R_{AVE} = \frac{1}{2} [R_1(T_0) + R_2(T_0)]$$

$$\alpha(T_{AVE}) = A_\alpha + B_\alpha T_A + C_\alpha T_A^2 + D_\alpha T_A^3$$

$$K(T_{AVE}) = A_K + B_K T_A + C_K T_A^2 + D_K T_A^3$$

$$\Delta R = \alpha(T_{AVE}) \times R_{AVE} \times (T_{AVE} - T_0)$$

$$R_1(T_1) = R_1(T_0) + \Delta R$$

$$R_2(T_2) = R_2(T_0) + \Delta R$$

WHERE T_0 = ROOM TEMPERATURE

S = VOLUMETRIC HEAT GENERATION RATE

Q = HEAT THROUGHPUT

$$T_1 - T_2 = \frac{1}{2K(T_{AVE})} \left[\left(\frac{Q_1}{\pi} - S_1 R_1^2(T_2) \right) \ln \left(\frac{R_2(T_2)}{R_1(T_1)} \right) + \frac{S_2}{2} \left(R_2^2(T_2) - R_1^2(T_1) \right) \right] \quad (1)$$

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Figure 1. Cylindrical Heat Transfer Model

$$T_1 - T_2 = \frac{Q}{2\pi K_G(T_{AVE})} \ln \frac{R_2(T_2)}{R_1(T_1)} \quad (2)$$

$$T_2 - T_3 = \frac{1}{2K_W(T_{AVE})} \left[\left(\frac{Q}{\pi} - S R_3^2(T_3) \right) \ln \frac{R_3(T_3)}{R_2(T_2)} + \frac{S}{2} \left(R_3^2(T_3) - R_2^2(T_2) \right) \right] \quad (1a)$$

$$T_2 = \frac{T_3 + \frac{1}{2K_W} \left[\left(\frac{Q}{\pi} - S R_3^2 \right) \left(\ln \frac{R_3}{R_1} - \frac{2\pi K_G T_1}{Q} \right) + \frac{S}{2} (R_3^2 - R_2^2) \right]}{1 - \frac{K_G}{K_W} \left(1 - \frac{\pi S R_2^2}{Q} \right)} \quad (3)$$

$$R_2 = R_1 \exp \left(\frac{2\pi K (T_1 - T_2)}{Q} \right) \quad (2a)$$

$$T_3 - T_{COOLANT} = \frac{Q_3}{2\pi H R_3(T_3)} \quad (4)$$

$$\text{CONVERGENCE CRITERION: } |T_i - T_{(i-1)}| \leq \epsilon$$

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Figure 2. Gas Annulus Sizing Model

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Figure 3. Gas Annulus Design Study

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Figure 4. Typical Computer Code Lattice for Analysis of Simple Heat Transfer Model

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Figure 5. Heat Transfer Models for Assymetric Fuel Rods

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Figure 6. Thermal Conductivity Correction Factor

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Figure 7. Flux and Temperature Shaping

NOT TO SCALE

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Figure 8. Axial Temperature Profile of Prototype Fuel Element Irradiation Assembly, Using Shaped Gas Annulus for Temperature Control

THERMAL FLUX CALCULATIONS FOR EXPERIMENTS IN THE MTR

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INTRODUCTION

Atomics International, a Division of North American Aviation, Inc., has been conducting fueled irradiation experiments in the Materials Testing Reactor (MTR) since 1956. These experiments have all been designed to operate at specific fission rates and temperatures. In general, experiment results are of little practical use, unless these specified values are met with at least a $\pm 20\%$ accuracy. This means that the thermal flux levels in the test position must be known accurately, and that the flux perturbation that will be caused by a given experiment must be calculated accurately. In some irradiation facilities, if the design power is not obtained in one test position, the experiment may be switched to another position that has the proper flux level. However, many experiments, like the one to be described, have special test position requirements; so they must be designed, well in advance of the actual irradiation, for a particular test location in the reactor. Also, it is undesirable to move experiments more than necessary, because of the increased probability of losing thermocouple instrumentation or accidentally damaging the experiment. This paper describes detailed calculational techniques which have given outstanding results in predicting experiment operating power in the MTR, and also gives one example of the use of these techniques.

CALCULATION TECHNIQUES

Reactor Flux Distributions and Neutron Spectrums

The results of the nuclear calculations for an experiment must be reduced to one number, termed the "perturbation factor," before it can be decided which of the test positions would be satisfactory for the experiment. The perturbation factor is defined as the ratio of the unperturbed thermal flux level (i. e., with no experiment in the test position) to the perturbed flux level (i. e., with

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the experiment present). The perturbed flux value refers to the average value in the fuel. As an example, if the average fuel flux value is equal to 20% of the unperturbed flux value, the perturbation factor is 5.

It has been found that irradiations conducted in different test reactors require different techniques, and often different nuclear computer codes. Various types of dosimetry materials are used; often, nuclear mockups of the experiment are inserted in the reactor to verify calculational techniques. Both one- and two-dimensional diffusion and transport theory codes are used to determine perturbation factors for experiments and prototype elements. The calculational techniques are refined and corrected as actual perturbation factor measurements become available from experiments inserted in the reactor.

The technique developed for MTR perturbation calculations is strongly influenced by the geometrical location of the test positions. As illustrated in Figure 1, the test positions are located external to the core, and at widely divergent distances from the core. Since the neutron spectrum is a strong function of the distance from the core, a broad range of spectrums will exist in the test positions. This fact implies that effective thermal absorption and fission cross sections will vary also. As explained later, this variance has a strong influence on the flux perturbation calculations. The nuclear analysis and design of a specific experiment will be described in detail, to illustrate the finesse that must be used to obtain accurate answers.

Four nuclear codes are currently used in nuclear analysis of experiments in the MTR. These are called ULCER, TEMPEST, SN, and THERMOS. The version of ULCER used has a total of 23 energy groups with 8 groups in the thermal energy range and 15 in the fast energy range. Groups 15 through 23 include transfer cross sections from energy gain (i. e., upscatter) as well as energy loss (i. e., downscatter) while Groups 1 through 14 include only downscatter cross sections. This code is particularly well suited for thermal neutron spectrum calculations. The SN code is a one-dimensional cylindrical geometry transport theory code. THERMOS is a 30-group thermal neutron (0.001 to 0.7 ev) transport theory code. These latter two codes are used to perform the experiment perturbation calculations, while ULCER is used to perform the core spectrum analysis. TEMPEST is a code which takes the

weight percentages of the elements in a cell, weights the neutron spectrum according to absorption cross sections, and then gives effective absorption cross sections for each element in the "hardened" spectrum.

A cross section of the MTR is shown in Figure 1. The lattice is a 5 x 9 fixture arrangement; the north 3 rows are fuel elements, and the south 2 rows are lattice test positions. Each of these areas are homogenized for the code. The 19-plate fuel element design lends itself well to homogenization. The flux levels are approximately the same in all the materials.

It is seen that the geometry and location of the test positions must be carefully considered, to obtain good "effective absorption energy" data. Two physical situations must be considered. Consider two imaginary lines, A and B. The fluxes from the lattice along Line A see only beryllium moderator if the 3% water volume fraction is ignored, as was done for these calculations. The fluxes from the lattice along Line B see beryllium moderator, then aluminum test positions, then beryllium moderator again. The nominal 1% water volume fraction in the aluminum test position is also ignored. Consider the flux values at 98.7 cm on Lines A and B. The fast fluxes should be about the same value. The thermal flux for Line B should be considerably less, since it would be depressed by the comparatively black test fixtures. Consider the fluxes on the two lines at 112 cm. The fast flux for Line B should be considerably greater, since aluminum does not have nearly as good moderating properties as beryllium. The fast neutrons, in essence, "stream" through the aluminum. The thermal flux for Line B should be considerably less than for Line A, since aluminum has greater absorption properties, and since only a small percentage of the fast neutrons is being thermalized. The actual fluxes in the test positions are a compromise of the fluxes on these two imaginary lines. The fast flux levels should be quite close to those of Line B. The thermal flux should be more of a weighted average of Lines A and B. The thermal flux in the beryllium near the aluminum is high, compared to the thermal flux in the aluminum; but, on the other hand, the aluminum (with included experiments) should cause quite a perturbation in the beryllium local thermal flux field. The magnitude of the perturbation is dependent on what experiments are in the test positions. The slope of the flux gradient through the test position is also dependent on the number of experiments, their blackness, and their locations in the test fixtures.

Fast and thermal flux distributions are shown in Figure 2. Since the fast and thermal fluxes are relative, the fast-to-thermal flux ratios can be simply obtained. The dashed lines on the thermal flux graph represent fluxes in the north and south "A" test positions. The measured data on the thermal and fast flux graphs represent values determined by the MTR personnel. The thermal flux plot consists of ULCER Energy Groups 18 through 23 (0.001 to 0.70 ev); the fast flux plot consists of ULCER Energy Groups 1 through 3 (1 to 10 Mev). This is how MTR reports the fluxes, with the slight exception that the 2200-m/sec, MTR-reported fluxes are plotted directly against the 20°C code fluxes. The difference is insignificant, in terms of reactor thermal flux profile shape.

In deciding what the true spectral situation is, the thermal flux gradients in the test positions must be considered. Consider the 28 to 38 cm interval in the north test positions and the 100 to 110 cm interval in the south test positions. The measured and calculated gradient slopes for these two locations are given in Table 1. It is seen that the slopes for Line B are quite close to those of MTR measurements. The slopes for Line A are quite different than the MTR slopes. The fluxes of B are much closer in magnitude to the MTR fluxes than A. For example, at 110 cm, the A flux is a factor of 11.8 higher than the MTR, while the B flux is only a factor of 2.4 lower. It is felt that the B flux lines, in the test positions, should be much closer to the actual MTR flux lines than they are. The edges of the test positions at 39 cm and 99 cm are not as black, in reality, as must be assumed in the analyses. The blackness increases gradually, going into the test positions, as experiments are encountered. The first few centimeters (even more, if the test positions are not filled) are a grey aluminum approximation. A "smearing" of the boundary blackness would greatly reduce the sharp drop between 44 and 40 cm, and between 96 and 100 cm. In summary, it is believed that Line B results are a good representation of the thermal and fast fluxes distributions in the MTR. Line B results, however, do not give good spectrums inside the test positions. To clarify this fact, the distribution of effective neutron energy (E_{eff}) is now considered in detail.

The effective absorption energies (E_{eff}) for the Cases A and B are listed in Table 2. They are arrived at by first calculating the average macroscopic absorption cross sections from the following equation:

$$\bar{\Sigma}_a = \frac{\sum_{i=1}^{23} \phi_i \Sigma_{ai}}{\sum_{i=1}^{23} \phi_i} \quad \dots (1)$$

where:

ϕ_i = Flux in i^{th} energy group

$\bar{\Sigma}_a$ = Average macroscopic absorption cross section

Σ_{ai} = Macroscopic absorption cross section in i^{th} energy group

The corresponding neutron energy is then read off a plot of Σ_a vs energy (from the ULCER code cross section library).

It is seen that, in Case B, E_{eff} increases considerably in the test positions. This is not actually true in the MTR, the reason being that the test positions are only 3 in. wide (not infinitely wide, as assumed by the code). The large majority of the thermal neutrons at any given point in the test positions come directly from the nearby beryllium moderator, and are not hardened by travel in the test position material. This is seen clearly, if the test positions are viewed as "islands" surrounded by an approximately isotropic neutron flux.

As stated previously, the test positions cause a perturbation in the beryllium thermal flux field. Therefore, the thermal flux level in the beryllium for a distance around the test positions is much closer to the level in the test positions than it is to the level in the pure beryllium (Case A). The fast-to-thermal flux ratios in this surrounding beryllium area are larger than the ratios indicated by Case A for the same position. The larger ratio logically results in a higher E_{eff} , as is seen by comparison of the E_{eff} values for Cases A and B at 98 cm, and at 38 cm.

Effective energy values at 38 cm and 98 cm are considered to be good values for the north and south test positions, respectively. A correction is applied to these values to take into account the fact that the MTR reflector is actually at 125°F, or 50°C (and not at 20°C, as the code assumes).

For convenience of experiment cell calculations, the energies are translated into temperatures. Whatever the material in an irradiation experiment, whether a $1/v$ or a non- $1/v$ absorber, the cross sections can be obtained directly from the temperature. The temperature is calculated from the relation:

$$E_{\text{eff}} = \frac{4 k T}{\pi} \quad \dots (2)$$

where:

k = Boltzman constant.

E_{eff} is the energy, for a Maxwell-Boltzman thermal neutron flux distribution, that corresponds to $\bar{\Sigma}_a$ for the distribution, for a moderator at temperature (T). This relation is derived, making the assumption of $1/v$ absorbers, from the Maxwell-Boltzman equation.

$$\frac{N(E)}{N} = \frac{2\pi}{(\pi k T)^{3/2}} \exp(-E/kT) E^{1/2} \quad \dots (3)$$

where

N = total neutron density

$N(E)$ = neutron density at energy (E).

The temperatures for the north and south test positions are:

North Test Positions -

$$T = \left(\frac{\pi}{4k} \right) E_{\text{eff}} = \frac{\pi}{(4) (8.61 \times 10^{-5})} (0.0593) = 540^\circ\text{K}$$

South Test Positions -

$$T = \frac{\pi}{(4) (8.61 \times 10^{-5})} (0.0437) = 400^\circ\text{K}.$$

Experiment Flux Perturbation Calculations

The preceding analysis of the MTR was performed for only one reason, so that the proper input is available to do experiment flux perturbation calculations. As stated earlier, these calculations are done, using the transport theory computer codes, SN and THERMOS. SN is used principally, since it gives nearly as accurate answers as THERMOS for considerably less computer time.

There are four main considerations in doing experiment perturbation calculations in the MTR. These are:

- 1) The size of the nuclear cell used to mockup the experiment and environs
- 2) The method of homogenizing the materials present in the cell
- 3) The flux gradients present in the actual situation
- 4) The degree of "hardening" of the thermal neutron spectrums and cross sections in the cell.

As a means of illustrating the influence of these factors, a specific irradiation experiment design which was irradiated in the MTR will be considered. This experiment is more complicated, from a nuclear design standpoint, than the average experiment that Atomics International designs for the MTR.

NUCLEAR DESIGN OF THE NAA 82-1 EXPERIMENT

Experiment Description

The NAA 82-1 Experiment contained a SNAP 2 prototype fuel rod. The experiment objectives are listed in Figure 3. It was quite important that the fuel rod operate within 20% of the design conditions.

The irradiation was initiated in the MTR in April 1964, and terminated in August 1965. The objectives were fully met; the experiment operated within 5% of the design objectives of burnup rate and total burnup. Atomics International Hot Laboratory (AIHL) postirradiation examination burnup results are shown in Figure 4.

There were two factors that caused the nuclear designs of this experiment to be quite involved. One was that the experiment was quite large; it required the space of four standard MTR experiment test positions. As such, because of the limited availability of test positions, the location required had to be decided upon well in advance of the irradiation. The second factor was the low thermal flux level required in the test fuel. Even taking the lowest flux positions in the MTR, the perturbation factor would have to be in the range of 10 to 20. Factors this large are susceptible to small theoretical differences, and consequently are difficult to predict.

Perturbation Factor Calculation

To satisfy the design requirements of flat flux across the diameter of the experiment and large perturbation factor, an eccentric poison shroud tube, made of 50 wt % cobalt Haynes No. 25 alloy, was built to fit over the experiment capsule. The flux depressor shroud tube is shown in Figure 5, which also shows a picture of the capsule. The tube varied in thickness, from 0.050 to 0.250 in., around its circumference. The thick part of the tube faced the reactor core and depressed the thermal flux; so that it was essentially flat inside the tube, where the experiment was located, as depicted in Figure 6. A more detailed plot of the thermal flux depression is given in Figure 7.

The thermal spectrum is hardened, coming from the beryllium moderator, through the flux depressor tube, and into the capsule. For purposes of flux perturbation, this hardening is considered as two stages. There will be three distinct sets of cross sections for the SN code:

- 1) For the beryllium moderator and aluminum test piece external to the flux depressor tube
- 2) For the flux depressor tube
- 3) For the capsule inside the tube.

The amount of hardening is determined by considering the spectrum change in highly boronated aluminum material placed at the test position locations. The ULCER results for this situation are presented in Table 2, and are graphed for the test position locations in Figure 8.

Hardening is a function of the macroscopic absorption cross section and the thickness of the material (t). It is assumed that the amount of hardening is identical for equal ($\Sigma_a \cdot t$) values going into the experiment and into the boronated aluminum material. The E_{eff} value for the flux depressor tube cross sections is the value at the middle of the tube; the E_{eff} value for the capsule cross sections is the value at the inside diameter of the tube. These two values change for different SN cases, as the thickness of the depressor tube is changed. Values used for different thicknesses and test position locations are given in Table 3. For example, the temperatures of Case II in Table 3 are calculated as follows:

Flux Depressor Tube -

$$\Sigma_a \cdot t (\text{Aluminum-Boron}) = \Sigma_a \cdot t (\text{tube}) \quad \dots (4)$$

$$t (\text{Aluminum}) = \frac{\Sigma_a \cdot t (\text{tube})}{\Sigma_a (\text{Aluminum})} \frac{(1.71 \text{ cm}^{-1}) \left(\frac{0.381 \text{ cm}}{2}\right)}{0.124 \text{ cm}^{-1}}$$

$$t = 2.6 \text{ cm} .$$

From Figure 8, $E_{\text{eff}} = 0.0473 \text{ ev} .$

Corrected to 125°F (MTR moderator physical temperature):

$$E_{\text{eff}_2} = \frac{T_2}{T_1} (E_{\text{eff}_1}) \quad \dots (5)$$

$$= \frac{325^\circ\text{K}}{293^\circ\text{K}} (0.0473) = 0.0525 \text{ ev} .$$

The corresponding temperature is: (Equation 2)

$$0.0525 \text{ ev} = \frac{4(8.61 \times 10^{-5}) T}{\pi}$$

$$T = 479^\circ\text{K} .$$

Capsule -

$$t = \frac{(1.71 \text{ cm}^{-1})(0.381 \text{ cm})}{0.124 \text{ cm}^{-1}} = 5.2 \text{ cm} .$$

From Figure 8, $E_{\text{eff}} = 0.0550 \text{ ev} .$

Corrected,

$$E_{\text{eff}} = \frac{325}{293}(0.550) = 0.0611 \text{ ev} .$$

The temperature is:

$$0.0611 = \frac{4 (8.61 \times 10^{-5}) T}{\pi}$$

$$T = 557^{\circ}\text{K} .$$

This technique assumes that the spectrum in the Aluminum-Boron material is effectively Maxwellian in shape, and that fast neutron thermalization in the material is negligible. Detailed inspection of the spectrums verifies that these are reasonable assumptions.

The next step in the nuclear analyses is actual calculation of the perturbation factor for the experiment. The one-dimensional, cylindrical geometry SN code is used. The cross sections were all obtained from the TEMPEST code. The moderator temperature input for TEMPEST was 125°F for SN Cases 1, 2, 4, 5, 6, 9, 10, and 11; for SN Cases 3, 7, 8, and 12, the proper effective temperature values from Table 3 were used.

A summary of the perturbation factors obtained is given in Table 4. Cases 7 and 8, where the flux depressor tube is half of its maximum thickness, are the best values. The factor for the south test positions is 13.7, and for the north test positions is 11.9. It is important also to use a large enough cell radius, so that the flux shape becomes asymptotic at the cell boundary of its own accord with the physical situation, and not because it was mechanically forced to by the code boundary conditions. In this analysis, an 18-cm cell

radius was ample, but a 10-cm radius was not, as seen by the perturbation factors listed in Table 4. The 10-cm distance, in essence, "cut off" part of the depressed flux curve. It is also seen that, for a proper 18-cm cell radius, if the physical moderator temperature of 125°F had been used, a perturbation factor of 20.8 would have been calculated, versus the correct values of 13.7 and 11.9. The thickness of the shroud tube would have been considerably greater, and the experiment would have been operated at ~60% of design conditions.

The SN perturbation factors were verified by the THERMOS code within 5%. Both the THERMOS code and the SN-TEMPEST combination give correct answers, with the latter using only about 50% of the computer time.

TABLE 1
THERMAL FLUX GRADIENTS

Case	(ϕ/ϕ_0) North	(ϕ/ϕ_0) South
MTR Measured	0.43	0.30
Line "A"	0.79	0.66
Line "B"	0.44	0.35

TABLE 2
ULCER CASES E_{eff} FOR
DIFFERENT LOCATIONS IN REACTOR

Reactor Location (cm)	E_{eff} (ev)		Highly Boronated Test Positions
	A	B	
5.6	0.0325	0.0646	0.0702
17.3	0.0343	0.0921	0.130
25.3	0.0352	0.0785	0.113
35.3	0.0380	0.0605	0.0704
45.4	0.0410	0.0495	0.0498
66.6	0.0556	0.0557	0.0559
72.5	0.0510	0.0513	0.0513
76.9	0.0405	0.0405	0.0405
83.4	0.0378	0.0382	0.0383
88.2	0.0365	0.0381	0.0380
93.4	0.0360	0.0392	0.0392
102.7	0.0343	0.0454	0.0530
112.7	0.0335	0.0580	0.0808
122.7	0.0335	0.0680	0.108
135.7	0.0300	0.0542	0.0630

TABLE 3
E_{eff} VALUES USED WITH SN CODE

Case	Tube Thickness (in.)	Test Position Location	Be Moderator		Flux Depressor Tube		Capsule	
			E _{eff} (ev)	T (°K)	E _{eff} (ev)	T (°K)	E _{eff} (ev)	T (°K)
I	0.050	South	0.0437	400	0.0465	424	0.0500	456
II	0.150	South	0.0437	400	0.0525	479	0.0611	557
III	0.250	South	0.0437	400	0.0594	542	0.0732	668
IV	0.150	North	0.0593	541	0.0723	659	0.0852	777

TABLE 4
RESULTS OF SN CASES

Case	Flux Depressor Thickness (in.)	Maxwellian at 125°F, 10-cm Cell Boundary	Flux Perturbation Factors		North Positions Analysis 18-cm Cell Boundary
			Maxwellian at 125°F, 18-cm Cell Boundary	South Positions Analysis 18-cm Cell Boundary	
1	0.0	5.89			
2, 3	0.050	6.56		10.3	
4	0.100	7.12			
5, 6, 7, 8	0.150	7.55	20.8	13.7	11.9
9	0.200	7.87			
10, 11, 12	0.250	8.08	26.7	17.2

7568-15211

Figure 1. MTR Cross Section

7568-15212

Figure 2. MTR Fast and Thermal Fluxes

AXIAL BURNUP PROFILE

1. PEAK BURNUP IS 0.08 ma % $\pm$ 5%
2. FLAT FUEL BURNUP AROUND FUEL PERIPHERY $\pm$ 10%

- MASS-SPECTROGRAPH DATA*
- RADIAL GAMMA SCAN DATA**
- △ AUTORADIOGRAPH DATA**
- AXIAL GAMMA SCAN**
- ▽ HEAT TRANSFER CALCULATION*

NOTES: *ABSOLUTE VALUES

**DATA HAS BEEN NORMALIZED TO 0.08 ma % PEAK BURNUP FOR COMPARISON.

7568-15221A

Figure 3. NAA 82-1 Irradiation Assembly Objectives

NUCLEAR OBJECTIVES: ($\pm$ 20% ACCURACY REQUIRED)

- 1) 0.08 ma %/POWER-YEAR BURNUP RATE
- 2) 0.08 ma % TOTAL PEAK BURNUP
- 3) FLAT THERMAL FLUX AROUND FUEL PERIPHERY

NUCLEAR DESIGN:

- 1) ECCENTRIC FLUX DEPRESSOR SHROUD TUBE AROUND EXPERIMENT (50 wt % COBALT ALLOY USED)

7568-15220

Figure 4. Results of Postirradiation Examination of NAA 82-1

Figure 5. NAA 82-1
Irradiation Assembly
Hardware

7568-15222

Figure 6. Thermal Flux
Depressing and Shaping

7568-15223

7568-15213

Figure 7. Results of Test Position Flux Perturbation

7568-15214

Figure 8. MTR Effective Neutron Temperatures

III. NOVEL CAPSULE DESIGNS - FISSILE MATERIALS

CAPSULE DESIGN FOR ISOTHERMAL IRRADIATION OF UO₂

By

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3.1.1

"CAPSULE DESIGN FOR ISOTHERMAL IRRADIATION OF UO_2 "

By G. A. Esswein

Four capsules have been irradiated in the GETR to determine the fission gas release fraction from sintered UO_2 fuel while operating isothermally at nominal temperatures of 1500, 1700, 1900 and 2100°C. The purpose of these irradiations was to refine the knowledge of the UO_2 fission gas release variation within the critical temperature range for release, i.e., the temperature range between ~1500°C and 2100°C over which the release fraction increases from below 1% to nearly 100%. A unique capsule design was evolved for the irradiations with the objective of minimizing radial and axial temperature gradients within the isothermal sample and maintaining a constant sample temperature over a period of days, despite continuous neutron flux variations. The actual fission gas data obtained from the capsules was partially compromised by repeated cracking of the tungsten cladding enclosing the isothermal UO_2 samples. However, the ultimate capsule design performed beyond expectations in achieving and maintaining the desired operating conditions and warrants a detailed description.

A cutaway drawing of the capsule design is given in Figure 1 with the major components identified. Each capsule contained an isothermal UO_2 sample consisting of four, one-half inch long, annular pellets. The pellet O.D. is 0.295 in and the I.D. 0.130 in. The total isothermal fuel weight is 18.6 gms and the UO_2 enrichment 1.5 percent. A one-eighth inch O.D., tungsten-rhenium alloy sheathed, tungsten-rhenium thermocouple is inserted in the central cavity of the isothermal UO_2 pellets to indicate the isothermal sample temperature during irradiation and provide the primary signal for the capsule automatic temperature control system.

The isothermal sample pellets are enclosed in a concentric tungsten clad tube which in turn is surrounded by larger diameter, annular UO_2 pellets. By virtue of their own internal heat generation under irradiation, these larger annular pellets develop the major portion of the temperature differential required to raise the isothermal sample to the desired operating temperature and, therefore, this outer UO_2 has been named the "heater-insulator" fuel. The "heater-insulator" fuel consists of two different enrichments, 2% and 3%. The axial enrichment arrangement was determined by computer analysis and is designed to minimize axial temperature gradients in the isothermal fuel sample from the known axial neutron flux distribution at the irradiation position and from axial heat conduction along the tungsten cladding toward the water cooled,

upper end plug. The total "heater-insulator" fuel weight is approximately 350 gms. The "heater-insulator" fuel is surrounded by stainless steel cladding and the clad temperature monitored by two chromel-alumel thermocouples fitted into milled slots in the stainless steel sheath outside surface.

The "heater-insulator" cladding is, in turn, surrounded by a second, concentric stainless steel clad, forming an annulus between the two with a cold gap thickness of 0.009 in. The capsule heat output is removed by cold water flow directly over the outer stainless sheath and measured by chromel-alumel thermocouples at the coolant inlet and outlet. The annulus between the two clads is filled with a mixture of helium and nitrogen gas to permit precise automatic adjustment of the capsule internal temperatures. The provision of an automatic, binary gas temperature control system avoids the need for frequent manual movements of the capsule in the GETR Trail Cable Facility despite continuing changes in the neutron flux at the capsule position. The automatic control system functions by feeding the output from the central tungsten-rhenium thermocouple to an automatic gas controller where the signal is compared with the pre-set reference temperature. The gas controller then increases or decreases the ratio of helium to nitrogen, and hence the thermal conductivity, of the gas mixture fed to the gas gap. The resulting change in temperature drop across the gas-gap reduces the deviation between the actual central temperature and the desired value. Figure 2 shows the experimentally observed variation in the "heater-insulator" clad temperature as a function of the volume percent nitrogen in the gas-gap for capsules 1 and 2. Also plotted is the analytically predicted variation assuming unrestricted expansion of the "heater-insulator" clad. The curves are based on an average heat flux of $150,000 \text{ BTU/h-ft}^2$ (47 W/cm^2) and an initial, cold gap of 0.009 in (.023 cm). The thermocouples on the "heater-insulator" clad can also serve for controlling the isothermal sample temperature at a constant value in the event the center thermocouple fails during irradiation. However, the capsule center thermocouple must operate long enough initially, at equilibrium conditions, to establish the correspondence between the two temperature values.

Figure 3 shows a plot of the performance data obtained during the irradiation of the first IFFGE (Isothermal Fuel Fission Gas Experiment) Capsule. Aside from the one "heater-insulator" thermocouple failure the irradiation was very successful. During the final 145 hours of irradiation, the sample temperature was controlled within $\pm 15^\circ\text{C}$ of the mean operating temperature. The other three capsules were all built virtually identical to the first and the irradiation performance from each was equal or better than the first capsule.

The fission gas released from the isothermal sample is collected following irradiation by puncturing through a blind, pre-drilled hole in the top end plug which provides access to the sample plenum chamber. Of the four capsules irradiated only the second, with an isothermal sample temperature of 1700°C, was found to have retained its tungsten clad integrity. As a result, the fission gas release in the other three capsules was a composite from both the isothermal sample and the "heater-insulator" fuel. This contamination of the isothermal gas release has rendered the release fraction data less accurate than originally desired. However, a reasonably good estimate of the isothermal release at each temperature has been obtained by using the data from the one unfailed capsule combined with careful temperature analysis. The results are given in Table I below and apply to UO_2 with very low burnup, on the order of 80-100 MWD/Te

TABLE I

FISSION GAS RELEASE RESULTS (PRELIMINARY)

<u>TEMPERATURE RANGE</u>	<u>PERCENT RELEASE *</u>	
	<u>Kr⁸⁵</u>	<u>Xe¹³³</u>
700-1500°C	.8%	1.4%
1650°C	1.0%	1.6%
1700-1900°C	4.2%	6.8%
1900-2100°C	~20%	~32%

* Results based on gamma spectrometry data and branching ratios from GEAP 3496 and GEAP 3440.

IFFGE CAPSULE

TUNGSTEN SAMPLE
CLADDING

TEMP. CONTROL
GAS GAP

ANNULAR "HEATER-
INSULATOR" FUEL

UO₂ FISSION GAS
SAMPLE PELLETS

SCALE-INCHES

0
1
2
3
4
5
6

Figure 1

Figure 2 "Heater-Insulator" Cladding Temperature Variation with Increasing Percentage of N₂

MOLTEN FUEL IRRADIATION TESTS^(a,b)

by

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Figure 3 Irradiation Record for Capsule IFGE-1

MOLTEN FUEL IRRADIATION TESTS

by

L. A. Pember, F. E. Panisko and M. K. Millhollen

With the advent of the Plutonium Recycle Test Reactor high power density program and the proposed steady state operation with near melting of the fuel, there have been several points of concern about the performance of oxide fuels. Two areas of concern are (1) what happens if fuel cracking occurs at high temperatures; will molten fuel impinge the cladding and cause a rupture? and (2) is there a possibility that the plutonium will migrate in a fuel rod, causing hot spots?

A series of tests have been irradiated in an attempt to determine the possibility of molten fuel impinging the cladding through cracks formed during a reactor shutdown or during operation. In an attempt to simulate rather gross cracks, capsules containing UO_2 pellets, one pellet of which had a hole drilled along a diameter, were irradiated. Two tests, one single rod element and one four rod cluster, were irradiated in the B-3 facility to determine the effect of a gross crack during a normal reactor startup. Four capsules were irradiated in the VH-4 hydraulic rabbit facility to determine the effect of a nearly instantaneous increase in power or of a crack occurring during operation.

The first of this series of tests was a Zircaloy-2 clad single rod, 1.08 inches in diameter with a 0.060 inch wall thickness. The fuel was 1.6% enriched UO_2 pellets which had a density 10.3 grams/cc. One pellet had a 0.10 inch hole drilled along a diameter of a pellet to allow free access of

molten UO_2 to the cladding. This pellet was positioned so that the hole was 2.50 inches from the bottom of the fuel length. This element operated at a calorimetrically determined average power generation of 43 KW/ft and attained an exposure of 500 MWD/MT. Post-irradiation examination gave no indication of any heat affected areas on the cladding near the position of the hole. The drilled hole was not located during destructive examination. The only explanation we can offer for this is that when the section was cut from the rod the cut-off wheel passed directly through the drill hole. From examination of the fuel microstructure, we estimate that 80% of the fuel diameter was molten. During examination of several transverse sections of this rod, one area was found where molten UO_2 did appear to reach the cladding through an actual crack. Measurements from photographs indicate that this crack was between 0.010 and 0.012 inches in width. Examination of the cladding in this area showed that this impingement had no apparent adverse effect on the cladding.

The second test was a four rod cluster clad with 0.565 inch OD by 0.030 inch wall Zircaloy-2. The arrangement of the fuel was similar to the first test in all four rods of this cluster. The only notable change was a variation of drill hole sizes in the pellet of each rod. The holes in this cluster were nominally 0.025, 0.050, 0.075, and 0.100 inches in diameter.

This element was designed to operate at a power generation of 30 KW/ft rod, but calorimetric measurements showed that the actual power generation was approximately 25 KW/ft. At the present time only one rod from this cluster has been destructively examined. This examination indicates that the fuel

in the area of the drilled hole was not molten during irradiation. The drilled hole, however, was closed by the formation of large columnar grains.

Since capsules irradiated in the hydraulic rabbit facility undergo a nearly instantaneous increase from zero to full power, this should be a much more severe test than any fuel rod will see during controlled power increases. The four capsules irradiated in the rabbit facility were double clad, Zircaloy-2 shrink-fit into an aluminum outer cladding. The fuel makeup and drill hole diameters in these capsules were the same as those in the four rod cluster. Design power was again 30 KW/ft, but no calorimetric data are available to calculate actual power generation. These four capsules were each charged while the MTR was at full power. Each was irradiated for a thirty minute period, after which reactor power was decreased to allow their safe discharge. At the present time the only examination completed on these capsules is a visual examination of the zircaloy cladding after removal of the aluminum. No indication of a heat affected zone is apparent on the zircaloy cladding.

Two diffusion test elements were irradiated in the B-3 loop of the MTR to obtain information on migration of plutonium in mixed oxide fuel during irradiation. These tests were four rod clusters in which mixed oxide pellets were interspersed with UO_2 pellets. In these tests, one inch of UO_2 pellets was placed on each side of a mixed oxide pellet one-half inch long. The mixed oxide pellets used in these tests were all from the same batch of UO_2 2% PuO_2 power. The UO_2 pellets used in the first test were natural UO_2 while those used in the second test were 2.9% enriched.

The first test was used to determine the diffusion characteristics in the region around the natural UO_2 pellet which separates the mixed oxide fuel from the fission gas plenum on the PRTR short core elements. The mixed oxide pellets in this test received an exposure of approximately 2100 MWD/MT of fuel. Examination of this test showed there was migration of plutonium into the UO_2 pellets approximate to the limit of large columnar grain growth and that the boundary of the plutonium concentration was sharp and well defined.

In the second test, the power generation of both the mixed oxide pellets and the enriched UO_2 pellets was approximately equal to that of an average high power density core rod. The exposure of this test was 4600 MWD/MT. Post-irradiation examination indicated that a small amount of vapor phase diffusion took place within the central cavity which formed during irradiation. The results of these tests indicate that the diffusion of plutonium should cause no undue hazard during high power operation of mixed oxide fuels.

20,33605

In-Pile Calorimetry*

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ABSTRACT

The purpose of this paper is to propose an experiment in which capacitor pulse heating techniques would be used to determine in-pile specific heat data for various materials of interest in reactor design as a function of exposure to integrated neutron flux.

The pulse heating method is a relatively new technique which uses the electrical discharge of a known amount of energy stored in a capacitor bank through a given sample to rapidly ($30 \mu\text{sec}$) heat it. By measuring the temperature rise and knowing the energy stored, the specific heat of conductors may be computed. Among the many advantages associated with this technique is the fact that this entire operation may be performed remotely and repeatedly over long time intervals with good precision. Hence, it offers the unique opportunity of providing data on specimens subjected to a neutron flux ($5 \times 10^{13} \text{ n/cm}^2 \text{ sec}$) without having to remove the specimen from the flux, as would normally be the case with more conventional calorimetric test methods.

The capsule design incorporates some interesting features:

- 1) Large insulated electrical cables capable of withstanding many thousands of amps and a reliable sealed electrical connection through the capsule and to the specimen must be used.

*Work performed under the auspices of the U. S. Atomic Energy Commission.

- 2) The design must make installation of the specimen and instrumentation easier.
- 3) The overall capsule dimensions should be small enough (3 in. in diam $\times$ 10 in. long) to fit in the sample facility and still provide enough electrical insulation to withstand the high (kV) potential applied by the capacitor bank across the specimen and capsule.
- 4) The capsule and interior bus work must provide a low-inductance electrical path.

Detailed capsule sketches and electrical bus bars are shown in the proposal, as well as a list of material and total integrated neutron flux exposures that the materials will experience. Analytical expressions governing the electrical pulse heating method are derived. In addition, the test procedure is outlined.

INTRODUCTION

In recent years, a large number of experiments have been performed to investigate the effects of radiation damage upon the mechanical properties of materials.^{1,2,3} However, very little experimental work concerning radiation effects on the specific heat of materials has been performed. This information would be useful for materials used in reactor design. In addition, basic data concerning nonreactor materials might shed more light upon the precise mechanisms involved in radiation damage. This paper describes an experimental procedure that may be used to investigate these parameters. Particular attention is given to capsule design and the problems associated with the reactor environment. A general description and derivation of the capacitor-pulse heating technique, along with its associated instrumentation, is shown.

Published work by other experimenters^{4,5} on the effects of irradiation upon the specific heat of graphite indicates that this is a promising material for experimentation. In addition, metastable uranium alloys, such as U-Nb, will also be investigated.⁶ The Cu₃-Au alloy⁷ will be irradiated and the effects on short-range ordering will be measured by means of changes in specific heat.

CAPSULE DESCRIPTION AND FUNCTION

The capsule shown in Fig. 1 is made of aluminum, is cylindrical, and has a 1/16-in. wall and 1/4-in. ends. Besides being a good material for long duration in-core irradiation, the design makes use of the low resistivity of aluminum, as the capsule structure is an integral part of the circuit design. The outside cylinder forms the coaxial shield for the return current, thus minimizing the inductance in the circuit. The current path is shown in Fig. 1. The 1/16-in. wall easily carries the high current and provides a sound structural support. The main conductor which carries the incoming energy pulse and the two thermocouple leads are electrically insulated from the structural body of the capsule by three high-alumina ceramic terminals welded into the top end-plate. The specimen is held by two screw clamps, one in the main conductor and one in the aluminum stud which is welded to the bottom end of the capsule. The stud and top-plate form a subassembly with two ceramic standoffs. Ceramic standoffs are necessary to preserve the coaxial effect of the cylinder. One standoff also serves to support the thermocouple leads. The thermocouple is small and delicate, and experience has shown that it is very profitable to take care to insure the integrity of the installation. The

outside of the cylinder is hard-anodized to a thickness of 2 mills to electrically insulate it from the reactor core.

The capsule is assembled by welding the three ceramic terminals into the top end-plate. The ceramic standoffs are screwed into the plate, and one end of the specimen is attached to the main conductor with the screw clamp. The aluminum stud is placed on the standoffs while the other end of the specimen is placed in a slot in the stud. The nuts are tightened on the standoffs and the specimen is secured with the screw clamp. Care must be taken in securing the specimen because some materials, such as graphite, have a coefficient of expansion that is quite different from aluminum. The specimen must be free to move while maintaining good electrical contact. If good electrical contact is maintained, the other instrumentation will not be affected by expansion characteristics. The thermocouple is then attached to the specimen and the leads wrapped around one of the ceramic standoffs.

After each of the following phases of assembly the thermocouple is tested: The subassembly is placed in the cylinder, held with a C-clamp, and the stud and top end-plate are welded to the cylinder. The welds are machined to the nominal outside diameter and the assembly is leak-checked. To accomplish this the capsule is immersed in water in a bell jar and the air above the water is evacuated. Bubbles from the capsule indicate a leak. Then the cylinder is hard-anodized except for the area on the top end-plate where electrical contact is to be made. The main conductor is attached to a junction joining it to the four RG-8 cables, and the top end plate is electrically connected to the shield of the cables.

PROCEDURE

This experiment is planned for the Livermore Pool Type Reactor (LPTR). The LPTR is a tank-type thermal, heterogeneous research reactor. The reactor core is light-water moderated and cooled, and will generate a peak thermal flux of $5 \text{ to } 10^{13} \text{ n/cm}^2/\text{sec}$ (at the C-4 position) when operating at 2 MW. This peak flux is in the flux trap at the center of a 5×7 element grid configuration of MTR-type fuel elements and beryllium reflector elements.

The fast flux at the C-4 position is $5.7 \times 10^{12} \text{ n/cm}^2/\text{sec}$ ($E > 3 \text{ MeV}$) and the cadmium ratio is 4.9. Since it is believed from other effects that the threshold energy of interest is 25 eV, it is calculated that this flux is $1.8 \times 10^{13} \text{ n/cm}^2/\text{sec}$ ($E > 25 \text{ eV}$).

The LPTR normally runs 100 hours a week, starting on Monday evening and shutting down on Saturday morning for the weekend. Thus, it will take about 15 weeks to reach an integrated flux of 10^{20} n/cm^2 . Besides recording the effects of the integrated flux on the specific heat and resistivity, data will also be taken to determine what, if any, effects take place over the weekend. The capacitor bank and instrument readout will be located on the reactor top to minimize lead lengths. The capsule will be lowered into the C-4 position by the four RG-8 cables. The specimen will be pulsed at zero power and several intermediate power levels on the initial run to check the function of all phases of the experiment. Probably a minimum of eight pulses per week will be taken after the initial start-up - two each at zero power, a stable 2 MW, midweek, and at the end of the week. In the event of inadvertent reactor shut-downs, measurements will be taken after the reactor has reached a stable 2 MW, as determined by a stable Δt . Because of induced radioactivity, the capsule will not be considered recoverable.

THEORY

The measurement of specific heat may be accomplished by a number of conventional methods, such as a drop calorimetry and differential thermal analysis experiments. In the drop-calorimetry technique, a given mass of material is heated to some equilibrium temperature, and then dropped into an insulated liquid at a different temperature whose specific heat and mass is known. The equilibrium temperature is then measured. The resulting heat balance may be used to determine the specific heat of the unknown material.

This method and variations of it have been used successfully in certain applications for many years. However, it does offer a prime disadvantage when trying to determine specific heats of materials exposed to a neutron flux in a reactor; namely, that the capsule and specimen (which may be radioactive) must be removed from the reactor core and the experiment performed. This of course may introduce a large number of unwanted variables in the experiment. The differential thermal analysis (DTA) and adiabatic calorimetry methods both require heaters and control mechanisms to maintain a particular heat input. Changes in neutron flux may introduce a heat input or heat loss, or effect the samples (as in DTA) so as to introduce errors. Moreover, these methods can be expensive and time-consuming when used in a reactor core. An alternate technique which appears to be readily adaptable to the requirements of in-pile calorimetry has recently been developed.

This new method employs the discharge of a known high-voltage, low-inductance capacitor through an electrically-conducting specimen, causing its temperature to increase rapidly ($30 \mu\text{sec}$). The temperature is monitored with 0.001-in.-diam chromel-alumel thermocouples welded to the specimen, forming an intrinsic installation. The two leads of the thermocouple are

spot-welded 1/32 in. apart to form two distinct junctions with the specimen material. This type of arrangement provides for a very small mass at the junction and hence a minimal thermal inertia, so that the temperature rise of the specimen may be recorded very quickly (approx 400 μ sec). With a means of generating a rapid temperature rise and a technique of recording it quickly, heat losses to the surroundings will be minimized so that an accurate determination of specific heat may be made. Figure 2 shows the electrical arrangement of the components in the circuit.

The following expressions may be derived to determine the specific heat:

$$CV^2/2 = \text{potential energy stored in the bank,} \quad (1)$$

$$\Delta TmC_p = \text{heat energy,} \quad (2)$$

where C_p = specific heat (cal/gm $^{\circ}$ C).

It can be shown that after complete discharge,

$$\Delta T = \frac{CV^2}{2} \frac{R_s}{R_b + R_s} \frac{0.239}{C_p M} \quad (3)$$

or rearranging,

$$C_p = \frac{CV^2}{2} \frac{R_s}{R_B + R_s} \frac{0.239}{M \Delta T} \quad (4)$$

where

C = bank capacitance in μ F,

V = bank voltage,

ΔT = temperature rise in $^{\circ}$ C,

M = mass of specimen in g,

R_s = specimen resistance in ohms, and

R_B = overall bank resistance in ohms.

R_B can be determined by replacing the specimen with a large heavy copper bus bar of very small resistance. The bank is then fired and the voltage decay monitored. Since the circuit is an under-damped R-L-C circuit, the R-L-C constants may be computed from the decay. The frequency of an underdamped R-L-C circuit is

$$W = \sqrt{\frac{1}{LC} - \frac{R^2}{4L^2}} = \beta, \quad (5)$$

β = a constant

L = circuit inductance in μH

Since $R^2/4L^2$ is small,

$$W = \sqrt{\frac{1}{LC}} \quad (6)$$

As W is also given by

$$W = \frac{2\pi}{t}, \quad (7)$$

then

$$L = \frac{t^2}{4\pi^2 C} \quad (8)$$

Where t is the time between each peak amplitude, the voltage decay is

$$V = Ke^{-Rt/2L} (\sin \beta t), \quad (9)$$

where K is a constant.

Using a log decrement, one may measure the amplitude decay and calculate the bank resistance.

INSTRUMENTATION

The diode-clipper circuit provides high-voltage isolation of the oscilloscope during the bank ring-down or high-voltage decay. This is necessary to prevent high-voltage saturation when the oscilloscope is measuring the small (mV) signals from the thermocouple. The bank is equipped with a Pacific Power Corporation dc power supply with 1 volt in 6000 volts resolution. Hence, only a static measurement of bank voltage and specimen resistance and a dynamic measurement of the temperature rise of the specimen is required to determine specific heat. In addition, one may also measure the change of electrical resistance of the specimen as a function of integrated flux by using the constant current source as shown in Fig. 2, and measuring the voltage drop across the specimen. This drop will be proportional to the resistance. Hence, in addition to measuring the variation in specific heat with the integrated neutron flux, resistivity may also be determined.

This system also has an additional flexibility in that the current used to monitor the change of resistivity of the specimen may also be used to increase the temperature of the specimen. Hence, specific heat data may be determined as a variation of temperature as well as integrated neutron flux.

DISCUSSION

The LPTR environment presents some obvious areas of difficulty in performing this experiment. Since it is a pool-type reactor, the electrical leads are necessarily about 30 ft long. This length, along with a portion of unshielded cable near the core, probably adds about $0.2 \mu\text{H}$ to the circuit inductance and tends to slow down the heating time of the pulse. However, the pulse speed is such that this inductance will not appreciably effect the circuit. Part of the

cable is unshielded because the four RG-8 cables must be joined into a single conductor and the coaxial cables should not be subjected to the intense radiation adjacent to the core.

The LPTR normally has 1100 gpm of water flowing over the core, and this will cause the capsule to vibrate in the core. This vibration could cause capsule leaks which the all-welded assembly should prevent. The thermocouple leads will be given a free loop near the point of installation to absorb vibration and shock. The vibration could also damage the outside anodized insulation. Since the high potential is always put on the inside of the RG-8 cables and the ground is on the outside cylinder of the capsule, this should not cause difficulty.

Since the thermocouple is made of chromel-alumel, a fairly low-z material, it is expected to endure the environment. Electronic noise is not expected to be an insurmountable problem, nor is safety. The electrical pulse to the specimen is about 200 J, or 0.06 W-hr.

CONCLUSION

The remotely-operated, pulse-heating method may be used to make in-pile measurements of the specific heat of conductors quickly and conveniently. Although the effects of long-term exposure to radiation and other environmental conditions to be encountered in the reactor may cause unforeseen component failure, no major modification of the present design is expected. In addition, it is hoped that at some later date the pulse-heating technique, in which apparatus quite similar to that previously described is used, may be used to simultaneously determine in-pile thermal diffusivity as well as specific heat of conductors exposed to some integrated neutron flux.

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Figure 1. Calorimetry capsule

CONTROLLED TEMPERATURE-PRESSURE
IRRADIATION CAPSULES

By

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Figure 2. Capacitor circuit diagram

CONTROLLED TEMPERATURE-PRESSURE IRRADIATION CAPSULES

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This paper describes the controlled temperature and controlled temperature-pressure, NaK filled irradiation capsules that were developed at the Pacific Northwest Laboratories in conjunction with a program designed to study the basic mechanisms of the swelling (increase in volume) which occurs during the irradiation of fissionable and fertile metallic fuels. These capsules are used to study the effects of temperature; external restraint; specimen composition, structure, and geometry; burnup; and burnup rate on the behavior of metallic fuel specimens during irradiation. Physical descriptions of the capsules which have a controlled temperature range of 200-800°C and a controlled pressure range of 30-5000 psi are provided. A brief discussion is included on the design of the pressure shell, a new weld configuration used, and the effects of high pressure helium on heat transfer. The high integrity components, the extreme care exercised during construction, and the improved instrumentation have provided a successful irradiation device for studying the behavior of fissionable materials. During the six years that capsules of this type have been irradiated, very few thermocouple or heater failures have occurred and no capsule has incurred general failure.

Much of the information presented in this paper has application to the general field of capsule construction and irradiation.

INTRODUCTION

This paper describes the controlled temperature and controlled temperature-pressure, NaK-filled irradiation capsules that were developed at the Pacific Northwest Laboratories in conjunction with a program designed to study the basic mechanisms of the swelling (increase in volume) which occurs during the irradiation of fissionable and fertile fuel materials. These capsules are currently used to study the effects of temperature; external restraint; specimen composition, structure, and geometry; burnup; and burnup rate on the behavior of metallic fuel specimens during irradiation. These parameters are evaluated by recovering the specimens from the irradiated capsule and subjecting the specimens to a detailed postirradiation examination. The information so obtained

enhances our fundamental understanding of irradiation effects and is used to improve the design of metallic fuel elements.

In order to satisfy the stringent requirements of the Swelling Program several unique capabilities had to be built into these capsules that are not normally required of irradiation capsules.

- . The temperature had to be maintained constant regardless of the power level of the reactor, including "down" time.
- . A broad range of heat dissipation capability had to be provided in order to accommodate various sizes, shapes, and enrichments of specimens irradiated over a wide temperature range.
- . In a more recent phase of the program a preselected pressure also had to be constantly maintained.
- . Small specimens with a variety of geometrical shapes had to be positively positioned without being externally restrained by the holder.
- . Thermocouples had to be precisely located to accurately maintain the specimen temperatures without providing external restraint or interfering with the holder requirements.
- . High purity NaK had to surround the specimens at all times while permitting complete freedom of capsule position during capsule construction.
- . The specimens had to be recovered after irradiation without damaging them and without introducing fire hazards from the highly reactive NaK.
- . The integrity of all components had to withstand the rigors of the reactor environment for periods up to a year and longer.
- . The capsule had to survive a fall of 20 to 30 feet into the discharge basin after irradiation without failing the outer tube or damaging the

specimens.

Many of the fabrication techniques and hardware that were developed to meet these requirements have been described in detail elsewhere(1,2) for the controlled temperature capsule. This report describes the subsequent improvements that have been made to this capsule, the modifications made to permit the control of both temperature and pressure, and the present instrumentation employed.

DESCRIPTION OF CAPSULES

Controlled Temperature Capsule

The controlled temperature irradiation capsule (Figure 1) essentially consists of two concentric cylinders separated by heat transfer ribs. The outer cylinder which is constructed of 304 stainless steel is approximately 24 inches long with an inside diameter of 1.38 inches and an outside diameter of 1.50 inches. The inner cylinder is also constructed of 304 stainless steel and is approximately 14 inches long (including a 4 inch expansion chamber) with an 0.875 inch outside diameter and an 0.760 inch inside diameter. The outer cylinder serves to isolate both physically and thermally the inner cylinder from the reactor coolant. The inner cylinder contains the specimens, specimen holder, and thermocouples which are kept in good thermal contact by using NaK (a sodium-potassium alloy which is liquid at room temperature.) Expansion of the NaK is allowed by using an expansion chamber connected by a flexible siphon to the specimen chamber.

The specimen holder illustrated in Figure 1 was originally machined as a unit from a rod of Zircaloy-2. The present specimen holder which is designed to contain 6 split tubes (12 specimens) of uranium is fabricated from a series

of Zircaloy-2 segments, each of which accommodates a sample. Any of the segments can be easily modified to accommodate various geometries. The segments are either joined with a rod threaded axially through the center of the segments or with rods welded to the periphery of adjoining segments. These modifications provide complete flexibility within the space limitations of the inner cylinder. Thin foils of nonfissionable metals for transmission electron microscopy and tensile specimens of high purity iron have recently been included in this holder for other program purposes.

Temperature is maintained at a predetermined value for all reactor conditions by a heater wound around the outside diameter of the inner cylinder which is grooved to accept the heater. The 1/8 inch O. D. heater (3) which is sheathed with 304 ELC stainless steel consists of a grounded nichrome heating element with an integrally sheathed nickel lead wire which is welded to the nichrome and extends to the front face of the reactor. The heater is MgO insulated and is rated at > 5 KW. Specimen temperatures are monitored by six thermocouples, one of which is used for temperature control. The thermocouples (4) are type K (chromel-alumel), MgO insulated, 304L stainless steel sheathed, 1/16 inch O.D., and have a grounded thermal junction. Flux monitors consist of 0.020 inch diameter Al - 0.1% Co wires approximately one inch long mounted on the outer cylinder supports. After discharge of the capsule the wires are recovered and analyzed for total exposure.

More than thirty of the controlled temperature capsules have been irradiated during the past six years with no major in-reactor failures occurring. This is due primarily to the rigid specifications to which heaters and thermocouples are purchased (3,4) and to the rigorous quality control inspections

that are conducted during construction of the capsule. Each weld is tested to assure complete impermeability to helium and each weld is completely radiographed. The completed capsule is bench tested at its anticipated operating temperature in flowing water to assure integrity of all components prior to insertion in the reactor.

Controlled Temperature and Pressure Capsule

The controlled temperature and pressure capsule (Figure 2) is in essence a "cold wall" autoclave, that is, the only stressed component is the cold outer shell which becomes the pressure vessel. The pressurized capsules are primarily a modification of the controlled temperature capsules. The thickness of the outer shell was increased and the NaK siphon was replaced with a stainless steel bellows. The outer shell was changed to 316 stainless steel on the basis of better resistance to corrosion and better high temperature strength than 304 stainless steel. The siphon was replaced with a stainless steel bellows to allow application of hydrostatic pressure to the specimens and to allow for the differential thermal expansion between the NaK and stainless steel components. The outer chamber is pressurized with helium and the bellows transmits the pressure to the inner chamber and the specimens. No pressure differential exists between the inner and outer chambers and little or no gas void exists on the NaK side of the bellows. The bellows which have 64 convolutions, have the following specifications: 1.00 inch O. D. x 0.50 inch I.D. x 0.008 inch thick, extended length - 3.5 inch, free length - 1.8 inch, nested length - 0.9 inch.

The criteria for determining the thickness of the outer pressure shell were obtained from the ASME Boiler and Pressure Code (Section III, Nuclear Vessels, 1965 Edition). After determining the maximum operating temperatures and

pressures for a specific capsule, the thickness of the pressure tube wall was obtained by using a formula from I-1 of the Code.

$$t = \frac{PR}{S_m - .5P}$$

t = thickness of pressure tube wall, inches

P = internal pressure, psi

R = internal radius, inches

S_m = design stress intensity obtained from Table N-421 of the Code

Hemispherical-shaped heads were used to eliminate stress risers. The thickness of the heads was determined by the following formula obtained from Section I-120 of the Code:

$$t = \frac{PR}{2S_m - P}$$

NOTE: The same symbols apply to both equations.

Basic stress intensity limits which consider the effects of combined stresses are described in Sections N-414.1, N-414.4, and N-417.6. The limits were calculated by application of formulas located in Sections I-221 and I-222 of the Code. Experimental tests have proved the design to be more than adequate. For example, an elevated temperature burst test on a capsule designed for 2000 psi operation failed at about 12,000 psi. Tests on another capsule designed for 5000 psi operation showed complete integrity of the capsule when pressurized with helium to over 12,000 psi with the wall of the outer cylinder at approximately 300°C. By using a unique weld design (see Figure 3) on the two circumferential end closure welds, complete penetration is obtained without using a backing ring.⁽⁵⁾ This simplifies construction and eliminates a potential stress riser where the backing ring would contact the inner wall of the pressure tube.

The same detailed quality control methods were employed during the construction of the pressurized capsules as were used with the controlled temperature capsules. The bench testing includes pressurizing the capsule at the anticipated operating temperature to twice its intended operating pressure to assure complete integrity.

Seven controlled temperature-pressure capsules have been successfully irradiated to date over the temperature range 400-625°C. The controlled temperature-pressure capsules can be operated over the same temperature range (200-800°C) as the controlled temperature capsule.

INSTRUMENTATION

The value of the data obtained from postirradiation examinations depends on close control and careful recording of the irradiation test parameters. These parameters include temperature, pressure, exposure, and time. Time and exposure are recorded while temperature and pressure are controlled. Since specimen temperatures must be controlled at preset values for periods of up to one year, reliability of the instruments is of primary importance. In addition, any temperature overshoot must be avoided and the reactor must be completely protected against potential hazards.

Temperature in either capsule is controlled within $\pm 2^{\circ}\text{C}$ while the reactor is either up or down. However, during a reactor scram the temperature may drop 30-40°C. The capsules are normally brought back to the control temperatures by the heaters in 5-10 minutes. In some instances it is desired to control at a lower temperature during reactor down times. This is accomplished by manually changing the controller set point. Accuracy of the thermocouples has been certified as $\pm 4^{\circ}\text{F}$ ($\pm 2^{\circ}\text{C}$) from 32-530°F (0°-277°C) and 3/4% from 530°F -

1600°F (277-871°C). Pressure is controlled within $\pm 2\%$. The control and recording system is depicted in Figure 4 with a schematic representation illustrated in Figure 5. The temperature control system is completely solid state for maximum system reliability and versatility. A General Electric Measurement and Control system (GEMAC) has replaced the tube type control instrumentation and silicon controlled rectifiers (SCR's) have replaced the saturable core reactors formerly employed. (1)

The GEMAC system consists of a mV/I transmitter, a three-mode controller, and a strip chart recorder. A millivolt signal from the chromel-alumel thermocouple is converted by the transmitter into a proportional 10-50 milliamp signal which is sent to the controller and the recorder. The controller compares the milliamp signal with a setpoint in a potentiometric circuit to determine the required output to the SCR "gater" and the recorder logs the temperature on a strip chart. The SCR's are capable of conducting 110 amps RMS with a repetitive peak reverse voltage of 600 with no limitation in peak forward voltage. The SCR's are controlled by a unit called the "gater" which obtains its signal (10-50 ma) from the GEMAC controller. The timing of the three volt pulse from the "gater" is proportional to the GEMAC 10-50 ma signal and determines when and how much the SCR's will conduct (Table I). The SCR's vary the power to the heater by varying the percent of the A.C. sine curve that they are conducting. Figure 6 schematically portrays the action of the SCR's. Two SCR's are required, of course, one for the positive half and one for the negative half of the A.C. supply voltage.

Since a grounded heater is used in these capsules the 240 volt supply voltage is provided by an isolation transformer to prevent hazards to personnel.

TABLE I
SCR OUTPUT

Controller Output %	Controller Output ma	Heater Voltage (RMS)	Heater Current (Amps)	Heater Power Watts (5 ohm load)	SCR Output Shaded Area Is "ON" State
0	10	0	0	0
20	18	20	4	80
40	26	65	13.0	845
50	30	110	22.0	2420
60	34	160	32.0	5120
80	42	185	37.0	6830
100	50	200	40.0	8000

NOTE: The Above Table Applies to a Variac Setting of 240.
If the Variac is at a Fraction of 240, the Above
Results must be Multiplied by that Fraction.

Variable autotransformers (variatics) are placed in the circuit to provide the means to limit the maximum power to the capsules by limiting the voltage. The power requirements to the heater may vary from a few watts with the reactor on to several kilowatts with the reactor down. With the fast response control system employed variatics are sometimes necessary during a reactor scram to prevent a surge of power from burning out the connector which links the in-reactor power leads to the ex-reactor power leads.

One thermocouple from each capsule is monitored by two solid state, current measuring temperature comparators (Magsense units) made by Control Data Corporation. One is used for overtemperature and the other is used for undertemperature. Should an out-of-limits condition occur, the device trips and latches a relay which cuts the power to the capsule heater. If the power supply failure occurs or if the thermocouple opens or shorts, the heater will also be tripped out. The trip point adjustments are calibrated potentiometers with a range of 0-1000°C, designed to be used with 185 ohm thermocouples. Response time is less than 100 milli-seconds and repeatability is $\pm 5^{\circ}\text{C}$.

Heater power is monitored by solid state power transducers made by Scientific Columbus, Inc. The transducers utilize the Hall effect by means of a Hall element located in an air gap in a magnetic circuit. A potential transformer supplies a control current proportional to the line voltage to the Hall element. The necessary magnetic field is provided by a core excited by the line current and is proportional to this current. The output signal from the transducer is a d-c millivolt signal proportional to the product of the electric current and the input from the magnetic field. (6)

High pressure helium is supplied to the pressure control system by commercial cylinders for the low pressure (≥ 1000 psi) tests and by special order cylinders for the higher pressure (1000-5000 psi) tests. These special order cylinders are filled with a gas pressurization system supplied by Pressure Products Industries, Inc., and are capable of storing 1/4 cubic foot of helium at 15000 psi. The basic components of the pressure control system are a Marotta regulator and a Heise high pressure gage. The regulator is manually operated to obtain the correct reading on the high pressure gage and to maintain the pressure at the desired level. High pressure tubing designed to withstand 30,000 psi links the pressure supply system to the pressure tube from the capsule. The pressure in this tubing is monitored by a bonded strain gage transducer made by Dynisco. The basic design is a double-walled cylinder. The pressure medium admitted to the gap between the two concentric cylinders expands the outer cylinder and compresses the inner cylinder. Circumferentially-wound strain gages bonded to the inner wall of the inner cylinder and to the outer wall of the outer cylinder form a four-active-arm Wheatstone Bridge. The output which is a d-c millivolt signal proportional to pressure is recorded by the data logger.

Temperature, pressure, heater power, and time are logged by a Minneapolis Honeywell ADRT 3060 LX data handling system which provides data logging or visual digital display of 60 input variables. The system has provisions for automatic logging, manual or remote demand logging, automatic point selection, and manual point selection.

HEAT TRANSFER

Extreme flexibility in heat transfer must be provided due to the wide range

of test temperatures. Enough heat transfer must be supplied to prevent an overshoot of the desired control temperature when the reactor is at its maximum power level. Care must also be taken to avoid providing too much heat transfer since the requirements for heat input must not exceed that capable of being supplied by the heater when the reactor is "down". To further complicate the situation, changes in water pressure from periods of reactor "up" time to "down" time show a marked effect on heat transfer and serve to increase the power requirements of the heater. In addition, changes in fuel loading are often made which alter the flux and the total heat supplied by gamma and fission heating. The necessary flexibility in heat transfer is achieved by using 304 stainless steel ribs located between the outer shell and the specimen chamber. By varying the number and the size of the fins, any temperature within the operating limits of the capsule can be realized. Temperature control of capsules with an unusually large heat generation at low temperature is achieved by filling the space between the pressure shell and the specimen chamber with steel shot.

Some difficulties have been encountered due to the large increase in heat transfer with increased helium pressure. Laboratory tests conducted on capsules designed to operate at 2000 psi indicated an increase in heat transfer of approximately 10% per 1000 psi increase in helium pressure. Similar tests conducted with the prototypic 5000 psi capsule indicate a 20-25% increase in heat transfer per 1000 psi increase in helium pressure. (See Figure 7). The temperature of the inside wall of the outer cylinder also varies with helium pressure. At low pressures the temperature of the wall close to a fin is higher than a point midway between two fins. However, at higher pressures the midpoint of the wall will be at the higher temperature with the capsule at the same control temperature.

Pressure has little effect on heat transfer by conduction or radiation⁽⁷⁾, but appears to affect the convection process considerably. Holmes⁽⁸⁾ indicates that the increase in heat transfer with increasing pressure is due to a convection process. In his studies he used two concentric cylinders; one with an I.D. of 2 1/2 inches and the other having an O.D. of 1 inch. Holmes was able to determine empirical relationships between heat transfer and helium pressure over a temperature range of 15-540°C and a pressure range of 15-500 psi (1-34 atmospheres). He concluded that the process was by convection. Beckman⁽⁹⁾, who investigated heat transfer by convection between two coaxial cylindrical tubes using air, hydrogen, and carbon dioxide, found a relationship between pressure and convective heat transfer. Beckman also found that convective heat transfer was negligible in gas annuli less than 0.125 inch; however, his studies were limited to diatomic gases, a pressure range of 1/6 to 1.9 atmospheres, (2.5-28 psi) and a temperature range of 20-65°C. The irradiation capsules described in this paper have a gas annulus of approximately 0.095 inch. Apparently Beckman's restriction with regard to the size of the gas annulus is not applicable to the present capsules due to the use of higher pressures (2-400 atmospheres or 30-6000 psi) and monatomic helium.

CONCLUSIONS

Both the controlled temperature and the controlled temperature-pressure irradiation capsules have proved to be highly reliable due to the high integrity of components and the extreme care taken during construction. Few thermocouple or heater failures have occurred and not one capsule has had to be discharged ahead of schedule due to a malfunction. The reliable instrumentation has

provided excellent temperature and pressure control. Simple adjustments in the heat transfer fins allow selection of a wide range of operating temperatures.

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FIGURE 1.

Controlled Temperature Irradiation Capsule for Basic Swelling Studies.

Neg. No. 061079

FIGURE 2.

Controlled Temperature-Pressure Irradiation Capsule for Basic Swelling Studies.

Neg. No. 0630653

3.4.17

FIGURE 3
Weld Design

Figure 4 Temperature Control and Recording Instrumentation for Controlled Temperature-Pressure Irradiation Capsules.

AN ASSEMBLY FOR IRRADIATION OF MOLTEN FLUORIDE FUEL TO HIGH
BURNUPS IN THE OAK RIDGE RESEARCH REACTOR (ORR)

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As part of the Molten Salt Reactor Program at Oak Ridge National Laboratory, an irradiation assembly has been designed in which experiments will be conducted to study the effects of radiation and fissioning on molten fluoride fuel salt and construction materials. Irradiation experiments in this assembly are to be carried out in a horizontal beam hole facility in the Oak Ridge Research Reactor (ORR).

Objectives of these irradiation experiments are to simulate conditions expected to exist in an operating molten salt reactor.^{1,2} These conditions include a nominal operating temperature of 1200°F, fuel circulating through regions of different temperatures in contact with Hastelloy N piping and graphite, high power density and removal of gaseous fission products, and extended operating time; at least one year is desired. The extended operating time is expected to result in 50% burnup of fuel with an opportunity to study fission product chemistry at high concentrations.

The experiment assembly, shown in Figure 1, is designed to circulate fuel salt thermally through a graphite core at a power density of 200 w/cc, for more than a year, using a $\text{LiF-BeF}_2\text{-ZrF}_4\text{-}^{235}\text{UF}_4$ fuel salt containing about 1 mole % $^{235}\text{UF}_4$ (liquidus temperature of 840°F). The main body of the thermal loop is fabricated of 2 in. Sch-40 Hastelloy N pipe with a graphite core. The graphite core contains eight $\frac{1}{4}$ in. longitudinal holes, which serve as fuel passages and provide a fuel volume of $\sim 43 \text{ cm}^3$ which is exposed to the highest thermal neutron flux ($\sim 5 \times 10^{13} \text{ n/cm}^2 \text{ sec}$). A reservoir tank, through which the salt then flows,

* Speaker

** Research sponsored by the U. S. Atomic Energy Commission under contract with the Union Carbide Corporation.

Figure 5 Irradiation Capsule Control System

Figure 6 A C Phase Control Using SCR's, Conduction Angle may vary from 0° to 180° Allowing Control from 0 to 100% of Line Voltage

Figure 7 Temperature-Power-Pressure Relationship for a Controlled Temperature-Pressure Irradiation Capsule

serves as a salt and gas reservoir and provides for a liquid-vapor interface. A cold leg return line completes the circuit for salt flow.

Calrod heaters and cooling coils, in which air-water mixtures are used as coolant, provide for temperature control and maintain the temperature ranging between 1200°F (core section) and 1000°F around a convection loop for circulation. Some 12 kw of fission and gamma heat (8 kw fission heat) are expected during in-pile operation. Performance tests of the combined heater-cooler unit have demonstrated adequate capacity to remove the heat load while operating and to maintain the salt molten during periods of reactor shutdown.

The gas space in the reservoir can be purged and sampled as desired, and liquid fuel samples can be withdrawn and fuel replaced during in-pile operation. Sampling of the fuel salt and cover gas during in-pile operation is considered necessary to obtain information about the effects of fissioning and irradiation on the compatibility between salt, graphite, and Hastelloy N, fission product chemistry, salt stability, and fission power generation as irradiation and burnup increase. Such samples of the salt are removed through a sample line some 12 ft long, traced with electric heaters to maintain the salt molten and fluid. The sample line is connected to a sample station at the face of the reactor shielding where fuel salt samples can be routed either to a holdup tank for storage or to a sample capsule for transport to hot cell facilities and subsequent chemical analysis. The sample station also contains necessary tanks, valves, and lines so that mockup salt can be added to the thermal loop experiment to replace that removed in sampling or for enrichment during in-pile operation.

Operating experience to date with two prototype assemblies in an out-of-pile mockup, which duplicates the in-pile facility, has shown a salt circulation rate of $\sim 10 \text{ cm}^3/\text{min}$ with a temperature difference of $\sim 150^\circ\text{F}$ between the core section and cold leg. This flow rate is considered sufficient to provide the mixing needed to obtain representative salt samples and to study transport of fission products between hot and cold regions and to the gas-liquid interface in the reservoir. The

adequacy of the salt sampling system was demonstrated by removing some twenty samples (10 cm³ each) of salt while at operating temperature. After each sample removal, an equivalent amount of salt was added as required to maintain the original salt inventory.

Based on these tests, an irradiation assembly and associated auxiliary equipment has been fabricated and assembled and the first in-pile experiment is scheduled to begin in June 1966.

References

1. An experimental molten salt reactor, designed to generate 10 Mw of heat, is now undergoing test operation at ORNL.
2. W. R. Grimes, "Molten Salts as Reactor Materials," Nuclear News, May 1964, pp. 3-8.

In-Pile Molten Salt Thermal Convection Loop

An Irradiation Capsule for Molten Pu Alloys*

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To examine fission product distribution between a molten Pu fuel and Na in a reactor environment, a static capsule irradiation of the two molten phases in contact was considered the most feasible approach.

The primary conditions desired in the irradiation were:

1. Fuel-container interface temperatures must be limited to $\sim 700^{\circ}\text{C}$ to minimize container corrosion.
2. During reactor shut-down fuel temperature must be maintained at $> 450^{\circ}\text{C}$ to prevent melt-freeze cycling of the fuel alloy and possible distortion, or rupture, of the container.
3. Fuel-Na containers must have the capability of being removed from the reactor without disturbing the balance of the capsule assembly.

Since the irradiation was to be conducted in a primarily thermal neutron flux, and because of a high Pu density in the fuel, feasibility of the experiment was first determined by a power density calculation. Prior experiments in the in-core irradiation position to be used furnished data from which an unperturbed flux could be calculated using the W. B. Lewis semi-empirical method⁽¹⁾ coupled with an experience factor. For a Pu density of 5.0 g/cc the fuel was essentially a perfect absorber of thermal neutrons, so the average power density was calculated on the basis of all incident neutrons being captured. Fuel dimensions were 0.375 in. O.D. x 1.5 in. height.

*Work performed under the auspices of the United States Atomic Energy Commission.

The flux depression factor resulting from the neutron sink was taken from Lewis's work with the Materials Testing Reactor (MTR). Justification for believing this method would include the effects of the fast component of the flux was that the Lewis flux perturbations were determined experimentally for the MTR and included the effect of the fast neutrons. Further, the Omega West Reactor (OWR), in which this irradiation was to be done, is an MTR type reactor and should display a similar neutron spectrum.

The result of these calculations for a 49.3% Pu, 9.9% Co, 40.8% Ce alloy with 5.0 g Pu/cm³ was an average power density of 213 watts/g Pu, or 1065 watts/cm³, where it is assumed that 178 Mev/fission is removed by thermal conduction and convection. Subsequent events showed the accuracy of the estimate to be satisfactory. The number of fissions in the irradiation of a 5.0 g Pu/cm³ alloy was determined by radio-chemistry and resulted in 193 watts/g Pu, or 965 watts/cm³. Another analyses of a 5.7 g. Pu/cm³ alloy gave 134 watts/g Pu or 765 watts/cm³. The latter analysis was based on Ru only and is believed to be low due to volatilization of Ru in the sample dissolution. Thermocouple data during the irradiation indicate that the power density was somewhat greater than in the alloy of lesser Pu density. It must be remembered that the figures quoted are average values and that power densities near the fuel surface are probably higher by a factor the order of 10. Table 1 gives a summary of irradiation conditions for one capsule.

To accommodate the high heat load under nuclear heating (~610,000 Btu/hr-ft² at the fuel surface) and keep the container-fuel interface temperature < 700°C, thermal conductivity to the coolant water must be high. At the same time, to enable an electric heater of limited power density to maintain a fuel temperature > 450°C when the reactor is down, the thermal conductivity can not be too high. To resolve these conflicting requirements the capsule was designed as shown in Fig. 1.

Table 1

Summary of Irradiation Conditions*

Fuel Composition	49.3% Pu
	9.9% Co
	40.8% Ce
Pu, g	12.6
Pu, g/cm ³	5.0
Na, g	3.93
Irradiation temperature at Ta-fuel interface, °C	600
Watts/g Pu, avg.	193
Watts/cm ³ of fuel, avg.	965
Average heat flux at fuel surface, Btu/hr-ft ²	610,000
Average heat flux at capsule surface, Btu/hr-ft ²	47,000

*Power calculations are based on 178 Mev/fission being removed by thermal conduction and convection.

Figure 1. Schematic of molten Pu irradiation capsule.

The Pu fuel alloy is contained in a sealed Ta pin 0.375 in. I. D. x ~6 in. long with a 0.025 in. wall. The pin is centered in a Ta safety container, provided in the event of fuel leakage, which is in turn surrounded by a stainless steel sheathed electric heater. Seven stainless steel sheathed Cr-Al thermocouples are located between the pin and the safety container. The capsule volume is filled with Na as a thermal bond and has a Zr wire near the fuel pin for continual hot trap purification of the Na. A He purge over the surface of the Na is monitored to detect fission gases resulting from a Ta fuel pin leak.

The capsule proper is a double-walled stainless steel vessel with the annular volume between the walls evacuated and sealed in the fashion of a Dewar flask. This permits the electric heater in this section to give Na temperatures as high as 300°C at power levels as low as 100 watts. The temperature attained and the power level required are primarily a function of the difference between the O. D. of the inner wall and the I. D. of the outer wall. This difference is about 0.007 in. in the capsules tested. When the inner wall is hot enough to make contact with the cool outer wall, because of a differential thermal expansion, the thermal conductivity of the system is sharply increased and an increase of heater power levels to 3 or 4 kw results in a temperature increase of only 100-200°C. Figure 2 shows Na temperature as a function of heater power in typical capsules and Fig. 3 shows Na temperature as a function of fission heating by the fuel in Mk VI capsule.

The net effect of this design is that a fixed temperature differential across the contact surface of the two walls is built into the system and this differential can be attained at a very low heat flux. Calculation of this differential temperature is based on the coefficient of thermal expansion of the wall materials at varying temperatures, difference of vessel diameters to ± 0.0002 in., and conductance between metal-to-metal contacts as a function

Figure 3. Molten metal bath temperature as a function of nuclear heat generated in fuel for capsule Mk VI.

Figure 2. Molten metal bath temperature as a function of electrical heater power for several capsules.

of pressure. While these parameters for a given system cannot be obtained easily with any great accuracy, it was found that for differential temperatures of $\sim 300^{\circ}\text{C}$ experiments agreed with calculations to $\pm 15\%$ in the greatest error, and were usually better.

Sealing of the evacuated volume was accomplished by pinching off a 3/16 in. O.D. x 0.035 wall Type 347 stainless steel tube that had been flame annealed while the interior surface was under vacuum. He leak testing of such seals showed no leaks where the tube had been fully annealed. Prior to sealing, the evacuated volume was heated to 500°C and held at a pressure of 5×10^{-5} torr for ~ 96 hrs. to out-gas. Performance of this feature of the design was very satisfactory in out-of-pile testing. Temperature vs. power was reproducible and Na melt-freeze cycling showed no effect. However, in a 2 hr exposure to the radiation environment of the reactor the insulating value of the Dewar vessel was affected by radiolytically induced out-gassing of the stainless steel, as shown in Fig. 2. For complete insulation the vacuum must be kept < 0.03 torr and this pressure was apparently exceeded. Whether prolonged irradiations would result in significantly greater effects is not known, but in the short irradiations reported here the rate of change became negligible.

Because of better heat transfer to the coolant water than anticipated, and a less than desired diameter differential between walls, heater powers larger than tested out-of-pile were required to maintain the fuel in a molten condition when the reactor was shut-down. Irradiations were therefore limited to one reactor cycle, or less.

The heaters used consisted of a single 0.020 in. diameter Nichrome wire resistor insulated by ~ 0.020 in. of MgO within a stainless steel sheath of ~ 0.017 in. wall thickness. A continuation of the same material, but larger in all radial dimensions by a factor of 2, extended from the coiled section up and out of the reactor core to copper leads. The coil was 43 in. in developed

length and generated 80% of the heat. The larger diameter extensions were each 20 in. long and, with up to 1.1 kw of the power generated in them, required good thermal contact with the water cooled wall of the assembly. This was accomplished by filling the annular space in which they ran with Al granules. Maximum heat flux at the heater surface was 385 watts/in.² at a voltage of ~210 V. No failures of this style heater were encountered as opposed to many difficulties with heaters in which two resistor wires were contained in a single sheath. The greatest, but not only, problem with the latter design was making reliable connections to the power leads. In one case in out-of-pile testing, a short circuit developed from the heater - power lead connector to the capsule wall. The resultant arc melted a 1/32 in. diameter hole through the 0.070 in. thick wall, permitting coolant water to enter the annular volume of the capsule. To prevent a recurrence, heaters were subsequently powered by isolation transformers with ground indicators.

Removal of the Ta fuel pin was accomplished by withdrawal into a shielded transfer lock (Fig. 1) at the reactor top using the 1/2 in. tube to which the pin was attached for handling. The lock consisted of a 3/4 in. tube with a Wilson seal at the top, through which the 1/2 in. tube passed, and a ball valve at the bottom. A second ball valve at the top of the irradiation assembly was closed after pin removal to protect the integrity of the capsule atmosphere. The transfer lock was then unscrewed from the irradiation assembly and removed. Introduction of a pin was the reverse of this procedure with a vacuum - He manifold provided for first inerting the lock atmosphere.

In operation it was found that the 100 cm³/min He purge, brought to the top of the reactor from the Na bath in 0.070 in. O.D. hypodermic tubing, read ~2 R/hr at contact with the tubing. The source of the activity was Ne-23 with a 38 sec. half-life produced by a Na-23 (n,p) reaction. Reduction in the purge rate to a few cm³ min and setting the counter discriminator to look for Xe-138, reduced this back-ground sufficiently to maintain a margin for detection of leakage from the fuel pin.

During nuclear heating, when heat flux and temperature gradients were greatest, a thermocouple near the fuel pin surface showed a variation between print-outs on a six-point recorder of 20°C . This was interpreted as being an indication of strong convective currents. Temperatures were also very sensitive to distance of the thermocouple from the pin surface since the radial thermal gradient in the Na near the pin was calculated to be as much as $0.8^{\circ}\text{C}/0.001\text{ in.}$ Temperature differences of $\sim 150^{\circ}\text{C}$ from the mid-height to the top of the fuel ($\sim 0.75\text{ in.}$) were also noted between thermocouples in comparable radial locations.

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ADVANCES IN NaK ENCAPSULATION TECHNIQUES
AT THE SAVANNAH RIVER LABORATORY*

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SUMMARY

The Savannah River Laboratory (SRL) NaK capsule irradiation program has emphasized the study of irradiation growth and cavitation swelling in uranium and uranium alloys.

Capsule design has remained simple because of the nature of the experiments, which involve the comparison of large numbers of different specimens under equivalent irradiation conditions.

A number of advances have been made in NaK encapsulation techniques. The techniques developed by Kittel's group at ANL have been modified to increase the safety and capacity of the encapsulation facility and to decrease the time and manpower required for encapsulation. In particular the capsule loading chamber has been modified to double the number of capsules that can be loaded at one time. A new method has been developed for sealing the capsules. The vacuum system has been modified so

*The information contained in this article was developed during the course of work under contract AT(07-2)-1 with the U. S. Atomic Energy Commission.

that the ultimate vacuum is improved, and the "pump-down" time reduced by a factor of ten. Still other modifications have been introduced that simplify and increase the safety during the mounting, filling, and sealing of the capsules and the cleaning of NaK from the system after loading.*

Advances have also been made in test specimen design. Techniques have been developed for strain-free machining of "spherical" specimens and for directionally indexing them with rectangular holes. The spherical specimen greatly reduces the necessary number of test specimens for irradiation growth experiments, since data for growth in all directions can be obtained from a single specimen.

INTRODUCTION

NaK-containing capsules have been utilized at SRL for basic studies of the irradiation growth and swelling of uranium and uranium alloys. Simple and inexpensive capsule designs were used because of reactor conditions and the nature of the experiments. The experiments took advantage of the relatively large regions of stable flux in the Savannah River reactors by providing comparisons of large numbers of different specimens under similar irradiation conditions. Elevated temperatures required for swelling studies were obtained by self heating of the specimens within thick-walled capsules, and the experiments were designed so that temperature gradients over thick specimens could be tolerated. Each specimen is irradiated in a separate capsule so that only small amounts of NaK are required in each capsule thereby reducing the danger of release of NaK into the reactor in the event of a capsule failure.

The irradiation of large numbers of capsules required that efficient and inexpensive encapsulation techniques be developed. The present report describes a number of advances that have been

*To date, over 600 capsules have been loaded and irradiated without a failure.

made in encapsulation, beginning with the techniques developed by Kittel's group at Argonne National Laboratory (ANL).

A discussion is also given of the machining, preliminary testing and design advantages of spherical specimens for irradiation growth studies.

DISCUSSION

ANL NaK Encapsulation Facility

The encapsulation facility first employed at SRL was patterned after that built by Kittel and Tedeschi⁽¹⁾ of ANL.

The ANL capsule is threaded at both ends. One end has a large hole for inserting the sample, and the other end has a conical valve seat, small enough to permit large pressures to be exerted when screwing in the top plug, yet large enough to admit NaK freely.

The ANL encapsulation facility has a capacity for filling eleven capsules in each loading. The encapsulation procedure is as follows:

- a) insert the specimens into the large end of the capsule then screw in and "Heliarc"* weld the end cap,
- b) place the capsules in the vacuum chamber on a rotating holder the shaft of which extends outside through a Wilson seal, and
- c) evacuate the chamber then rotate each capsule under the NaK containing burette (the burette extends into the chamber through a Wilson seal) and fill with NaK. If desired, helium may be added up to atmospheric pressure.
- d) After filling the capsules rotate the capsule holder so that each plug (held above the capsules in Wilson seals in the chamber lid) is directly over a capsule. Push each top plug down through the Wilson seal and screw it tightly into the capsule to make a mechanical seal.

* Trademark of Union Carbide Corp.

- e) Remove the capsules from the vacuum chamber and leak test by inverting the capsules for a few hours in an evacuated bell jar. If the mechanical seal is good, saw off the excess plug length and "Heliarc" weld. The welded capsule is tested by autoclaving in steam for 24 hours at 125 psi and 178°C.

Modifications of the ANL Encapsulation Facility

Figure 1(a) shows the improved SRL NaK encapsulation facility and identifies components in the system. A schematic of the system is shown in Figure 1(b). Equipment used in construction of the system is listed in Table 1. The modifications are described below.

Increased Capsule Filling Capacity

The ANL vacuum chamber had a capacity for filling⁽²⁾ and sealing eleven capsules. In the design that was copied from ANL, the capacity was increased to nineteen capsules by increasing the physical size of the loading chamber as shown in Figure 2(a). The ultimate vacuum attainable was increased and the pump-down time was decreased by using a larger capacity fore pump and diffusion pump and by replacing, where practical, the rubber tubing used in flexible connections by metal bellows.

Relocation of the Capsule Top Plugs

The vacuum chamber has to be sealed off from the pumps during filling hence it must have a low leak rate to prevent it from being contaminated with atmospheric air. The chamber lid from the ANL design has many Wilson seals that may leak under static conditions. There is one for the main shaft of the capsule holder, one for the NaK filling burette and nineteen for the capsule top plugs. The top plugs are long cylindrical stems, Figure 3(a), with pointed conical tips on one end and a hexagonal head (for application of a wrench) on the other end. The stems feed through a Wilson seal in the chamber lid. The stems must be long, must have a good surface finish, and must be coated with vacuum grease along the section that passes through the Wilson seal.

The disadvantages of having the top plugs mounted in the chamber lid are as follows:

- a) The high leak rate of atmospheric air into the chamber contaminates the NaK, especially for the capsules that are sealed last.
- b) The top plugs are expensive to make.
- c) The vacuum grease creates a possible explosion hazard⁽³⁾ if in contact with oxides of sodium and potassium. The latter are usually formed during disassembling and cleaning a batch of capsules.
- d) If not watched carefully, the plugs suck through the Wilson seals into the chamber and break the vacuum.

These four disadvantages were eliminated by relocating the top plugs on a movable carrier inside the chamber. The carrier, Figure 2(b), is mounted on the same shaft that holds the capsules. The plugs are manipulated and sealed by a socket wrench which enters the chamber lid through a single Wilson seal. Hence the chamber lid has only three Wilson seals instead of twenty one, the plugs can be shortened, and they require no surface finish or vacuum grease. The reduced number of Wilson seals, together with the high capacity pumps and metal bellows, shortens the pump-down time by a factor of ten.

Redesign of the Capsule Top Plug

The capsule top plug, Figure 3(a), has a pointed conical tip that is screw driven into the capsule plug seat, Figure 3(b), after charging with NaK. If any misalignment occurs between the plug and the capsule as the plug is screwed into the capsule, the plug touches one section of the plug seat first, galls, and will not seal. Hence to get a high yield on these mechanical seals, it was necessary to machine each top plug to a close tolerance and "mate it" to a capsule. This also resulted in considerable bookkeeping and separate handling during preparation and loading to make sure that each capsule was matched with the top plug that was machined for it.

The top plug was redesigned as shown on the right in Figure 3(a). The top of the plug is self centering to allow for misalignment, and is made of annealed nickel so that under pressure it deforms readily to the contour of the plug seat. The arrangement of the parts after sealing is shown in Figure 3(b). With the two-piece top plug the yield of good mechanical seals is usually 100 percent.

Positioning Capsules for Filling with NaK

In the original design each capsule had to be held in position during filling by applying a counter torque (with a wrench) to the rotating stage. An automatic positioning arrangement, Figure 2(b), has been installed which makes it possible for one man to operate the facility instead of two.

The manner in which the capsule is mounted on the rotating stage has also been improved. Previously the capsule was held in a cylindrical seat by three radial set screws. The new rotating stage employs a set of keys which fit into a slot machined in each capsule thereby eliminating the time-consuming operation of tightening 3 screws on each of 19 capsules.

The new center shaft that holds the rotating capsule and top plug stages is demountable. Originally the chamber lid, center shaft, top plugs, rotating stage and capsules (approximately 50 pounds) all had to be inserted or removed together creating a considerable danger that parts might be broken or NaK spilled. The center shaft can now be left in position while the chamber lid, glass chamber wall, and rotating stage can be inserted or removed individually, reducing the weight to be handled to less than ten pounds.

Modifications to Burette and Filling Tip

The objections to the ANL burette assembly were:

- a) There were places where NaK could collect without being seen.

- b) Some connections had to be taken apart and remade each time the system was used.
- c) The glass tip of the burette was delicate and easily broken during a loading, in which case no more capsules in that batch could be loaded.
- d) The place where the burette went through the top of the chamber was easily broken, in which case NaK could be spilled.

The burette assembly was redesigned, as shown in Figure 2(b).

The merits of the new design are:

- a) There are no places in the system where NaK can collect unnoticed because the entire system is made of glass and can be disassembled easily at tapered joints, and steam autoclaved to guarantee the removal of all traces of NaK.
- b) Several burette assemblies and connections to the NaK tank can be used in rotation, which reduces the time between loadings.
- c) A "Teflon"* tip extension was put on the burette tip which eliminates tip breakage.
- d) A ball and socket joint between the burette and the tip reduces the possibility of a break occurring at that point.

Miscellaneous

A new NaK tank was also constructed. It is similar to those supplied by the NaK manufacturer, except it was thoroughly cleaned before being filled with NaK and is equipped with precision valves. NaK is stored in and pumped from this tank with a minimum of contamination.

To reduce the pumping impedance in the system, glass tubes of larger diameters were used and the system length reduced. As mentioned before, metal bellows replace rubber tubing where practical.

* Du Pont trademark for its fluorocarbon plastic.

Spherical Irradiation Test Specimens

From the point of view of heat transfer, uniform stress distribution, and isotropy of self restraint, the spherical specimen is ideal for basic irradiation studies of materials. However spherical specimens are expensive and time-consuming to machine so as to preserve a reference to the previous fabrication directions.

In some cases, e.g., basic irradiation growth studies, the expense and time is justified for a number of reasons:

- a) Ease of interpretation of results - Irradiation growth occurs without a change in volume and is, to a very close approximation, a second order tensor type deformation.⁽⁴⁾
Hence, a spherical specimen deforms under irradiation into an ellipsoid, which may be specified completely by the lengths of its principal axes and their orientation with respect to the fabrication directions.
- b) Isotropic shape - Growth is very easily restrained. In one case,⁽²⁾ a uranium cylinder in a NaK capsule was observed to grow around a low-yield-stress spiral spring, made of 20-mil stainless steel wire, rather than to push it open. It is frequently observed that uranium will grow up to the wall of a capsule then turn around and grow in another direction. Hence in a growth experiment the specimen should be as isotropic as possible to prevent anisotropic self restraint due to the shape of the specimen.
- c) Reduction in the required number of irradiation test specimens - In measuring the effect of a given type of preferred orientation on growth one usually selects samples from about ten directions including the principal direction of growth and shrinkage. If one is, in addition, studying the effect of alloying on temperature, the number of samples is considerably larger, e.g., three alloys at three temperatures require about 90 specimens, whereas

the same information could be obtained from only nine spherical specimens.

The machining and referencing problems associated with spheres may be solved by employing spark-erosion machining techniques as shown in Figure 4(a). Since rectangular reference holes can be easily machined by these methods, the edges of the hole can be made to coincide with the fabrication directions.

The spheres, Figure 4(b), may be conveniently mounted by threading and suspending them from a steel rod in the capsule. Alternatively they may be suspended from wires, a few mils in diameter. Holes on the order of 0.005 inch are easy to drill in these small spheres.

An irradiation test was performed to see if cylindrical uranium specimens threaded and suspended from steel rods, Figure 5, would have the same growth rate as companion specimens positioned by spiral springs made of 20-mil wire. The growth rates were the same for both positioning schemes. Hence it is believed that the spheres, which are about the same size as the small cylinders, will not be restrained if suspended from threaded rods.

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3. Liquid Metals Handbook, Sodium (NaK) Supplement. 3rd ed., p 386, Government Printing Office, Washington, D. C. (1955).
4. Sturcken, E. F. and Croach, J. W. "Tensor Properties of Oriented Metals" USAEC Report DP-927 (October 1964).

TABLE 1.

NaK Equipment for Encapsulation Facility

1. Vacuum pump, Welch "Duo-Seal"*, Model Number 1405-H, pumping speed 33.4 liters per minute, resulting pressure 5×10^{-5} mm.
2. Bellows stainless steel OD $\frac{13}{16}$ in., ID $\frac{1}{2}$ in. Robertshaw-Fulton Co.
3. Cold trap, cold-finger type, outside tube dimensions 65 mm OD 76 cm length, inside tube dimensions 45 mm OD 15 cm length.
4. Stopcock, high vacuum, 15 mm bore, offset type, Corning No. 7546.
5. Glass manifold ID 25 mm, length 1m.
6. Vacuum gage, thermocouple type 501, National Research Corp.
7. Gage, pressure, Vacuum -30 to +30 (USG).
8. Stopcock, high vacuum, 8 mm oblique bore, Corning Number 75, 42.
9. Stopcock, high vacuum, 2 mm oblique bore, Corning Number 75, 42.
10. Helium supply line.
11. Trap, charcoal type, outside tube, 45 mm OD x 8 in. long; inner tube and outlets, 12 mm OD tubing.
12. Manometer, closed end, 10 mm standard wall, length 89 cm, legs 37 mm apart on center, closed end leg has a constriction at "U" bend to approximately 3 mm. Filled by vacuum distillation.
13. Rubber hose - red - $\frac{3}{16}$ in. ID, $\frac{3}{16}$ in. wall, Fisher Scientific Company.
14. Burette, 20 ml, volume graduated in 0.1 ml. Three-way, 2 mm oblique-bore stopcock at bottom for filling.
15. High vacuum ball valve, National Research Corp. Number 1213, $\frac{1}{4}$ in. All stainless construction required.
16. Tank, stainless steel, 8 in. diameter x 8 in. high.
17. Glass envelope 12 in. OD x 14 in. high. Made from two Corning jars, Number 6942, sealed together in middle.
18. Trap, splash type, 38 mm diameter, has 10 mm OD stand pipe inside, 20 mm high.
19. Valves, High vacuum angle valve, "Veeco"*** FL Type, Number R625.
20. Diffusion pump, "Veeco" type EP2, Vacuum-Electronic Corp. Water cooled, 3 stage, metal oil diffusion pump.
21. Vacuum pump, Welch "Duo-Seal" Mechanical Vacuum pump, Model No. 1402B.
22. Tapered joints 30/10 Corning Number 6560.
23. "Kovar"†-to-glass seal, $\frac{3}{4}$ in. OD, Labglass, Inc., Number LG10850.
24. Metal parts in system.
25. Glass cylinder, OD 32 mm, length 90 m.
26. Ion gage, VG-1A "Cenco"†† #94175.
27. Thermocouple and Emission Regulated Ion Gage Control, Type 710, National Research Corp.
28. Vent.
29. Rubber hose, 1.4 in. OD wall, $\frac{7}{16}$ in. ID wall, Fisher Scientific Company.

* Trademark of The Welch Scientific Co.

** Trademark of Vacuum-Electronics Corp.

† Trademark of The Carborundum Co.

†† Trademark of Central Scientific Company

- A. High-vacuum capsule chamber
- B. Filling burette
- C. Pumping manifold
- D. NaK transfer lines
(NaK is forced into the filling burette by differential helium pressure)
- E. Mercury manometer to measure partial pressure of helium in capsule chamber
- F. NaK reservoir tank
- G. Roughing pump to evacuate helium purification train and NaK transfer lines.
The roughing pump and diffusion pump for the capsule chamber are below the table

FIG. 1a SRL NaK ENCAPSULATION FACILITY

FIG. 1b NaK ENCAPSULATION FACILITY (Schematic)

a. Original Chamber

- A. Rotating capsule stage
- B. Capsule
- C. Center shaft
- D. Rotating top plug stage

b. Modified Chamber

- E. Top plugs
- F. Positioning key
- G. Ball and socket joint
- H. "Teflon" tip

FIG. 2 COMPARISON OF THE ORIGINAL AND MODIFIED NaK FILLING CHAMBER

a.

- A. 1/4-inch-diameter stainless steel rods
- B. Annealed nickel tip

b.

- A. Top plug
- B. Annealed nickel tip
- C. Top section of capsule
- D. Nickel tip, deformed on capsule seat

FIG. 3 ORIGINAL AND MODIFIED TOP PLUGS FOR NaK Capsules

a. Schematic Showing how Sphere is Machined

b. Uranium Sphere

FIG. 4 SPARK EROSION MACHINING OF SPHERICAL SPECIMENS

DESIGN, ASSEMBLY, AND OPERATION OF URANIUM ZIRCONIUM-HYDRIDE FUEL IRRADIATIONS

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Atomics International

INTRODUCTION

The SNAP 8 Program has been in existence for a number of years. During this time, a major facet of the effort in support of this program has been the in-pile irradiation of many experiments. Specifically, the topic discussed here will be the latest experiment which has completed the cycle of analysis, design, fabrication, assembly, operation, and postirradiation examination. This particular experiment, which has been designated NAA 115-1, also represents a new generation of experiments, with regard to type and complexity. In addition to program requirements demanding this degree of sophistication, it is important to mention that it has been possible to satisfy these requirements only because of the availability of large reactors, in particular the Hanford-type reactor, as shown in Figure 1.

A general idea of the size of an assembly that can be irradiated in reactors of this type is indicated by Figure 2, which shows the NAA 115-1 Experiment being inserted into the reactor. Depending on the power requirements of the given experiment, the loading may be from front-to-rear in the reflector region (as shown), front-to-rear in the fueled core region, or side-to-side in the fueled region.

NAA 115-1 IRRADIATION ASSEMBLY

Basically, the NAA 115-1 in-pile assembly consists of six separate experiments in series, each having independent instrumentation and control systems. Two of the experiments have four sublength fuel elements, while each of the other four experiments have one sublength fuel element, for a total of 12 specimens. The number of experiments (in this case, six) is determined by the number of operating temperatures of interest. The number of specimens in each experiment is determined by the number of variables which are to be investigated at that particular operating temperature. Figure 3 shows why this particular experimental approach was chosen for investigating the three-dimensional matrix of variables which must be accommodated, in order to optimize the particular

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In previous work⁽²⁾ the specimen stands in the capsule and is positioned by a spiral spring of 20 mil stainless steel wire

FIG. 5 SUSPENSION MOUNTING SCHEME FOR NaK CAPSULE SPECIMENS

fuel design in question. The dots shown in the figure are the fuel compositions being investigated. It follows that this matrix of variables has required a statistical approach to the problem. This, together with the development of the unique instrumentation and control system, the excellent in-pile performance, and the wealth of pertinent data that the experiments provided the program, gave rise to the earlier statement that NAA 115-1 was the first of a new generation of irradiation experiments. Two additional closely related assemblies are currently operating at Hanford, and others are scheduled for insertion in FY 67.

TEMPERATURE CONTROL

Each of the six experiments are independently controlled to achieve the desired temperature, and basically operate as shown in Figure 4. Details of the control and instrumentation systems are covered in a paper to be given at this meeting, entitled "Instrumentation and Data Reduction Techniques for Irradiation Experiments." The binary gas system is adjusted from separate gas modules, and is hand regulated for the desired temperature after the reactor has reached power. The data acquisition and temperature control systems used for these irradiation assemblies are shown in Figure 5. The left group of consoles are from NAA 115-1; while the right group, a new solid-state revision, operates the NAA 115-2 Assembly.

FLUX DEPRESSORS

In addition to temperature control, flux depressors are designed and used to achieve the desired burnup rate. A typical approach is shown in Figure 6. It should be pointed out, however, that the flux depressors for the NAA 115-1 were all of the same thickness. The figure shows the design used for the NAA 115-2 Assembly, currently in operation at Hanford. Although the flux is relatively flat over the entire assembly, it was found, during the operation of the NAA 115-1 Experiment, that greater precision in the burnup rate could be achieved by sizing the flux depressor as shown in Figure 6 for subsequent experiments of this type (assuming the flux to be flat over each experiment within the assembly).

Figure 7 shows a typical flux depressor (shroud tube), together with several of the components which make up a given experiment. The figure shows the components used for an experiment containing four sublength fuel elements.

The "outer sleeve," when slipped over the "fuel sleeves," forms the gas annulus for the temperature control.

FABRICATION

Figure 8 shows the four sublength fuel specimens after having been instrumented with three or more thermocouples and dosimeters, prior to assembly of the components shown in Figure 7. Typical experiments, after assembly and NaK loading, are shown in Figure 9 without the outer sleeve or flux depressor being installed. The experiments shown are for the NAA 117-1 Assembly, and differ only in length from four of the six experiments used in the NAA 115-1 Assembly.

Figure 10 depicts the various stages of assembly of several experiments. Those shown in the figure represent the NAA 115-1 and the NAA 117-1 Assemblies. The NAA 115-1 Assembly is shown in Figure 11, just prior to shipment. Also shown is the 120-ft cable, carrying the instrumentation leads and gas lines for temperature monitoring and control. This instrumentation cable is connected to the instrumentation system shown in Figure 5.

PERFORMANCE

Probably the most outstanding statement that can be made at this point is that the NAA 115-1 Assembly performed extremely well throughout its operating period. During the postirradiation examination, only 1 of the 12 specimens irradiated in this test was observed to have failed. It is felt that the failure (cladding rupture) was due to thermocouple failures at the point where the thermocouples were brazed to form the NaK seal. Low-temperature readings at this point called for the reactor operator to admit a gas mixture to the experiment which would raise the fuel temperature. Not realizing that the thermocouples were in error, the operator increased the fuel temperature; and thus the pressure became excessive, and cladding rupture resulted.

ADVANCED TECHNIQUES

Quality Assurance

Aside from the extensive and sophisticated analytical and design effort that was involved, due to this experiment being the first of its type, the Quality

Assurance Program during assembly and pre-irradiation documentation was formidable. For example, one group of equipment, used to assure quality throughout the assembly of the experiment, is shown in Figure 12. This equipment consists of a search-ray system for examining small parts, and includes a closed-circuit TV system which employs a Vidicon camera, enlarging the image approximately 25 times and projecting onto a 17-in. TV screen. Also shown is a tape recorder which is used to document what has been viewed on the TV screen. This equipment was extensively used on small welds and, in particular, thermocouple junctions.

An example of pre-irradiation instrumentation is shown in Figure 13, which shows the profilometer, together with the digital readout system and X-Y recorder, used for precision measurement of fuel diameters before and after assembly into the cladding. Equally important, in this regard, are the measurements made after irradiation. Figure 14 shows an in-cell profilometer that duplicates the measurements which were made prior to assembly, in order that detailed dimensional characterization can be established during postirradiation evaluation. Details of this and other pre- and postirradiation measurements are covered in a paper given at this meeting, entitled "Assessment and Correlation of Errors in Pre- and Post-Irradiation Measurements."

Thermocouple Development

Since thermocouple performance in irradiation experiments has historically been a problem, from the standpoint of life expectancy, a great deal of the success of the NAA 115-1 Assembly is attributed to the prior work done on extending the life of thermocouples that were used in the experiment. A summary of this work (shown in Figure 15), which was done by E. L. Babbe, was presented at the October 1963 meeting. A report, NAA-SR-10511, discusses this work in greater detail. Basically, the data shows increased thermocouple life with increased insulation pack density. Insulation pack density is determined by dye penetrant times through short lengths of thermocouple. A graphic example, using these test results, is shown in Figure 16. The thermocouples used in the NAA 115-1 Assembly are listed, together with dye penetrant times, on 1/2-in. samples of the thermocouples used. Of significance, in this figure, is the order of failure for the respective thermocouples. It can be seen that a definite correlation exists between early failure of thermocouples and lower dye

penetrant times. As a result of these data, subsequent tests use only thermocouples which have dye penetrant time of 190 min or greater for 1/2-in. samples of thermocouple. Experiments now in operation have thermocouple histories as follows:

Test	Insulation Pack Density	Dye Penetrant Measurement	Number of TC's	Operating Time	Number of Failures
NAA 115-2	High	Yes	46	Several months	None
NAA 117-1	High	Yes*	48	Greater than one year	3
NAA 82-1	High	No	16	17 months	None

*No minimum dye penetrant time was specified for thermocouples used in this experiment.

Pressure Measurement

In addition to the thermocouple work described, other instrumentation techniques have been developed which are presently being employed in other irradiation assemblies of this type, or will be incorporated in future ones. A brief description of some of these follows. A pressure sensor, using all-welded construction, is shown in Figure 17. The pressure sensor uses the null-balance system of detecting pressure changes in irradiated fuel elements during operation. The sensor is also used to detect failures during stress-rupture tests of cladding as well as the endurance limits of fuel elements. The device uses an all-welded double seal against fission gas, and operates as shown in Figure 18. The pressure to be measured (bottom) is balanced by the helium reference pressure (top), and the coupling disc moves to make or break contact with the thermocouple junction contact. The helium pressure is read as the contact is broken, and thus the fission gas pressure is known. More detail on the sensor is given on page 55 of the Transactions of the June 1964 Meeting of the American Nuclear Society. During inspection of the sensor, the search-ray system (mentioned earlier) is used, as shown in Figure 19, which presents an x-ray picture of the device as seen on the TV screen. Dynamic viewing of moving parts, using this system, has been a definite aid in assuring proper operation of precision components.

Flux Monitors

Figure 20 shows a flux monitor (self-powered type) currently being tested for irradiation experiments. Discussion of this type of flux monitor is given on page 579 of the Transactions of the November 1965 Meeting of the American Nuclear Society. Flux monitors of identical geometry are currently being used in the NAA 115-2 Irradiation Assembly. The materials used, however, are beta emitters (e.g., silver, rhodium, etc.), and replace the 0.040-in. stainless steel shown in the figure. These monitors have proven extremely valuable in separating nuclear effects from thermal effects, and have helped in evaluating the operating characteristics of both the reactor and the experiment.

Gas System Control

In the area of temperature control, techniques are being developed which will permit automatic regulation of the binary gas system (as opposed to hand adjusted, as mentioned earlier), and will permit constant temperature to be maintained under varying flux conditions.

Figure 21 shows an actual chart, using this system with simulated power input. Temperature variation within $\pm 10^\circ$ was maintained while power adjustments, ranging from 300 to 900 w, were made on the sample.

Flux and Temperature Shaping

Finally, some of the more sophisticated analytical techniques, involving shaped axial flux profiles together with shaped axial temperature profiles, are illustrated in Figure 22. The particular technique shown is used with full-size fuel elements, to simulate actual operating conditions. Greater detail was given in a paper presented at this meeting, entitled "Digital Computer Techniques for the Heat Transfer Analysis of Radiation Assemblies." Successful radial flux shaping has also been employed where flux gradients in experimental positions were excessive (e.g., at the edge of the reflector region) and symmetrical heat generation was not possible. A technique to correct this type of situation is shown in Figure 23. This technique was used in the NAA 82-1 Experiment. Figure 24 shows the actual hardware (eccentric flux depressor) used with the experiment. The precision achieved in this case was exceptional, and is explained in depth in a paper given at this meeting, entitled "Thermal Flux Calculations for Experiments in the MTR."

SUMMARY

The NAA 115-1 Assembly, throughout its operation and evaluation, demonstrated the validity of the concept of integrating the control of more variables than had been accomplished heretofore. The sophisticated approaches used in analysis, design, instrumentation, fabrication, assembly, operation, examination, and evaluation have paved the way for a third generation of experiments which are now in the design stage. These experiments will be aimed toward the automatic control of temperature and burnup profiles, including the spectral effects of the radiation environment. They will use digital data acquisition and data processing systems to provide real-time display and control signals for operation, as well as for complete chronological documented histories of operating performance.

00-101412

Figure 1. Front Face of Hanford Reactor

7568-51403

Figure 2. NAA 115-1 Assembly Being Inserted in Reactor

7568-0452

Figure 3. Variables Investigated
in NAA 115-1 Irradiation

THERMAL CONDUCTIVITY

<u>GAS</u>	<u>@ 1300°F (Btu/hr - ft°F)</u>
HELIUM	0.196
ARGON	0.025

7568-15124

Figure 4. Temperature Control Concept

3.8.10

7568-18757CN

Figure 5. Data Acquisition and Temperature Control Consoles

7568-15199

Figure 6. Flux Depressor Concept

7568-01119

Figure 7. Components of NAA 115-1 Assembly

Figure 8. Instrumented Fuel Specimens, Prior to Fabrication

7568-51244

Figure 9. Typical Experiments, Prior to Addition of Outer Sleeve or Flux Depressor

7568-51355

7568-51351CN

Figure 10. Assembly Stages for NAA 115-1 and NAA 117-1 Experiments

7568-51276
Figure 11. NAA 115-1 Irradiation Assembly, Prior to Shipment

7623-1879

Figure 12. Search Ray System for Examination of Small Parts

00-18412

Figure 13. Profilometer System for Pre-irradiation Diameter Measurements

7568-15202

Figure 14. In-Cell Profilometer for Postirradiation Measurements

7568-15196

Figure 15. Thermocouple Failure Trend

INCONEL SHEATH				304 SS SHEATH			
TC NUMBER	DYE TIME (min)	ORDER OF FAILURE	TIME TO FAILURE (month)	TC NUMBER	DYE TIME (min)	ORDER OF FAILURE	TIME TO FAILURE (month)
29	*	1	2	6	259		
14 SS	*			3	337		
33 SS	*	5	7	30	393		
31	112	3	4	21	419		
25	118	2	2	9	420		
34	120	4	6	11	430		
10	120			28	448	7	12
22	125			8	456		
16	127			12	480		
13	137			35	490		
1	160			19	>500		
20	172			36			
7	178			24			
4	182			15			
				32			
				27		8	12
				26		6	7
				23			
				18			
				17			
				5			
				2	>500		

*NO DYE PENETRANT TIME AVAILABLE

7568-15200

Figure 16. NAA 11501 Thermocouple History

Figure 17. Adit Pressure Sensor

00-101197

Figure 18. Schematic of Pressure Sensor

7568-15197

Figure 19. Search Ray Examination of Pressure Sensor

00-101196

7568-15195

Figure 20. Alpha Particle Energized Neutron
Flux Monitor

7568-15194

Figure 21. Temperature History Chart, Using Binary Gas System for Control

Figure 22. Flux and Temperature Shaping

Figure 23. Flux Shaping Using Eccentric Flux Depressor

Figure 24. Eccentric Flux Depressor Used in NAA 82-1 Irradiation Assembly

7635-5134A

CAPSULE FOR RECOIL FISSION PRODUCT STUDIES

DESIGN CRITERIA

It is desired to determine the effect of fission products on the properties of refractory metals. The design concept for a recoil fission product capsule consists of a tungsten test specimen surrounded by a surface containing fissionable U^{235} . During irradiation, the fission products recoiling from this surface would then strike and diffuse into the test specimen.

The fueled layer would have to be as concentrated and as thin as possible to permit the maximum concentration of recoil fission products from the fuel surface while minimizing heat production, which in turn would lower the temperature difference between the specimens and reference thermocouples located in the surrounding structure. Ceramics personnel at GE-NMPO established the feasibility of fabricating a ceramic tube with a 0.001-inch-thick layer of fully enriched UO_2 on the inner surface. For compatibility, depleted UO_2 was chosen for the ceramic tube, although its low thermal conductivity required a thin wall tube. The test specimens inside the ceramic fuel tubes must be sealed in a metal tube to prevent escape of fission products. Five such tubes, made of rhenium, are to be inserted in a massive cylinder of molybdenum to provide a means of smoothing out temperature perturbations, and to provide a secondary temperature reference which could be simply instrumented.

In the final design, the estimated temperature difference between test specimens and the large molybdenum cylinder is about $400^{\circ}F$. To achieve the desired maximum sample temperature of $3270^{\circ}F$ ($1800^{\circ}C$), the molybdenum cylinder

temperature must be about 2870°F. This requires an insulating 0.105 inch annulus filled with a mixture of helium and argon surrounding the cylinder. The outer surface is cooled by reactor water. About one-half the generated heat is due to gammas, the balance to fission heat. By varying the helium-argon relative concentrations, test reactor local power variations and design uncertainties can be compensated for.

The specimen temperature will be monitored by thermocouples sealed inside the ceramic fuel tubes. Thermocouples measuring the molybdenum cylinder temperature will provide a secondary reference in the event of failure of the primary thermocouples.

Close tolerances are required throughout since each one mil variation in the heat transfer path or 10 percent variation in the fuel layer thickness will generate a 35° temperature difference between the sample and the reference thermocouples.

The test is scheduled for insertion in the ETR reflector with a peak unperturbed thermal flux of about 1×10^{14} neutrons per square centimeter per second.

GENERAL DESCRIPTION

The fully annealed tungsten test specimens are designed to determine the effects of creep rupture properties at test conditions. The specimens of a cylindrical button-head type have an overall length of 2 inches. The heads are 0.060 inch in diameter and 0.25 inch long; the shoulders are 0.050 inch diameter and 0.25 inch long, and the gauge section is 0.040 inch in diameter and one inch long.

Ceramic tubes having a 0.100-inch O.D. and a 0.062 inch I.D. house the specimens. The tubes are either depleted UO_2 with a 0.001-inch-thick layer of enriched UO_2 on the inside diameter, or depleted UO_2 alone. A subassembly consists of six specimens in tandem inside the UO_2 sleeves, sealed in a rhenium tube 0.125 inch in O.D., 0.104 inch in I.D., and 14.25 inches long. The subassemblies, sealed at the ends after helium leakchecking are outgassed and backfilled with pure helium to 3 psia. The rhenium tube extends beyond the specimen length at either end for a distance of approximately one inch. Three of the subassemblies are coated with enriched UO_2 surrounding the six tungsten specimens, and two have depleted UO_2 tubes only, to provide control specimens with a minimum of fission products.

The depleted uranium oxide tubes are extruded to a developed dimension, dried and pre-fired. Deposition of the enriched UO_2 is accomplished by flowing an aqueous slurry composed of enriched UO_2 , water, and a binder of ammonium alginate through the inside diameter. The tubes are then dried and furnace treated in hydrogen at 1675°C for 2 hours. Two W / W - 20Re thermocouples are located in a "thermocouple well" which extends three inches into the uranium and act as primary temperature sensors immediately adjacent and tandem to the specimens. (Figure 1).

The five subassemblies are fitted in five holes in the massive cylinder of molybdenum, having an outside diameter of 0.850 and 12.6 inches long, on a 0.56 inch bolt circle. For ease of fabricability, the cylinder is made up of half-inch lengths which upon assembly are held as one continuous cylinder by an external thin walled tube of molybdenum.

A center 0.25 inch hole down the length of the cylinder serves as a gas return and a receptacle for secondary temperature measurement by four W / W - 20Re thermocouples.

The molybdenum cylinder is held concentric within a 1.065-inch I.D., 1.125-inch O.D. tube by stainless steel end spacers (forming the principal controlled heat transfer gas gap to achieve the design temperature).

DESIGN PROBLEMS AND FABRICATION TECHNIQUES

Because of the high temperature, all materials in the hot section must be within the classification of "Refractory Metals."

The desired test material was established as tungsten. The sleeves were made of depleted UO_2 to insure matching the thermal expansion coefficients and also to assure good cohesion with the enriched layer of deposited UO_2 . The rhenium was selected as the tube material UO_2 as it is an effective oxygen diffusion barrier thus preventing any change in urania stoichiometry.

The end pieces to be welded to the rhenium tubing for sealing are of Mo - 50Re which has good weldability. Tantalum end pieces were initially used, but were found to be unworkable due to an extreme brittleness at the boundary layer between the rhenium and tantalum.

Considerable experimental work was necessary to find a refractory metal to serve as a "pinch-off" tube for each of the subassemblies. The tube material had to be weldable to the end pieces and have the ductility to be pinched off and welded as a final subassembly seal after helium leak checking, out-gassing, and helium back-filling. Molybdenum-50 rhenium alloy fulfilled these requirements.

All thermocouples are W / W - 20Re stranded wire having a 0.013 inch outside diameter. Thermocouples are made up of 0.003 inch individual wires. This type of thermocouple is exceedingly flexible; e.g., the tungsten lead may be knotted with no ill effects. The loss rate for this type thermocouple has been exceedingly low after numerous applications up to 2200°C for many test hours in-pile.

CAPSULE FOR THERMOCOUPLE STUDIES

DESIGN CRITERIA

The capsule for thermocouple studies was conceived as a reference mass, the temperature of which could be continuously monitored by thermocouples undergoing irradiation, and calibrated periodically by identical thermocouples which could be easily inserted and withdrawn to minimize exposure to irradiation. The calibrating thermocouples can be replaced if necessary without interrupting the test.

Since gamma heating varies with reactor operation, and because of uncertainties in the assumed value for gamma heating and heat transfer variables, some means of temperature control is necessary. A variable gas mixture in insulating gaps between source and sink was chosen for reliability and wide control range where heat is transferred by conduction.

In the capsule, the principal source of heat is a molybdenum cylinder. To improve the temperature control range, the fraction of heat transferred by radiation was reduced by surrounding the cylinder with a series of three thin-wall concentric radiation shields. An operating temperature range between 3000° and 4000°F is indicated if the estimated parameters are within 25 percent of the true value.

The NMPO in-pile facility in the A-2 location of the ORR core was selected for this test because it permits partial withdrawal of the capsule for calibration runs at temperatures below 3000°F.

GENERAL DESCRIPTION

The specimens consist of nine W / W - 26Re stranded thermocouples permanently attached to a tungsten disc 0.499 inch in diameter and 0.120 inch thick.

External to this disc is a molybdenum cylinder having an outside diameter of 0.800 inches. Surrounding this assembly are three 0.020-inch-thick molybdenum heat shields gapped 0.060 inch apart. External to the shields, a water cooled can, 1.445 inch in outside diameter with a 0.035-inch-thick wall, is provided.

As shown in Figure 2, a molybdenum probe, 0.187 inch in O.D. with a 0.037-inch-thick wall with a tungsten bullet nose, may enter the hot zone carrying two W / W - 26Re thermocouples. The probe enters the fixed tungsten block, reading from the same temperature source as the nine fixed thermocouples. All thermocouples are insulated with BeO in the hot zones and with Al_2O_3 elsewhere. The entire probe will be outgassed and back-filled with helium to minimize temperature differences that may exist.

In the normal test condition, the probe is withdrawn out of the flux to minimize the dosage received by these control thermocouples. The probe will be used once every 48 hours for a 30-minute duration to monitor the fixed thermocouples for changes in the thermo-electric output that theoretically will occur with increasing neutron dosage. The cartridge will have a flow of $< 100 \text{ cm}^3/\text{min}$ high-purity helium circulating through the inner shell; provisions are made to dilute the helium with argon or to evacuate for control of test temperatures. All nine fixed thermocouples, being attached to a single tungsten block, allow readout of any combination of leads, including leads from the non-irradiated probe thermocouples.

The probe housing will maintain an independent helium flow system exhausting helium past the probe to a restricted passage and to the reactor off-gas system. This second separate positive pressure flow system is necessary because, unlike the cartridge system, pure helium is continually required. The possibility of bonding of the refractories (probe assembly to cartridge assembly) is lessened with helium and the necessity for providing a movable seal to enable manual operation of the probe is facilitated.

DESIGN AND FABRICATION TECHNIQUES

This design required joining a refractory metal to a stainless steel component at several points. The joint requirement was primarily for effecting a helium leak-tight seal at temperatures below 800°F.

These joints were accomplished by welding 304 stainless steel tubing over a close-fitting molybdenum tube by inert arc methods. Tests were made to determine the effects of thermal cycling from 800°F to room temperature as judged by checking the joint with a helium leak detector. The cycling indicated no ill effects and additional use will be made of this joint in subsequent in-pile tests. In all cases, the stainless-to-molybdenum joints were made in zones where temperatures would not exceed 800°F.

Gas Purity

High-purity gas must be used in all tests containing refractory metals. Gas with 1 ppm impurity is purchased from a vendor or gas of < 6 ppm is flowed over titanium chips at 1500°F to reduce impurity content to 1 ppm.

All cartridge materials in contact with the clean gas are evaluated in respect to gas retention properties at initial design; porous fibers or organics are avoided. Prior to the assembly of the cartridges containing refractory metals, extreme caution is taken in cleaning all components and "white glove" procedures are maintained during assembly.

A brief summary of the pre-irradiation cleanup procedure as followed by the Oak Ridge (LITR) and Idaho (ETR) to outgas the cartridge and connecting inert gas lines is given below:

1. The inert gas lines and cartridge are connected to a vacuum system and evacuated to $\sim 1 \times 10^{-5}$ mm of H_g, followed by a high-purity argon backfill.
2. An auxiliary heat source (electrical heaters on the lead tubes and a tube furnace for the cartridge) is used to increase the temperature of the entire assembly to $\cong 600^{\circ}\text{F}$ and again evacuated to $\sim 1 \times 10^{-5}$ mm of H_g. The heat is maintained until a stable pressure exists, then the assembly is allowed to cool to room temperature and pressurized to ~ 2 psig with argon. This system is repeated a minimum of three cycles. The assembly is then sealed prior to transfer to the reactor.
3. The seal is broken while tubing connections are made at reactor installation, a vacuum system is then connected to a system spurline and evacuated again to a stable pressure, then backfilled with pure helium at a positive pressure, and maintained over the duration of the test at a flow of < 100 cubic centimeters per minute.

IN-PILE MEASUREMENT OF THE THERMAL EMF OF $\text{UO}_2^{(a,b)}$

by

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This paper is based on work performed under Atomic Energy Commission
Contract AT(45-1)-1830.

Fig. 1- Capsule for recoil fission product studies

Fig. 2- Capsule for thermocouple studies

IN-PILE MEASUREMENT OF THE THERMAL EMF OF UO_2

by

L. A. Penber

The purpose of this test was to determine the thermal EMF produced by a UO_2 fuel core during irradiation. This thermal EMF results from the completion of an electrical circuit including two dissimilar materials--in this case UO_2 and tungsten--which are in contact at a hot junction near the center of the element and at a cold junction near the outside of the element. A nuclear reactor incorporating a very large number of such elements can produce useful thermoelectric power in addition to nuclear power.

The irradiation capsule was clad in 304L stainless steel with an outer diameter of 1.440 inches and a 0.030 inch wall. The overall capsule length was 16-1/2 inches. UO_2 pellets which had a sintered density of 10.3 grams/cm³ were centerless ground to 1.25 inch diameter and a 0.250 inch diameter hole was drilled along the axis of each pellet. The UO_2 pellet stack was surrounded by a tungsten grid comprising 0.020 inch wires at 1/2 inch intervals. Four legs of the grid were allowed to extend past the pellet stack to act as contact between the grid and the electrical lead. A tungsten helix was inserted into the axial hole in the fuel stack and allowed to expand to contact the UO_2 . A straight extension of the helix protruded from the UO_2 to form the hot electrical contact. The pellet stack was slipped into the stainless steel cladding with the grid and the helix in position. Arc fused UO_2 powder was then vibrationally-compacted into the outer annulus and the central cavity to ensure intimate contact between

the UO_2 and the tungsten during irradiation. Aluminum oxide pellets were used as electrical insulators at the top of the fuel length. These were formed in such a way that the tungsten extension could be mechanically connected to the electrical lead.

The electrical lead used was a stainless steel clad, MgO insulated nickel wire. The central tungsten lead was mechanically attached to the nickel wire in a small nickel connector block. The outer grid was mechanically connected to the stainless steel sheath which was grounded to the cladding.

The lead exited from the fuel chamber through a Conax fitting. This type fitting was chosen in order to prevent any possible stress on the internal connections. Since it is difficult to ensure a gas tight seal with a Conax fitting, a secondary seal was made using a Swagelok fitting. The openings of the fitting were then brazed to ensure gas tight operation.

This capsule was irradiated in the B-3 facility of the MTR. The capsule was installed in a standard B block basket and iron-constantan thermocouples were used to monitor bulk water ΔT . Since the flow in the B-3 facility is only 20 gpm, a ΔT large enough to be useful in temperature profile calculations is available.

During reactor startup, measurements of open circuit voltage, short circuit current, and load voltage versus load current were made. At a calculated core temperature of 1000°C , an open circuit voltage of 330 millivolts and a short circuit current of 56 milliamperes were measured. At this power a resistance of 3.4 ohms was calculated. With an increase in power a polarity reversal was noted. After a reactor scram, readings during power recovery followed a curve of the same general shape but the curve was slightly displaced from the original curve. This indicates that enough structural development had taken place to improve the thermal conductivity appreciably.

① CAPSULE FOR ELECTRICAL RESISTIVITY AND EMF MEASUREMENT

A DILATOMETER FOR MEASURING AXIAL DISPLACEMENT
OF FUEL DURING IRRADIATION

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ORNL-P-2179

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Axial displacement of UO_2 fuel pellets during irradiation has been measured with a dilatometer employing a commercial pneumatic linear motion transducer. A unit similar to one used by Notley¹ and others for measurements of the change in cladding length was redesigned to fit a modified EGCR type fuel irradiation capsule assembly.² A bellows sealed push-rod, coupling the fuel stack to the transducer, provided a unique arrangement for penetrating the fuel encapsulating can.

The transducer, near the top of the test assembly shown in Fig. 1, consists of three chambers separated by two small sharp-edged orifices. A precision-tapered pin moves in the second orifice. The pin, attached to the tip end of the push-rod, moves axially with fuel displacement and thus varies the flow area of the second orifice. Nitrogen flow to the transducer is adjusted to give a predetermined differential pressure across the upper orifice (ΔP_{12}). The differential pressure across the second orifice (ΔP_{23}) is then a function of the position of the tapered pin and hence the length of the fuel stack. To insure repeatability the exhaust pressure (P_3) is held constant.

The thin-wall (0.0016 inch) nickel bellows which provides the fission gas seal between fuel can and push-rod is lightly compressed. The push-rod is thereby held against the insulator on top of the fuel stack and follows axial movement of the fuel. The lower end cap of the fuel

can, the NaK-filled primary containment tube, and the gas-filled secondary tube are rigidly attached to provide a stable reference plane against which the fuel movement is related. The differential thermal expansion of the fuel can and NaK tube is accommodated by a separate nickel bellows for each.

Measurements are taken by manually adjusting the nitrogen flow and exhaust pressure to the values used during calibration and observing the differential pressure (ΔP_{23}). The differential pressures (ΔP_{12} and ΔP_{23}) and the exhaust pressure (P_3) are measured with secondary standard quality pressure gages. Complete instrument system calibration during assembly enabled us to obtain an accuracy of ± 0.001 inch over a range of 0.290 inch.

Two capsules so instrumented have performed well in irradiation tests at conditions similar to EGCR test requirements.³ Measured values of the fuel stack heights at design temperatures agree well with expected thermal expansion of the 9-in. fuel stack. Repeatability at lower temperatures during imposed thermal cycles is excellent (± 0.0005) and the values clearly demonstrate that the overall height of the fuel stack decreases during irradiation. Consecutive measurements at zero reactor power, which duplicates the initial temperatures, produce the most interesting results. The fuel stack height in unit No. 1 had shrunk 0.031 inch at 96 days exposure when it was removed because of a failure in another part of the capsule; and unit No. 2, still operating in mid-February after 108 days, had shrunk 0.045 inch without indication of leveling off.

The contributions of G. R. Samuels and the helpful discussions with J. R. MacEwan of AECL are gratefully acknowledged.

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Pneumatic Dilatometer Transducer for Determining
Axial Movement of UO₂ Pellets During Irradiation.

IVa. CAPSULE OPERATIONS - FISSILE MATERIALS (BINARY GAS TEMPERATURE CONTROL)

The Temperature Control of Encapsulated Irradiation Specimens.

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INTRODUCTION

1. One of the major problems of testing of fissile and non-fissile specimens in test reactors is the precise achievement of some required temperature level over the specimen, and the maintenance of this level during changes in reactor flux patterns and operating power levels. Required specimen temperatures are normally well above those of the coolants available in the reactor and consequently some thermal insulation is required. In the case of encapsulated specimens the containment walls of the capsule offer some insulation but extra thermal resistance between them and the specimen or heat sink may have to be provided.
2. A method commonly adopted for temperature control is the provision of ballast electrical heating to a capsule, this being automatically controlled by the signal from a thermocouple on or near the specimen. However this solution leads to complexity in capsule design.
3. Since temperature specification is normally more precise than that concerning the heat output of a specimen, it is sometimes permissible to regulate temperatures at the expense of heat rating by moving the specimen along a suitable neutron flux gradient.
4. In this paper some other methods of achieving and controlling the temperature of a specimen that are under development at D.E.R.E. are discussed. Some of these methods are novel and may well offer the experimenter and capsule designer improved control of irradiation conditions. Several of the methods exhibit the common property of strong dependence of the thermal insulation on the pressure of the gas filling.

LOOSE CERAMIC POWDERS

5. The effective thermal conductivity of a packed bed as shown by Laubitz⁽¹⁾ and others (2, 3, 4, 5) depends largely on the thermal conductivity of the gas phase, the packing fraction, and normally to a lesser extent, on the thermal conductivity of the solid phase. Deissler and Eian⁽²⁾ obtained good agreement between experimental data and a derived expression based on a mathematical model of orthogonal packed spheres. This work was extended by Schotte⁽⁴⁾, but both these and other workers considered only the case of relatively large packed powder beds.
6. Now as can be explained by the Kinetic Theory of Gases, the gas phase conductivity is dependent upon the void spaces for as the size of these decreases to the order of the mean free path of the gas molecules, the thermal conductivity of the gas phase and thus the effective conductivity is reduced. The so called "break away pressure" is given⁽²⁾ by

$$P_b = 3.82 \times 10^{-20} \frac{T}{D \delta^2}$$

where P_b is the break away pressure in lbs/in² abs.

T is the temp. °K.

D is the average particle dia. in inches

δ is the gas molecule size in inches.

Thus, for a 10 μ size powder/helium mixture it is about 1,000 lbs/in² abs. at 300°C and increases correspondingly for a smaller powder size or higher temperature.

Hence, in any case, to obtain the MAXIMUM order of thermal conductivity, the bed must operate ABOVE the break away pressure, as can be seen from experimental and predicted values plotted in Fig. 1.

7. Further, since the conductivity of the gas phase is dependent upon the effective gas space between the particles, it will be effected by any change in the particle size and packing fraction. Thus, as shown by Evans in a detailed theoretical study carried out on our behalf by I.R.D. Co. Ltd. (6), unless the packed bed can provide a noticeable increase in the thermal conductivity above that of the gas alone, the bed size may be insufficient to provide the flexibility required for stability, particularly when temperature cycling occurs.

8. Experimental work was carried out at D.E.R.E. to determine the effective conductivity and stability of powders in small gas filled gaps at high heat transfer rates, and their suitability for use as the controlling heat transfer medium in small rigs and capsules.

9. The simple rig used is shown in Fig. 2. It consisted of a hollow water cooled copper 1.0 inch diameter tube surrounded by an annular gap 0.01" wide and 4" long which could be filled with powder. This was surrounded by a 3 KW mineral insulated, metal sheathed heater, embedded in copper giving a heat rating of approx. 300 watts/cm. length. The outer part of the annulus was maintained at 600°C. The heater assembly was supported on thin plates and struts and well insulated with heat shields to minimise losses. The temperatures across the annulus were measured by six 1 mm. mineral insulated cable chromel alumel thermocouples embedded at various points in the outside wall of the annulus and six 0.5 mm. thermocouples embedded at corresponding points on the inner, water cooled wall. The heat flow through the powder was determined from measurements of the rate and temperature rise of the coolant flow.

10. A packing fraction of 0.54 was obtained by vibro compacting the powder in situ in the rig. Although this compares well with a theoretical estimate of 0.57 based on data by McGeary(7) it was much less than might at first appear possible, owing to the relationship of the particle to containment size and wall end effects. Again, it might be expected that high packing densities could be obtained with fine particles but as shown by Wegl(8), the repelling electrostatic charge forces on the particle surfaces become comparable with the weight of the particles themselves and they become as if suspended in solution resulting in a lower density.

11. Good agreement was reached between predicted and experimental values of the effective thermal conductivity of 125-150 μ size silicon carbide, and 400-500 μ aluminium oxide powder filled helium gas gaps as shown in Fig. 1. The effective conductivity of the 125-150 μ size powder filled bed is seen to be approximately ten times that of the helium gas alone.

12. Surprisingly, there was no significant change in the effective conductivity over the short range of mean bed temperatures attained. However, there was a noticeable change after temperature cycling between 100°C and 600°C as shown in Fig. 3. This was presumably caused by a noted degradation of the powder particle size, which in one case approached 25%, by the crushing effect of differential expansions during thermal cycling.

13. The partial reduction of the powder size would normally be expected to give rise to a higher packing density and a consequent increase in the effective thermal conductivity. The variations were, however random, probably on account of (a) bridging effects and (b) the creation of smaller gas voids.

14. Investigations were also conducted into the use of powders as a direct heat transfer medium at higher temperatures in the range 700°C to 1,000°C. The simple apparatus is shown in Fig. 4. It consisted of a 0.20 dia. heavy current tubular heater fitted axially in a helium and powder filled, water cooled, cylindrical pressure vessel. The radial gap of 0.37 inches was filled in turn with samples of 125-150 and 400-500 μ size Al₂O₃ powders, vibro compacted in situ in the rig.

15. The experimental values of thermal conductivity were in this case found to be about 20% above theoretical predictions based on the Schotte⁽⁴⁾ and Deissler and Eian⁽²⁾ correlations; giving a maximum thermal conductivity about 7.0 times that of the helium gas alone. However, the deviation probably represent the experimental error in this exploratory work. Owing to the lack of adequate instrumentation, the experimental results were based on the arithmetic mean temperature of the bed; determined from temperature measurements of the heater and coolant wall with a small correction for end losses applied. As the mean bed temperature never exceeded 500°C the radiation effects were ignored. However, examination of the radiation term

$$K_r = \frac{p}{\frac{1}{K_s} + \frac{1}{K_r^0}} + (1 - p) K_r^0$$

(K_r is radiation contribution K_s thermal conductivity of solid, $K_r^0 = \frac{H_r}{D_p}$ where H_r is heat transfer coeff., for radiation and D_p is the particle size).

given by Schotte⁽⁴⁾, indicates that although with these powders the radiation contribution is insignificant at the mean bed temperature, it will be noticeable in the higher temperature zone of the bed.⁽⁹⁾ Thus the effective overall bed thermal conductivity will be higher than that predicted on the basis of the mean bed temperature.

16. The degradation, after some thirty temperature cycles between 1,000°C and 700°C, was relatively small. The thermal stability after temperature cycling and over periods of 200 hours was also quite good, particularly when filled with 125-150 μ Al_2O_3 powder.

17. Our work on powders does indicate, therefore, that although the requirement of a high thermal conductivity and bed flexibility are incompatible, packed powders can be a predictable and stable heat transfer medium when the bed size is sufficient to allow a measure of relaxation to occur.

18. The effective thermal conductivity of a packed powder bed is gas pressure dependent. Schotte⁽⁴⁾ has shown that the effective gas conductivity (K_{ge}), in this range, can be calculated from the Equation:

$$K_{ge} = \frac{K_g}{1 + 4.4 \times 10^{-21} \left(\frac{\gamma}{1 + \gamma} \right) \left(\frac{p}{1 - p} \right) \left(\frac{T}{PD^2 N_p} \right)}$$

Where K_g is the conductivity the gas phase
 γ is the ratio of specific heat C_p/C_u
 T is the temperature in °K
 P is the gas pressure in lbs/in² abs.
 N_p is the Prandl number
 D is the average particle dia. in inches
 δ is the mean dia. of the gas molecule in inches
 p is the packing fraction

19. By substituting K_{ge} for K_g in the Deissler and Eian expression

$$\frac{K_e}{K_g} = \frac{1}{2} \frac{\pi}{\left(\frac{K_g}{K_s} - 1 \right)^2} \left[\left\{ \frac{K_g}{K_s} - 1 \right\} - \ln \frac{K_g}{K_s} \right] + 1 - \frac{\pi}{4}$$

the effective bed conductivity, K_e , can be obtained for any packing fraction (p) and solid phase conductivity (K_s). Again we would refer to our own experimental results and theoretical predictions for 100 - 120 mesh Si Carbide and 30-40 mesh Al_2O_3 powders shown in Fig. 1.

20. Obviously, as previously noted, the break away pressure increases for a finer powder. Thus a balance is required between the powder size and the control range to keep the pressure within practical limits.

21. However a useful potential exists for temperature control by variation of gas pressure of the bed. We have not employed this system so far being more interested in methods of control of more highly rated capsules than this is considered suitable.

22. In our work on sprayed ceramics it was found that as might be expected, this form of system is also gas pressure sensitive. Indeed our few experimental results indicated that it had a significant effect with a 10% change in the effective thermal conductivity for 20 p.s.i. helium pressure rise. Thus a noticeable potential exists for temperature control of capsules and other experimental rigs using this system. Further the potential can be extended by changing the simple ceramic barrier to a cermet i.e., a very wide range of heat transfer rates could be accommodated.

SPRAYED CERAMICS

23. From further considerations of the two phase gas/solid system it is obvious that the effective thermal conductivity could be noticeably improved if the packing density could be increased. One method of achieving this is by using sprayed ceramic or metal powders. Obviously it is most important that the sprayed material should remain stable, particularly under temperature cycling, and also remain bonded to the heat transfer surface.

24. As a result of E.M.R. work carried out by the I.R.D. Co. Ltd. on our behalf, a successful irradiation capsule using this principle has been built and tested in D.M.T.R. The capsule is shown in Fig. 5. The stainless steel clad specimen fuel pin is encapsulated in a sodium filled container. A composite copper tube drawn over this containment is used to provide a good bond to the sprayed alumina and pockets for the monitoring thermocouples. The required thickness of alumina is sprayed by a controlled process and then covered with a thickness of sprayed copper. The external, coolant tube is imploded on to this outer copper layer to provide a good bond to the copper and reduce interface resistance here to a minimum.

25. As no real data on the thermal conductivity of sprayed powders were available, theoretical calculations were again based on the work of Schotte⁽¹⁰⁾. However, the nature of the sprayed particles made assumptions, of the void space in particular, difficult. It was found, as since also reported by Boganov⁽¹¹⁾, that such coatings effectively consist of splattered, mechanically bonded droplets, and appear to be more like an assembly of disc-like platelets. It is therefore not surprising that the predictions based on spheroidal packing of powders could not be applied.

26. The required thickness of coating was, therefore, determined experimentally. The effective thermal conductivity was found to be of the order of $0.013 \text{ watts/cm}^2/\text{°C/cm.}$ at a mean temperature of 300°C and helium gas pressure of 20 psi (g) and with a void fraction of 0.22.

27. The prototype capsule was filled and sealed with a small positive pressure of helium designed to give a working pressure, at temperatures of 20 psi(g). It was assembled as a simple stem rig and irradiated in D.M.T.R. using the reactor heavy water as a heat sink, with heat transfer occurring by phenonema of sub-cooled nucleate boiling.

28. The performance of the capsule was very much as predicted and heat transfer i.e., specimen temperature remained quite stable over 6 weeks in a thermal flux of 10^{13} n.cm²/sec.

29. The capsule is now undergoing post irradiation examination. Neither this or radiographs taken by X-rays and by a new scheme of neutron radiography has so far indicated a bond failure or break up of ceramic.

WRAPPED METALLIC FOIL

30. This novel type of insulation is obtained by placing a large number of wrappings of a thin metallic foil within a gas gap. Its main feature is that the overall thermal conductance of the metal and gas barrier is a strong function of the gas pressure. This occurs because the foils introduce many gas-metal interface resistances which have the property of being dependent on the molecular mean-free-path (L) in the gas. While this effect is present in any gas gap because there are always at least two such interfaces, it is not normally apparent for the resistance is relatively small. However, if many surfaces are interposed between the heat source and sink, the resulting insulation layer can have predominantly pressure-dependent properties.

31. The wrappings may be regarded as a set of concentric cylinders and for simplicity of explanation, the distance between cylinders can be chosen to be much greater than 'L' over a practical pressure range. Assuming radial heat flow only, the system may thus be represented by two resistances in series: a gas layer composed of the sum of all the layers among the foils with a conductance independent of pressure, and another layer with a pressure-dependent resistance equivalent to that on one surface multiplied by the total number of surfaces present.

32. The usual conduction equations can be used to compute the properties of the invariable portion and the variation of gas conductivity with temperature may also be included. The surface resistance phenomenon has been both experimentally and theoretically investigated and is dealt with extensively in Ref. (12). Several treatments are available and one simple model permits the resistance to be envisaged as equivalent to an extra layer of gas of width 'g' on each surface, in which free-molecule conduction occurs.

33. The dimensions of 'g' may be calculated (Ref.12) from:

$$g = \frac{2}{Pr} \cdot \frac{2 - \alpha}{\alpha} \cdot \frac{\gamma}{\gamma + 1} \quad L = e.L \quad \dots\dots\dots (1)$$

where γ is the ratio of the Specific Heats at constant pressure and volume,

Pr is the Prandtl Number of the gas, and

α is the accommodation coefficient between the gas and the solid.

The variation of 'g' with temperature between 250 - 500°C appears to be small, (Ref. 13) and the Prandtl Number of helium for example increases by 10% between 50°C and 500°C.

34. The thickness of the layer can thus be regarded as directly proportioned to 'L' for a given surface and gas, and hence inversely proportioned to the pressure of the gas. With helium and aluminium for example, it has been found by Guilliano and Mustacchi (Ref. 13) and at D.E.R.E. that the layer is approximately ten times 'L'. The extra resistance on each surface is thus equivalent at N.T.P. to 0.186×10^{-3} cm. of helium.

35. Reference (12) contains a useful list of molecular data for many gases which includes the dimensions of the index 'n' necessary for computing the dependence of 'L' on the absolute temperature. At constant density, 'L' is proportional to $T^n -0.5$ and at constant pressure to $T^n +0.5$. The proposed insulation would work under the latter condition.

36. The sensitivity of the surface-resistance insulating-layer to temperature and pressure changes can be determined as follows:

- (1) The temperature drop, ΔT , may be expressed as:

$$\Delta T = q \frac{MeL}{\pi d_m K} \dots\dots\dots (2)$$

where q is the heat rating per unit length,
M is the number of surfaces (two per wrap),
K is thermal conductivity of the gas, and
 d_m is the mean diameter of the wrapping.

(2) Now, $L = L_0 \left(\frac{T_m}{288} \right)^{n + 0.5} \times \frac{1}{P} \dots\dots\dots (3)$

(for variable density conditons)

where T_m is some mean temperature within the wrapping,
 L_0 is the M.F.P. at 1 atmosphere pressure and 288°K,
P is the gas pressure in atmospheres, and
n is a constant (see para. 5).

- (3) The thermal conductivity of the gas is approximately dependent on T_m to the same index 'n';

$$K = K_0 \left(\frac{T_m}{288} \right)^n \dots\dots\dots (4)$$

where K_0 is the conductivity at 288°K.

- (4) Equation (2) now becomes:

$$\Delta T = q \cdot \frac{Me}{\pi d_m} \cdot \frac{1}{P} \cdot \frac{L_0}{K_0} \cdot \left(\frac{T_m}{288} \right)^{0.5} \dots\dots\dots (5)$$

$$\text{or } \Delta T = A \cdot \frac{1}{P} \cdot \left(\frac{T_m}{288} \right)^{0.5}$$

37. It can be seen that the temperature drop is inversely proportional to the pressure and directly proportional to the square root of the mean temperature in the wrapping. The physical property group; $\left(e. \frac{L_0}{K_0} \right)$ will obviously vary with the gas.

38. Tests have been performed with simple laboratory apparatus with three combinations of foil thickness and number of wraps. Typical results are plotted in Figs. 6, 7 and 8 and it is obvious that the variation of temperature with pressure can be considerable. The technique is now being applied to the design of in-pile capsules.

VARIABLE WIDTH GAS GAP

39. Another method of achieving some degree of temperature control with a gas gap is shown in Fig. 9. In this design, temperature control is achieved by variation of the gas gap between two conical co-axial surfaces. The

amount of taper on the two surfaces provides amplification of the axial movement required for control and the nuclear flux change is insignificant. Following laboratory investigations a prototype having a 1 in. 50 taper was successfully operated in D.M.T.R. Several of the type have now been constructed to provide a temperature control band of 200°C for only 0.5" total axial movement and capable of being set within $\pm 5^\circ\text{C}$ of required control point. This displacement is obtained by a simple worm and nut device at the rig head. At present the capsules have only been considered for work in the range 200-800°C but it would appear to be quite feasible to extend this range by using say, a molybdenum primary containment, for operating at higher operating temperatures.

40. These capsules are excellent when temperature control independent of flux changes are required. This range can of course be extended by variation of the gas mixture or content of the rig.

VARIABLE POSITION DEVICES

41. The majority of simple stem rigs fitted in D.M.T.R. depend on previously predicted gas gap sizes or thickness of solid heat transfer media, such as zircalloy to achieve the required specimen temperature in a specified reactor/flux position. As such there is no provision for temperature control.

42. Temperature control of such rigs is however provided by variation of the rig nuclear heating by altering the rig to reactor core position. The majority of stem rigs in D.M.T.R. are now fitted with adjustable height devices.

43. A number of rigs so fitted are shown in Fig. 10 positioned in a multi-facility reactor shield plug prior to installation in D.M.T.R. A simple removable drive is used to rotate a captive pinion and nut which in turn rotates and lifts the threaded stem of the complete stem cum capsule rig. The amount of lift is limited to 11" by restrictions within the shielding void at the top of the D.M.T.R. reactor.

DOUBLE-WALLED CAPSULE WITH CIRCUMFERENTIAL FINS

44. This type of capsule has been used by the Battelle Memorial Institute and the Argonne National Laboratory for the irradiation of long specimens immersed in liquid metal. One report, (Ref. 14) gives a detailed description of a particular capsule design. The pitch between the fins of constant width was different in each of the capsules tested and data from the in-pile tests are presented in terms of the specimen temperature and reactor power-level for a complete range of fin contact areas. This example illustrates the advantages of this type of capsule wall. Apart from the obviously strong construction, its thermal insulation characteristics span the region between the practical limits of simple gas-gaps and solid metal walls. The variable fin-spacing permits the designer to select a suitable conductance for a particular heat rating with standard capsule parts.

45. Work on this topic by D.E.R.E. has been concerned with determining theoretically and experimentally the characteristics of this type of thermal barrier and containment in order to establish a simple and effective design procedure. The analysis has not been limited to dealing with the form of construction described in Ref. 14 where the outer tubes have been shrunk onto the inner with a pre-stressed condition.

46. The thermal resistance of the double wall is obviously a function of the thickness of each wall and the size and pitch of the fins. However, the contact resistance between the fin-tips and the outer wall is also very important and its accurate estimation can be difficult, for it is dependent on the local interfacial stress at the fin tips, the fin material and surface finish and the gas composition and pressure.

47. The initial clearance between the fin-tips and the outer wall when the capsule is assembled and the heat output from the capsule are the two most significant design parameters that affect this contact resistance for a given fin size and pitch. The stress and thermal patterns in the capsule thus interact and their formulation, solution and experimental checking have been the object of a study carried out under extra-mural contract at the International Research and Development Co. Ltd., Newcastle-on-Tyne.

48. The temperature distribution within the double wall has been determined analytically by dividing it into three cylinders: the inner tube, the fin and gas space and the outer tube. Laplace's equation for radial and axial heat flow is solved for the inner and outer cylinders giving series summation solutions which are linked by assuming radial heat flow only in the fin and gas space.

49. The series-summation solutions have been computed over a wide range of capsule dimensions and the results are given in graphical form in terms of non-dimensional ratios of the dimensions. Isothermal conditions are assumed along the bore of the capsule while three separate conditions are considered along the outside surface;

(a) uniform heat transfer coefficient, (b) boiling heat transfer coefficient (heat flux dependant) and (c) isothermal conditions.

50. A simple thermal-stress analysis has been developed and the results combined with the series-summation temperature solutions are given in graphical form. The cold, fin-tip clearance required to produce a specified fin-tip pressure can be obtained from this analysis and also the maximum stress checked to avoid local yielding which would alter the cold clearance and therefore the capsule performance.

51. IRD have made experimental measurements of overall temperature drop on a series of simulated capsules over a range of heat ratings and inter-fin gas conditions. Typical results are shown in Fig. 11. The experimental results were combined with the theoretical analysis to produce a correlation of thermal contact resistance against fin contact pressure. This correlation, although not necessarily comparable with absolute measurements, does apply to the type of surface finish obtained in capsule production and will embody any errors due to approximations in the theory.

REFLUX CAPSULE

52. Nucleate-boiling is one mechanism of heat removal from a surface immersed in a liquid that is particularly attractive for achieving or controlling the temperature of that surface. Heat-transfer coefficients associated with nucleate-boiling can be very high indeed and mean that large changes in heat flux are accompanied by relatively small temperature changes. If the pressure of the liquid in which the specimen is immersed can be varied, then its saturation temperature can be altered and hence the temperature of the surface can be controlled. Typically, the difference between the surface and the saturation temperatures will be of the order of ten Centigrade degrees for liquids such as water, organics and also liquid metals.

53. R.J. Schiltz and D.R. Patterson of A.N.L. have already operated a rig using these principles (Ref. 15) for irradiating a fuel pin in water that was externally pressurized. In order to investigate the practicability of a self-regulating capsule using boiling for automatic control of the surface temperature of a specimen or group of specimens, a study has been made (for the A.E.A.) at the National Engineering Laboratory, of a capsule with an internal reflux condenser. The capsule system has been described elsewhere in detail (Ref. 16).

54. The proposed design of a capsule is shown schematically in Fig. 12. The specimen, for convenience, is assumed to be a slender cylinder but could be a cluster of cylinders or any other configuration preferably of constant cross-section. It is completely immersed in a liquid (for example sodium) and retained approximately equidistant from the internal wall of the capsule. The two walls of the bottom portion of the capsule are thermally insulated and the two walls of the top portion have essentially little thermal resistance. The void in the top is initially filled with a non-reactive and non-condensing blanket gas. The heat generated within the capsule is removed by coolant in contact externally with the top part of the capsule.

55. As the reactor power is increased from start-up, the heat from the specimen will not escape from the liquid bath until it is boiling. Natural convection in the blanket gas will dissipate an increasingly larger fraction of this heat to the walls of the top part and the temperature of the liquid will increase to the saturation temperature associated with the total pressure of the gas and vapour.

56. At some reactor power, the heat from the specimen will eventually be removed by evaporation of the liquid and by nucleate boiling on the surface of the specimen. The vapour will be condensed on the walls of the chamber and will run back to the bath, slightly subcooled. Any further increase in reactor power will not alter the surface temperature by more than a few degrees above the saturation temperature.

57. The mass flow of vapour to the surface through the liquid bath will increase as the heat output rises. The down-flow of liquid replacing this vapour by mass balance will also increase until the well known phenomenon of flooding (or cessation of liquid flow) of such a counter-current system is reached at the top of the specimen. This critical situation will limit the total heat output permitted from the specimen for liquid starvation will lead to the formation of a continuous vapour film on the specimen and burn-out. Flooding is strictly a hydrodynamic phenomenon and is described for a similar configuration by Griffiths et al (Ref. 17).

58. The characteristics of the condensing part of the capsule will mainly determine the necessary mass of gas that must be present in the capsule when it is sealed during manufacture so that the specimens will run at the predetermined surface temperature. The probable performance of the capsule has been investigated theoretically at N.E.L. by D. Chisholm and T.F. Provan. Equations have been developed which take into account the void fraction of the vapour in the liquid, the critical velocities for counter-current flow, the heat transfer across the condensing film in the presence of a non-condensable gas and the diffusion of the vapour through the non-condensable gas to the condensing surface. Their solution has been obtained using a computer programme for several capsule and specimen geometries with sodium or water as the working fluids and argon or air as the filling gas.

59. The results for the systems studied, indicate that the dependence of temperature on heat flux is not as weak a function as it is with simple constant-pressure pool boiling, and that heat fluxes on the surface of fissile specimens will need to be low for practical condenser sizes. Typical results are shown in Fig. 13.

60. Work at present is devoted to laboratory testing at N.E.L. a water-air capsule with an electrical heater to determine if the theoretical model of the capsule is adequate.

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COMPARISON BETWEEN EXPERIMENTAL &
THEORETICAL RESULTS.

FIG. I.

4.1.13

FIG. 2.

DIAGRAMMATIC LAYOUT OF RIG.

THE STABILITY OF K_s WITH THERMAL CYCLING.

91117

FIG. 5. CERAMIC/METAL BONDED IRRADIATION CAPSULE.

THERMAL RESISTANCE VS. INVERSE OF PRESSURE.

FIG. 6.

FIG. 7.

4.1.20

FIG. 9. GENERAL ARRANGEMENT OF ADJUSTABLE TAPERED GAS GAP CAPSULE.

Figure 10

4.1.21

PERFORMANCE OF TWO CAPSULES WITH DIFFERENT FIN WIDTHS.

FIG. II.

SCHEME OF IRRADIATION CAPSULE WITH INTERNAL BOILING.

FIG. 12.

VARIATION OF CAPSULE PRESSURE WITH SPECIMEN HEAT FLUX.
(CAPSULE LENGTH 5'-0" ; DIA. 1")

FIG. 13.

GENERAL ATOMIC
DIVISION OF
GENERAL DYNAMICS

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DEVELOPMENTS IN IRRADIATION CAPSULE
TECHNOLOGY AT GENERAL ATOMIC*

by

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ABSTRACT

The nuclear irradiation behavior of reactor materials is of prime importance in the development of nuclear power systems. The development of high-fuel-temperature power reactors is imposing challenging problems to the capsule irradiation engineer who must provide experimental devices for irradiation materials at high temperature.

This report describes the approaches taken to evaluate and design test capsules used in the nuclear development programs at General Atomic. Improvements in capsule technology and assemblies associated with capsule design are discussed as well as a specific design associated with a broad, high-temperature controlled irradiation capsule.

INTRODUCTION

The Capsule Irradiation Group at General Atomic provides irradiation experiments for the major materials development programs related to our nuclear power programs. These materials development programs are directed toward advancing high-temperature materials technology to achieve high thermal efficiency in nuclear power systems. The high temperatures, 1000° to 1800°C , of irradiation have challenged the capsule engineer in problems of design to achieve and maintain these temperatures, measurements of these temperatures, and other problems associated with materials compatibility.

The Capsule Irradiation Group has met this challenge in three different materials development programs. The following written presentations summarize the approaches taken by the capsule engineer. The capsule engineer leads a team of specialists who are responsible for the fabrication of the capsule and associated controlling instrumentation, operation of the capsule at the test reactor, data reduction, and disassembly of the capsule in General Atomic's Hot Cell. The control and advancement of improvements in capsule fabrication and quality has contributed to our successful capsule irradiation experiments. The papers discuss materials selection, design features, and capsule improvements.

SECTION A

A GUIDE TO MATERIALS SELECTION FOR MULTIPLE- GAS TEMPERATURE CONTROLLED CAPSULES

A-I. INTRODUCTION

Irradiation capsules perform a singular function; they contain test materials during an irradiation period, while maintaining the experimental design conditions. Failures in capsule irradiation experiments, which cause premature termination of the test, often can be attributed to poor materials design. The proper selection of materials for a specific irradiation test capsule is dependent on the mission and test conditions required. As the temperatures of irradiation tests increase and high flux exposures become important, even greater emphasis must be placed on providing the proper selection of materials and the design which incorporates or combines these materials. A selection of proper materials for a specific capsule design is presented in this report, with emphasis on high-temperature and high flux exposure requirements and on physical integrity in the capsule design.

A-II. GENERAL DESIGN CRITERION

A typical irradiation capsule program at General Atomic is the Advanced Beryllium Oxide Concepts program. Possible structural materials for the program include SiC, SiC coated graphite, graphite, and BeO. In various advanced concepts, these materials are used separately and in combination over a wide temperature range. The irradiation program for these materials involves engineering tests at various temperatures to determine dimensional changes, thermal conductivity, strength, and

compatibility. At each test temperature, it is desirable to test as many specimens of each type as possible. For this reason, the specimens were selected to be long and narrow, and internal heaters were rejected as a heating source to save space. The temperatures are attained by using internally generated gamma heating. Because the test materials are all materials of low "Z" number, the amount of heat generated per unit volume will necessarily be low; therefore, efficient use of this heat must be made to achieve the design temperatures through the use of axial and radial radiation shields. In the advanced reactor concept, the structural materials must remain integral in the temperature range of 500 to 1700°C and to radiation exposures as high as 5×10^{21} nvt > 1 MeV. Therefore, several test temperatures need to be evaluated to determine the integrity of the materials at high radiation levels. To minimize expenditures and time, one multicell capsule was selected to cover the entire temperature range. To interpret the observed changes, accurate temperature measurement and control must be maintained throughout the irradiation period.

The test reactor also influences the general design. Semifixed sizes and geometries limit the ultimate size of the irradiation capsule. For fueled experiments and most high-temperature experiments, double containment of the specimens is required. General integrity and chemical inertness of the exterior surface of the capsule are test reactor requirements. The experiment, of necessity, must not present an undue radiological hazard to the reactor personnel. The unpredictable variable heating rates from cycle to cycle affect the general design mostly in maintaining constant temperature.

As a result of the above criterion, the general design of the Advanced Beryllium Oxide Concepts Materials Capsule becomes a doubly contained, multicelled, multiple dual gas controlled, temperature monitored, high-temperature capsule, as depicted in Fig. A-1.

Fig. A-1
ADVANCED BERYLLIUM OXIDE CONCEPTS;
STRUCTURAL MATERIALS IRRADIATION CAPSULE

A-III. SELECTION OF MATERIALS

Once the general design of the capsule has been completed, the selection of materials for the capsule components may proceed. Components that are required as a result of the general design are (1) the outer or secondary containment, (2) the thermal spacer, (3) the inner or primary containment, and (4) the thermocouples. The methods of joining the selected materials (i. e., welding and brazing) must also be considered.

SECONDARY CONTAINMENT

The secondary containment is that component system which protects the reactor coolant from the materials inside the capsule. The system consists of an outer shell, end caps, and a lead tube. The selection of materials for this portion of an irradiation capsule is governed solely by test reactor restrictions. For example, the system for the Materials Testing Reactor (MTR) and the Engineering Test Reactor (ETR) requires that:⁽¹⁾

1. The materials of construction of the outer shell, which will be in contact with the water, must not corrode, slough off, or in any other way cause deleterious effects on the high quality process water.
2. The use of copper, cadmium, and lead cannot be tolerated. (Recommended materials include aluminum or aluminum alloys (of less than 0.5% copper), stainless steel 347 or other stabilized stainless steel, and zircaloy. Heliarc welding is recommended for joining metallic parts. Silver solder is unacceptable as a brazing compound.)

In addition, the MTR-ETR system requires that the lead tube of an instrumented capsule be made of stainless steel. This provides extra strength in the section of the lead tube which is subjected to high reactor water turbulence in the reactor tank above the core.⁽¹⁾ To efficiently join

the lead tube to the capsule shell, the capsule shell should be made of a like stainless steel. The end caps should also be of the same stainless steel. The secondary containment must, therefore, be fabricated from a stabilized stainless steel such as 304, 321, or 347. Proper selection of the tubing size depends on the stresses that the containment will have to withstand.

THERMAL SPACER

The functions of the thermal spacer are (1) to provide an entrance to and exit from the capsule area for gas tubes and thermocouples and (2) to prevent high-temperature operation of primary containment.

Because the general design of the capsule calls for temperature monitoring and temperature control, provisions for penetrating each primary containment with gas tubes (to change the conductivity of the gas mixture in the primary containment) and thermocouples must be considered. To conserve the limited volume in the fixed capsule geometry for the specimens, the gas tubes and thermocouples should occupy a minimum volume in the capsule. The gas tubes and thermocouples also serve important functions as portions of the primary containment. Their integrity throughout the irradiation period must be maximized. The strength of the tubes and thermocouple sheaths remains high if low operating temperatures are maintained.

The general capsule design also calls for several primary containment cells. Gas tubes and thermocouples from cells at the bottom of the capsule must pass by or through those cells which are higher in the capsule. Gas tubes or thermocouples from cells near the bottom of the capsules which pass through all the cells increase the possibility of leaks because of multiple penetrations of the higher cells. This type of design is extremely difficult to fabricate. Therefore, the gas lines and thermocouples should pass by the higher cells in the annulus between the primary and secondary containment cans. This method would increase the integrity

of the gas tubes and thermocouples because the annulus is the lowest temperature region of the capsule.

To control the temperature of the specimens and internal capsule components, the annulus through which the gas tubes and thermocouples pass must be filled with a thermal spacer. The radial gap between the primary containment and thermal spacer can be designed to allow for temperature control within the limits of the experiment and the restrictions of the internal capsule component. Therefore, given enough reactor coolant to cool the capsule, temperature control of the capsule essentially begins at the internal surface of the thermal spacer. The spacer has served its twofold purpose of providing the volume for gas tube and thermocouple penetration and control of the internal capsule temperature.

The thermal spacer does not perform as a containment vessel or load-bearing member of the capsule system; therefore, strength is of concern only insofar as one would wish the thermal spacer to remain integral during assembly and operation. The most important physical property of the thermal spacer material is thermal conductivity. The exterior spacer area, as well as a sizeable volume of the spacer, is taken up by gas tubes and thermocouples. Thus, the heat flow area and transfer area are grossly reduced. The inner spacer surface is the effective temperature control surface; therefore, the lower the inner surface temperature the greater the total control and the lower the internal capsule component temperatures. To accommodate the reduced heat flow area and low inner surface temperature, the thermal spacer should be made of a high thermal conductivity material. Also, the material should be of low density (i. e., low "Z") to reduce γ -heating in the spacer.

Two common low density, high conductivity materials are available to serve as the spacer, graphite and aluminum. It is desirable to have an interference fit between the spacer and the secondary containment during operation to improve the heat flow characteristics of the assembly. This consideration together with integrity during fabrication causes the

rejection of graphite as the spacer material. Graphite would tend to crumble and erode during fabrication and operation. Therefore, aluminum is the obvious choice for the spacer material. The operation temperature of the spacer (normally in the 150 to 300°F range) coupled with its high thermal conductivity [$\sim 100 \text{ Btu}/(\text{hr})(\text{ft})(^\circ\text{F})$] makes it an ideal material for the thermal spacer. Almost any aluminum alloy would suffice as the spacer. However, from the availability, strength, and machinability aspects, aluminum alloy 6061 has been found to serve quite adequately.

PRIMARY CONTAINMENT

The primary containment is probably the most important structural component of the capsule design. It must contain the specimens in the proper geometry as well as maintain its integrity to prevent leaks between the separate gas control systems. The primary containment is made up of a can, end caps, gas tubes, and thermocouple sheaths. The various components must withstand a temperature range of 975 to 1100°C and neutron irradiation doses up to 5×10^{21} nvt (fast and thermal). The components must be compatible with or protected from impurities from the specimens and control gases.

The normal nuclear metals, such as carbon steels, stainless steels, zirconium alloys, and aluminum alloys, are unacceptable as the primary containment materials because the temperature strength properties of these materials are not sufficiently high to warrant their use. The nickel-base alloys remain as essentially the only possible materials which could satisfy the high-temperature strength requirement for the primary containment. These alloys include the Inconel alloys and the Hastelloy alloys.

The Inconel alloys are the high nickel alloys such as the Inconel 600 series and the Inconel 700 series. The Inconel series are Ni-Cr-Fe alloys, with the Inconel 700 series normally having less iron than the Inconel 600 series. From the high-temperature strength standpoint, Inconel 600 (formerly Inconel) and Inconel 702 have the best high-temperature properties

in the Inconel groups. Inconel 600 contains 76% Ni while Inconel 702 contains 79.5% Ni.

The Hastelloy alloys have a lower percentage nickel content than the Inconel series. Hastelloy alloys have an Ni-Fe-Cr-Mo basic composition. Hastelloy X (57% Ni) and Hastelloy R-235 (62% Ni) have the best high-temperature properties in the Hastelloy group.

Irradiation experiments, designed to evaluate the effects of high-temperature exposure to thermal and fast neutrons, in nickel-base alloys have been completed by several investigators. ⁽²⁻⁴⁾ Table A-1 illustrates some of the published test results. Several effects can be observed from the published data on postirradiation elevated temperature testing of materials. They are as follows:

1. With the exception of Inconel 702, the elongation of the otherwise suitable alloys decreases markedly after irradiation. The alloys are therefore brittle. The order of increasing brittleness appears to be Inconel 702, Hastelloy X-280, Hastelloy R-235, and Inconel 600.
2. For similar doses, the reduction in elongation becomes greater as the irradiation temperature gets higher.
3. The tensile strength of all the alloys decreases after elevated temperature irradiation. The order of decreasing strength after equivalent doses appears to be Hastelloy R-235, Hastelloy X-280, Inconel 600, and Inconel 702.

If one were to select a suitable alloy on the basis of the unirradiated strengths at the possible extreme operation temperatures of 975 to 1100°C, the order of decreasing strength would be Hastelloy R-235, Hastelloy X-280, Inconel 600, and Inconel 702. If one considers the high-temperature test and irradiation data from Table A-1, Inconel 702 would be unacceptable because the strength would be too low. Hastelloy R-235 and Inconel 600 would be unacceptable because the induced embrittlement damage is too great to withstand any high strain condition. Therefore, Hastelloy X-280

is the obvious choice for the primary containment can. The alloy's strength is good, and elongation damage is tolerable. Joining by brazing and welding can be done satisfactorily with proper surface preparation and joint design. The only major problem is that Hastelloy X-280 is not readily available. Normally, special mill runs must be made to acquire required shapes. There is a significant lead time associated with special fabrication of tubing, sheet, and rods.

As a second choice, based essentially on availability, Inconel 600 could be used for the primary containment can. Experience has shown that Inconel 600 can withstand several in-core cycles of operations in the General Electric Test Reactor (GETR) or ETR at temperatures up to 850°C , if it is not subjected to undue thermal or rupture stresses or significant strains.

Regardless of the nickel-base alloy used in a high-temperature in-core capsule design, operating temperatures should be minimized to reduce irradiation damage. Short term exposure to high temperature and neutrons can probably be tolerated; however, efforts should be made to reduce exposure to these conditions because accelerated damage will undoubtedly occur. According to recent theories concerning irradiation damage in high-temperature alloys, the production of hydrogen and helium atoms and atom displacements can cause significant mechanical property changes in nickel-base alloys. The presence of Ni, Fe, Ti, Cr, and natural N^{14} and B^{10} in fast and thermal fluxes causes production of hydrogen and helium by (n, p) and (n, α) reactions. Only a few parts per million gas in the grain boundary and the bulk grain areas can cause a significant decrease in ductility, and exposure to high operating temperatures during irradiation probably allows gases to migrate to and agglomerate in these critical areas.

THERMOCOUPLES

From an experimental standpoint, the measurement of the specimen temperature is undoubtedly the most important information generated during

capsule operation. The control of the experimental temperature is entirely dependent upon the measured temperature. Therefore, the temperature measuring system must be as dependable as possible. Thermocouples are undoubtedly the most dependable and practical system available at this time for in-pile experiments.

Postirradiation interpretation of irradiation damage in the Advanced Beryllium Oxide Concepts structure materials is extremely dependent on irradiation temperature information. There is always a substantial uncertainty in observed temperature generated from a thermocouple in a high gamma field. Heat transfer uncertainties can also be significant due to dimensional changes and emittance changes which can occur throughout the irradiation period. These uncertainties, coupled with the basic uncertainties assumed in the heat transfer calculations, indicate that the measuring thermocouple position should be selected so that it will operate at a temperature which is as close as possible to the specimen temperature. The logical position, from a symmetry standpoint, is the radial centerline of the primary containment cell. Two thermocouples can be positioned on the centerline, one from the top and one from the bottom. These thermocouples serve as the primary thermocouples and are subject to an adverse environment and high-temperature service. Therefore, secondary thermocouples are a desirable facet of the design. Although the secondary thermocouples are subject to the uncertainties mentioned above, a correlation between the primary and secondary thermocouples during the initial phases of the experiment often allows an acceptable uncertainty in the temperature measurement and a continuation of the experiment.

The choice of the type of thermocouple for the secondary thermocouples and for the specimen thermocouples for temperatures up to 1000°C is the Chromel-Alumel type because they have a good history of emf stability in a nuclear environment.

The specimen temperature thermocouples serve as part of the primary containment; therefore, they have to be sheathed. Sheathing should also be used to protect the thermoelements from possible contamination in secondary

annulus thermocouples. The sheathing material for the Chromel-Alumel thermocouples should be the same as that used in the primary containment can, Hastelloy X-280. The MgO insulation is satisfactory for insulating material at these relatively low temperatures ($\leq 1000^{\circ}\text{C}$).

The operating temperatures of high-temperature specimens (1000 to 1800°C) exceed the operating range of Chromel-Alumel thermocouples. Therefore, higher temperature thermoelements, insulation, and sheathing must be selected. The practical thermoelement choice for high operating temperatures is restricted to the noble metal and refractory metal thermocouples. The noble metal thermocouples include (1) the Pt-Rh alloys, (2) the Ir-Rh alloys, (3) the Pt-Ir-Pd alloy, and (4) the Platinel alloys. The Pt-Rh and Ir-Rh alloys must be eliminated from consideration because of the high transmutation rate of Rh and the resulting emf decalibration. The Pt-Ir-Pd thermocouple and the Platinel thermocouple must be eliminated from consideration because of their low melting points. The refractory metal thermocouples include (1) the W-W-Re thermocouples, (2) the W-Re-W-Re thermocouples, (3) the W-Mo thermocouples, (4) the Ta-Mo thermocouple, (5) the Ta-W thermocouple, (6) the Nb-W thermocouple, (7) the Nb-Mo thermocouple, (8) the Re-W thermocouple, and (9) the Re-Mo thermocouple. The Nb-Mo, Nb-W, Ta-W, Ta-Mo, and W-Mo thermocouples are all unstable from an emf standpoint and have polarity reversals in the temperature region of interest. The high rhenium Re-Mo and Re-W thermocouples would be very unstable in a neutron environment due to the rapid transmutation of rhenium.

The W-W-Re and W-Re-W-Re alloy thermocouples remain as the only ones which are likely candidates for use at high temperature in a neutron environment. These are the W-W-26% Re, the W-5% Re-W-26% Re, and the W-3% Re-W-25% Re thermocouples. All three types have a high emf output which does not have a polarity reversal. All three are reasonably stable at high temperatures and in a neutron environment. The W-W-26% Re combination can generally be eliminated because of the brittle character of the "W" leg. The choice between the W-5% Re-W-26% Re and the

W-3% Re-W-25% Re thermocouples is difficult because experience has shown that both can operate satisfactorily or unsatisfactorily, seemingly by chance. A long term out-of-pile test is presently underway at General Atomic to evaluate the two thermocouples. Until the results of this test are analyzed, either type of W-Re-W-Re alloy thermocouples could be considered satisfactory for high-temperature in-pile service.

The sheath of the high-temperature thermocouple assembly serves as a portion of the primary containment and protects the thermoelements from contamination originating from outside the assembly. Therefore, it must remain integral throughout the irradiation.

The severe operating temperature of the thermocouples restricts the possible sheath materials to the refractory metals (i. e., Nb, W, Ta, Mo, and Re) and alloys of these metals. The sheath will be in contact with graphite, BeO, and insulating material. Possible insulating materials include BeO, ThO₂, and Y₂O₃. Obviously, none of the refractory metals will be compatible with graphite at 1700°C. Therefore, it becomes necessary to protect the sheath from the graphite and to select a sheath material which will form a protective carbide layer on its exterior surface. Niobium, tantalum, tungsten, and molybdenum all form protective carbide layers.⁽⁵⁾ Niobium, however, reacts with BeO at ~1500°C⁽⁶⁾ and therefore cannot be considered. Tungsten is too brittle to cold-work, and during the construction of the capsule, cold-working must be accomplished. However, low gas-impurity tungsten tubing is now available which may have sufficient ductility to be worth investigation. The Mo-Re and W-Re alloys may also be useful for sheathing material; however, they are brittle, and many modifications would have to be done on them before they could be used for thermocouples which have to be cold-worked during capsule fabrication.

Tantalum and molybdenum form integral protective carbide layers. Neither reacts with BeO up to 2000°C.⁽⁶⁾ The cold-work in tantalum is less deleterious than in molybdenum for equal strain and is therefore the choice for the sheath material.

The tantalum sheathed thermocouples must be protected from carburization while operating in the capsule. This can be accomplished by providing a diffusion barrier and a vapor transfer barrier between the exterior sheath surface and the specimens. The vapor barrier is a gas gap, while the diffusion barrier is a sacrificial 0.002-in. -thick tantalum tube. The tantalum tube will carburize to TaC and Ta₂C, forming a diffusion barrier to carbon atoms. Figure A-2 illustrates a Ta sheathed, BeO insulated W-5% Re-W-26% Re thermocouple after operation for 520 hr at 1500°C in the GETR. Figure A-3 illustrates an area of the sacrificial sheath showing Ta₂C in the grain boundaries. Figure A-4 illustrates what can happen to a tantalum sheathed thermocouple after 525 hr at 1700°C, out-of-pile, in contact with graphite. The outside layer is TaC, the center layer, Ta₂C, and the inner layer, tantalum.

The insulating material must be compatible with tantalum and the W-Re thermoelements. It must also serve as a high-temperature insulator. The order of decreasing insulation value at temperatures up to 2480°C is generally BeO, ThO₂, (HfO₂, Y₂O₃, ZrO₂), and MgO.⁽⁷⁾ The order of decreasing initial reaction temperatures of the oxide insulators with tantalum is ThO₂ (2795°C), BeO (2245°C), ZrO₂ (1955°C), MgO (1790°C), and Y₂O₃ (1705°C). When one considers the insulating value, compatibility, and γ -heating rate (causing increased heating in the thermocouple) in the thermocouple assembly, BeO is the best material for the insulation. Figure A-5 illustrates the lack of interaction between tantalum and BeO after 156 hr at 1850°C, out-of-pile. Figure A-6 illustrates the interaction between Y₂O₃ and tantalum after 156 hr at 1850°C. Note the formation of a liquid phase and dendrites in the Y₂O₃.

Until further improvements are made in the dependability of thermocouple materials, the combination of W-Re-W-Re thermoelements, BeO insulation, and tantalum sheath is the best choice for the high-temperature, in-pile measurement of temperature with the thermocouple in contact with graphite.

M-159-71

(50×)

Fig. A-2--Transverse section of irradiated tantalum sheathed, BeO insulated, W-5% Re-W-26% Re thermocouple and protection tube after 520 hr at 1500°C

M-159-73

(600×)

Fig. A-3--Two-mil tantalum sacrificial sheath after 520 hr at 1500°C in neutron environment of $\sim 5 \times 10^{20}$ nvt

M-10525-1

(135×)

Fig. A-4--Transverse section showing carburization of tantalum thermocouple sheath after 525 hr at 1700°C

M-10652-18

(70×)

Fig. A-5--Lack of interaction between BeO and tantalum after 156 hr at 1850°C, out-of-pile

M-10652-1

(70×)

Fig. A-6--Reaction between Y_2O_3 and tantalum after
156 hr at $1850^{\circ}C$, out-of-pile

In general, compatibility interactions between the sheath, insulation, and thermoelements will increase in number and severity with increasing contaminants in the thermocouple assembly. The high temperature resistivity of the insulation will also decrease with increasing thermocouple assembly contamination. Obviously, the assemblies should be fabricated from the cleanest and purest materials available. Extreme care should be taken to maintain this high purity during manufacture. The high-temperature thermocouples used in General Atomic irradiation capsules are fabricated from reactor grade tantalum tubing and 99.9+% pure BeO in a purified argon atmosphere with water and oxygen levels of less than 10 ppm and 2.5 ppm, respectively. All manufacturing steps are required to be accomplished in the purified argon. Tests are currently under way to confirm that the high purity thermocouples are more reliable than the normal high-temperature thermocouples for nuclear use.

METHODS OF JOINING

The methods of joining the parts of the capsule, particularly the primary containment parts, are critical. No leaks through welded or brazed areas can be tolerated either in the primary or secondary containments. Primary leaks would result in the loss of temperature control, and secondary leaks would result in water or high pressure helium contamination.

The joints between parts can be made by welding or brazing. The like materials can be welded, with the exception of the thin-walled thermocouples. The Hastelloy X-280 cans and end caps can be welded satisfactorily, providing the proper precautions are taken. Hastelloy X-280 is a "hot short" material; i.e., it is susceptible to root cracks in the weld area. The cracks can be avoided by joining the thin parts and providing an ample supply of base metal in the welded area. The thin parts allow the material in the area adjacent to the weld to deform enough to prevent the high stresses of freezing from initiating root cracks. The large supply of base metal can be melted down to prevent any reduced area in the weld. (8)

Proper design of the weld is necessary to reduce the chances of cracking. The parts to be welded should fit as tightly together as possible. They should also be of approximately the same thickness so that they will deform relatively uniformly during the critical freezing time. Wall thickness in Hastelloy X-280 in the 0.010- to 0.030-in. range can be easily and satisfactorily welded.

The dissimilar materials and the thin walled thermocouples should be brazed. The gas tubes can be welded and brazed. The gas tube braze acts both as a secondary seal and as a stabilizer to prevent bending in the weld area. This type of joint is illustrated in Fig. A-7.

The braze material for joining Hastelloy X-280, Inconel 600, and tantalum must of necessity be a high-temperature braze. The brazed area may operate at temperatures as high as 1800°F. In an effort to determine the most suitable braze material, tests using Nicrobraz 30, a Nicro braze, a Cu-Ti braze, and an OFHC copper braze were conducted.

Figures A-8 and A-9 illustrate typical areas in a tantalum-Nicrobraz 30 braze after ten thermal cycles to 1500°F. The Nicrobraz 30 is two phase and extremely brittle. In some areas, cracks originating in the braze penetrate the tantalum thermocouple sheath.

Figures A-10, A-11, and A-12 illustrate the conditions of Inconel-Nicro and tantalum-Nicro brazes. The braze material reacts with both the tantalum and the Inconel. The tantalum-Nicro interaction is particularly severe at the top of the braze fillet. Large amounts of porosity are evident in the Inconel-Nicro braze, especially in the core of the capillary.

The Cu-Ti braze material reacts vigorously with Inconel and slightly with tantalum, as depicted in Figs. A-13 and A-14. In the case of Inconel, the reaction product layer is evidently a higher melting product which bridges the capillary to stop the flow of the braze material down the capillary.

The OFHC copper braze reacts only slightly with Inconel 600, Hastelloy X-280, as shown in Figs. A-15 and A-16. It wets both of them very well and adheres well if the surfaces are properly prepared

Table A-1
TEST RESULTS OF IRRADIATION EXPERIMENTS

Alloy	Irradiation Dose ($E > 1$ MeV) (nvt)	Irradiation Temp. (°C)	Test Temp. (°C)	Tensile Stress (1000 psi)	Yield Stress (1000 psi)	Elongation (%)
Inconel 600 ^a	Unirradiated	Unirradiated	700	46	28	29
Inconel 600 ^a	1.2×10^{20}	650	700	33	32	3
Inconel 600 ^b	4×10^{20}	370-425	593	--	34	6
Inconel 702 ^a	Unirradiated	Unirradiated	700	95	87	5
Inconel 702 ^a	1.2×10^{20}	650	700	21	11	17
Hastelloy R-235 ^a	Unirradiated	Unirradiated	700	127	85	19
Hastelloy R-235 ^a	1.2×10^{20}	650	700	88	87	4
Hastelloy X-280 ^a	Unirradiated	Unirradiated	700	72	36	48
Hastelloy X-280 ^a	1.2×10^{20}	650	700	58	40	9
Hastelloy X-280 ^c	1.57×10^{20}	280	650	52	26	17
Hastelloy X-280 ^c	1.97×10^{20}	740	650	67	34	6
Hastelloy X ^a	4×10^{20}	370-425	593	--	42	13

^a Data from Ref. 2.

^b Data from Ref. 4.

^c Data from Ref. 3.

Fig. A-7 Weld braze gas tube penetration design for
primary containment

M-10087-1

(400×)

Fig. A-8--Crack, originating in the Microbrazed 30 fillet, penetrating into tantalum sheath of a high temperature thermocouple

M-10387-1

(135×)

Fig. A-9--Cracks in Microbrazed 30 fillet on tantalum after 11 cycles from room temperature to 1500°F

M-10301-1

(135x)

Fig. A-10--Ni-oro reaction with tantalum

M-10301-4

(135x)

Fig. A-11--Ni-oro reaction with Inconel 600 and tantalum (etched)

M-10301-5

(135×)

Fig. A-12--Porosity in Nicro braze in contact with
Inconel 600 (etched)

M-10390-1

(135×)

Fig. A-13--Reaction stopping capillary flow in
Inconel 600 - Cu-Ti braze

M-1390-4

(135×)

Fig. A-14--Reaction of Cu-Ti with Inconel 600 and tantalum (etched)

M-10300-2

(270×)

Fig. A-15--Tantalum-copper-Inconel 600 braze in capillary area

M-14898-1

(100×)

Fig. A-16--Inconel 600 tube brazed to Hastelloy X-280
end cap with OFHC copper

before brazing and a vacuum is maintained at $<5 \times 10^{-5}$ torr during brazing. The OFHC copper braze is ductile and less susceptible to cracking and irradiation damage than the other brazes that were investigated. For the irradiation capsule of interest, it is obviously the best choice for brazing Inconel 600, Hastelloy X-280, and tantalum.

A-IV. SUMMARY

The following materials should be used for a high-temperature irradiation capsule for operation in a significant fast flux:

1. The secondary containment should be fabricated from a stabilized stainless steel.
2. The thermal spacer should be made from aluminum alloy 6061.
3. The primary containment alloy should be Hastelloy X-280. If Hastelloy X-280 is not available and the containment is not subject to large strains, Inconel 600 would serve satisfactorily.
4. The controlling high-temperature thermocouples should be fabricated from either W-3% Re-W-25% Re or W-5% Re-W-26% Re thermoelements, BeO insulation, and tantalum sheaths. (Care should be taken to maintain a high degree of cleanliness and base material purity.)
5. The material joints should be TIG welded where possible and OFHC copper vacuum brazed where joints cannot be properly welded.

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SECTION B

NOVEL DESIGN FEATURES IN A BROAD TEMPERATURE- CONTROLLED IRRADIATION CAPSULE (FUELED)

B-I. INTRODUCTION

This paper describes the design features of a class of irradiation capsules developed at General Atomic to support current research programs directed toward the development of coated fuel particles for use in power reactors. Recent operating experience in the Engineering Test Reactor (ETR) is presented for the initial capsule of this design, and comparisons between predicted and actual performance are made.

Fuel used in General Atomic's HTGR is in the form of thorium-uranium particles coated with various layers of graphite. The irradiation capsules described in this paper are used to test a large number of coated particles with varying coating parameters (1) to a high fast flux exposure, (2) at an accelerated burnup rate (as compared to service), and (3) at a stable irradiation temperature (1400°C). The requirement that a high fast flux exposure be obtained within a reasonable time period was the primary reason for choosing the ETR as the test facility. Choice of the ETR for this series of irradiations presented problems in meeting the criteria of a stable irradiation temperature and the desired accelerated rate of burnup because of the well-known unpredictability of thermal neutron fluxes in the ETR core positions. These fluxes routinely vary within a span of $\pm 30\%$ from cycle to cycle in the I-13 core position used by General Atomic (a recent flux change of $+67\%$ was noted in this position). Burnup rates in our particles are established based on fissions per initial heavy metal atom (FIMA). Once the thorium-uranium ratio has been established for a given capsule fuel loading, no further alteration is possible to accommodate flux

changes. Uncertainty in flux is therefore reflected directly in uncertainty in burnup. Irradiation temperatures, since they are a function of capsule power generation, are very flux-dependent as well. It is our desire that the design irradiation temperature of the particles be maintained throughout the irradiation period within 20°C. An added complication to this aim is the reduction in heat generation rate due to burnup as the irradiation progresses. A means has been successfully employed to overcome the problem of maintaining a stable irradiation temperature through use of a dual binary gas control system. Succeeding paragraphs describe in detail the design of the capsule and the operating experience for three cycles of irradiation in the ETR.

B-II. BASIC DESIGN

The irradiation position for which this capsule was designed is the I-13NE core position of the ETR. The capsule used an X-basket which accepts a capsule diameter of 1.125 in. Since gas must be supplied to the capsule and data must be obtained via thermocouple measurements during irradiation, a lead-type capsule is required. Figure B-1 shows the essential parts of the capsule and a portion of the lead tube. The three major areas of the experiment are the lead tube, the secondary containment, and the primary containment, each in a separate environment. The lead tube portion of the experiment is that section above the outer end cap. All of the gas lines and thermocouples from the capsule itself pass through the lead tube, which is also equipped with a purge and pressurization line that supplies helium to maintain the lead tube pressure (somewhat above reactor water pressure, in this case about 225 psig). The lead tube is fabricated from Type 304 stainless steel. This tube terminates at the outer end cap, from which the capsule outer can, another Type 304 stainless steel tube, originates. The portion of the experiment below the outer end cap, consisting of the primary and secondary containments, is considered the capsule.

A gas-tight seal is maintained between the lead tube and the capsule environment by OFHC copper-brazing the gas tubes and thermocouples to the outer end cap. The pressure environment of the capsule is approximately 10 psig. The primary containment of the capsule is formed by an Inconel 600 primary can, within which are contained five graphite crucibles. Each crucible contains from one to three coated particle samples. These samples are contained in graphite split thimbles and consist of as many as 8,000 coated particles. Actual volume occupied by the particles is small, since the largest particles are typically about 500 μ in diameter (a more detailed description of the crucible design is presented in a later section). The Inconel primary can is sealed by Inconel end caps. These end caps are welded to the can, and thermocouple and gas tube penetrations into the primary can are sealed by OFHC copper braze. A separate mixed gas inlet and exit tube provides the primary containment with its own gas control system. A controlled gas gap for radial heat transfer is provided between the o. d. of the graphite crucible and the i. d. of the Inconel can.

The primary can is contained within the outer stainless steel capsule can. The volume below the outer end cap, interior to the outer capsule can but exterior to the primary can, is referred to as the secondary containment. The space between the o. d. of the primary can and the i. d. of the outer capsule can is occupied by an aluminum thermal bond. The function of this thermal bond is to provide a heat transfer medium with a controlled gas gap between the Inconel and stainless steel cans, and to act as a carrier for gas tubes and thermocouples. The secondary containment is provided with a gas inlet tube just below the outer end cap, and an exhaust tube at the bottom of the capsule. This tube routes the secondary exhaust gas through the aluminum thermal bond, up through the lead tube, and ultimately to the ETR stack.

B-III . TEMPERATURE CONTROL SYSTEM

It has been pointed out that one of the design criteria for this capsule was a stable irradiation temperature. Since a thermal flux variance of $\pm 30\%$ was expected, any successful temperature control system would have to accommodate a minimum of a 30% power change in either direction. This must be regarded as a minimum, since the gamma heating level in the experimental position will vary as well. In addition, and of perhaps greater significance, a percentage of the control range provided in any capsule will be spent overcoming the effects of machining tolerances and material emissivity uncertainties on heat transfer. For a fueled capsule such as this one, a further uncertainty results from inevitable inconsistencies in fuel loading. To round out possible sources of error, mention must be made of uncertainties in flux perturbation effects on the actual heat generation in the fuel. In the face of the sum total of these uncertainties, one can only hope that some of them will cancel the effects of others and concentrate on providing the maximum possible control range to take care of the remainder. In practice, of course, this is exactly what happens, though the remainder is generally substantial.

The means chosen for temperature control of this capsule was a dual binary gas system. It is a dual system because separate gas systems are used for primary and secondary containments. The gases chosen were helium and argon, since these are inert gases and differ significantly in thermal conductivity in the range of interest. The induced activity in the argon presents little difficulty, since it is easily handled by routing the gas through an activated charcoal delay trap prior to releasing it to the ETR stack system.

The capsule is designed so that a minimal amount of heat is transferred from the fuel by axial conduction. This is accomplished by radiation shields and large axial gas gaps. The majority of the heat transferred is by radial conduction and radiation. The two control gas gaps of significance

Fig. B-1--Fuel particles irradiation capsule assembly

are the 3-mil gap between the o. d. of the graphite crucible and the i. d. of the Inconel primary can, and the 6-mil gap between the o. d. of the Inconel primary can and the i. d. of the aluminum thermal bond. The primary gap is maintained by four spacer knobs located diametrically on the upper and lower ends of each crucible. The secondary gas gap is maintained by spot-welding spacer wires on the outside of the Inconel can. All other interfaces are contact resistances and are so treated in the analyses.

Use of two separate control gas gaps and gas systems may seem of questionable utility, since one such gap could be provided and an equivalent heat transfer system obtained. There are several important reasons for providing two gaps. As will be seen, use of this temperature control system provides an extremely broad control range. The theoretical control range for the particular capsule under discussion was approximately $\pm 450^{\circ}\text{C}$. If a single gap were used to provide this control, an overtemperature condition that triggered an automatic cooling of the capsule (typically by means of a high-pressure helium dump system) would result in a severe thermal shock to the capsule. This is an undesirable event, since such shocks are the most common cause of thermocouple failure. By adjusting the gas mixture in the secondary containment to provide a specified cooling effect and routinely using only the secondary annulus for the helium dump, such severe thermal shocks can be minimized. Another reason for providing the dual system was that future (now current) designs were to incorporate individual crucible gas control. The system described here provides a single control capability for all five crucibles. It was thought initially that a gas mixture could be locked in the primary containment, with only the secondary system flowing, but this proved impractical because of stratification.

If it were not possible to flow gas over the fuel samples (because the data might be compromised by the effects of gas impurities), the control range of the capsule would be considerably reduced. To realize the maximum effect from a change of gas composition (i. e., thermal conductivity), this

change should take place in the region with the highest available temperature, so that the maximum ΔT will be effected. If this were done entirely outside the primary can, the can wall temperature would be a limitation on the control range. With a gas gap inside the primary can, ΔT changes obtained by the alteration of the primary gas mixture do not affect the can wall temperature, and it is not a limit on the obtainable fuel temperature.

Table B-1 lists both theoretical and actual thermal performance data for GA5-P13E capsule described in this paper. In interpreting these data, no direct comparison between theoretical and actual temperatures should be drawn because of the uncertainty in actual fission heat generated at the time the data were recorded. It is significant to note that the control range indicated in the theoretical calculations was roughly $\pm 450^{\circ}\text{C}$ from the desired temperature. The actual temperatures obtained are somewhat lower than those predicted. However, the maximum measured temperature (1483°C) was probably not the maximum obtainable. A further increase in temperature could undoubtedly have been effected had this been desired. Note that the actual control range obtained was 745°C , 82% of the theoretical range.

Table B-2 presents the results of a series of low power tests which were carried out during the initial startup of the capsule. Note that at this low power level it was possible to increase the crucible thermocouple temperature by 59% by changing the gas in the secondary system from helium to argon. Use of argon in the primary system and helium in the secondary caused an increase of 325°C or 126%. When argon was used in both systems an increase of 464°C or 180% resulted.

All of these tests were a significant demonstration of the capabilities of the dual gas control system. In the particular capsule described here, temperature differences of several hundred degrees centigrade existed among the five crucibles. This was the result of a combination of factors (chiefly, flux differences along the test length). Current capsules are being designed with each crucible in a separate primary can so that individual temperature control can be provided.

Table B-1

FULL POWER DATA
(Data for Crucible No. 2 at 175 MW)

Theoretical Fission Heat ^a (watts)	Gas Mixture (% by Conductivity)		Fuel Temperature (°C)	
	Primary	Secondary	Theoretical	Actual
650	100 He	100 He	926	738
650	50 He, 50 Ar	50 He, 50 Ar	1370	---
650	100 Ar	100 Ar	1665	1483 ^b

^a Actual fission heat at startup unknown.

^b This temperature represents a maximum attained, but a higher temperature probably could have been reached.

Table B-2

LOW POWER TESTS
(Data for Crucible No. 2 at 40 MW)

Primary Gas	Secondary Gas	Crucible Thermocouple Reading (°C)
He	He	257
He	Ar	410
Ar	He	582
Ar	Ar	721

The GA5-P13E capsule was irradiated for three ETR cycles. Desired temperatures were maintained throughout this period without difficulty. Similar tests carried out April 1966 on two current capsules at the ETR produced equally dramatic results. Replacing the helium in both control gas annuli with argon at the low reactor power level of 40 MW produced an increase in temperature of over 620°C, from 457°C to 1080°C, in a single cell. Similar results were obtained for the other cells (current designs, as indicated previously, incorporate individual cell control).

B-IV. TEMPERATURE MONITORING SYSTEM

Temperature data acquisition is one of the most important functions of the capsule. Irradiation temperature is a significant parameter affecting the performance of the coated particles in the reactor environment. In the capsule described in this paper, one high-temperature W-5% Re-W-26% Re thermocouple is provided for each of the five crucibles. In the two outermost crucibles, the thermocouple is in the center of the single fuel thimble; in the center three crucibles, it is in the center of one of the three fuel thimbles. A detailed description of the relationship of the thermocouples to the fuel is presented in the next section. The hot junction of the thermocouple is positioned in the center of the fuel thimble and will sense the maximum fuel temperature, or very close to the maximum. Figure B-2 illustrates the placement of the thermocouple in the fuel. The tungsten-rhenium thermocouple is BeO-insulated within the fueled region. The sheathing material is tantalum. Since the thermocouple is to be inserted in a graphite tube in the center of the fuel, as shown in Fig. B-2, a sacrificial tantalum sheath is first placed in the tube so that it will be selectively carburized. A gap is maintained between the two sheaths by wrapping the thermocouple sheath with tantalum wire prior to inserting it in the tube.

To provide a certain amount of redundancy in temperature control of the experiment, seven Chromel-Alumel thermocouples are provided as backups to the five tungsten-rhenium thermocouples in the fuel. Five of the Chromel-Alumel thermocouples "float" in slots in the aluminum thermal bond, one opposite each of the high-temperature thermocouples. The remaining two are spot-welded to the o. d. of the Inconel primary can. The chances of the backup thermocouples surviving a long-term irradiation are much greater than those of the fuel thermocouples both because the Chromel-Alumel thermocouples are inherently more reliable in a reactor environment and because their temperature of operation is considerably

Fig. B-2--Detail design of specimen crucible

lower. The fact that the backup thermocouples operate at much lower temperatures than the fuel thermocouples lessens their effectiveness, since it is extremely risky to attempt to extrapolate through ΔT 's of 1000°C , especially if gas composition changes are made between the two points. Attachment of the thermocouples to the primary can wall improves this situation considerably and provides for monitoring of primary can temperatures as well. Current practice is to attach the backup thermocouples to the cans, using a nickel-palladium braze technique.

B-V. CRUCIBLE DESIGN

Figure B-3 is a photograph of the graphite crucible and allied components. The crucible shown is of the most typical design, with three holes for fuel compacts. The other smaller holes are used for "piggyback" experiments which take advantage of the high irradiation temperature available in the crucible. The large holes contain the graphite split thimbles. These thimbles are drilled in the center, as shown in the figure, for most of their length. A smaller drill is used to continue the basic thimble hole a short distance beyond the terminus of the larger diameter. A graphite tube is placed in the center of the thimble. This tube seats in the smaller drilled hole in the lower portion of the thimble. When the thimble and tube have been inserted in the crucible, the fuel is poured into the annular space between the o. d. of the central tube and the thimble i. d. Provision for differing fuel loadings is made by varying the wall thickness of the thimble, which varies the annular volume. After the fuel has been loaded in the thimble, an annular plug is inserted in the top of the thimble to prevent loss of any particles. After all three thimbles have been loaded, a second retaining plug is placed above each thimble, and the graphite cap is screwed on so that it is impossible for the fuel to fall out of the thimbles. After inserting the sacrificial tantalum liner in the graphite tube, the fuel thermocouple is inserted in the tube in the center of one of the thimbles.

4.2.40

Figure B - 3 Specimen Container

If the other fuel thimbles are not provided with thermocouples, a graphite rod is inserted in the center tube.

Hot cell disassembly and removal of the coated fuel particles was a major consideration in the crucible design. In the basic crucible below the thimble, a hole of smaller diameter than the hole into which the thimble fits allows a rod to be inserted (after removal of the screw-on crucible cap) to push the individual thimbles out of the crucible after irradiation. When the split thimble is pushed out, the two halves fall apart and the particles are immediately exposed without the application of any force to the particles themselves. This is an important consideration, since the purpose of the experiment is to evaluate only irradiation damage.

The screw-on cap provides positive closure both prior to and during irradiation for the particles in the crucible; yet, it has presented no problems in remote disassembly after irradiation.

B-VI. CURRENT DESIGNS

The irradiation capsule described in this paper operated successfully for three cycles in the ETR, to a maximum fast flux exposure of 2.7×10^{21} nvt (>1 MeV).

Current designs are very similar to the capsule discussed, with the important difference that individual crucible temperature control and a fuel thermocouple in each fuel thimble are provided. In essence, the capsule is the same as the one described, but instead of five crucibles in a single primary containment, there are five primary containments, each containing a single crucible, and the temperature of each may be individually controlled.

Calculations for the five-cell capsules currently in use indicate that a power swing of $\pm 90\%$ can be accommodated by the dual binary gas control system. A theoretical temperature control range in excess of 1220°C may be obtained at design power.

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SECTION C
CAPSULE IMPROVEMENT AT GENERAL ATOMIC

C-I. INTRODUCTION

As in any capsule irradiation program, there are areas in capsule design and fabrication where problems arise. Hopefully, most of these problems are resolved prior to the irradiation test, but generally there are some which result from these in-pile tests. This section of General Atomic's contribution relates some of the experiences associated with problem areas in our specific type of capsule design. The areas of interest include approaches taken to solve problems or improve techniques in material compatibility, brazing, thermometry, dosimetry, and instrumentation.

It is very important to reemphasize that these approaches or improvements were guided by our requirements to irradiate fueled and unfueled material at high temperatures, to measure these fuel or material temperatures as accurately as possible, to provide a constant temperature of irradiation, and to maintain the integrity of the test materials until irradiation results could be determined. The previous presentations have generally described the uniqueness of design and the materials of construction for such a particular type of capsule. In general, where designs or material for construction were questionable from a usefulness or compatibility standpoint, analytical or laboratory tests were conducted to establish feasibility. This approach helps to strengthen the capsules because only those changes in the design or construction which are proven to be advantageous are made.

C-II. MATERIALS COMPATIBILITY

An important feature of our capsule design is the measurement of the operating temperature of fueled or nonfueled materials during irradiation exposure, with a high-temperature thermocouple. This necessitated penetration of the primary containment can and placement of the thermocouple in the hot region. A tantalum sheathed, W-5% Re-W-26% Re, BeO insulated thermocouple was selected to measure this temperature ($\sim 1400^{\circ}\text{C}$). The original design placed the thermocouple in a hole directly in the center of a graphite crucible for the fuel particle capsules. Thermocouples of this design did not last long, and it was believed that the tantalum sheath and thermocouple wires at the hot junction had carburized and perhaps caused fractures to occur in the thermocouple. Improvement in the design was accomplished by a broadly spaced spiral wrap of 0.005-in. tantalum wire, spot-welded in a few places, around the thermocouple sheath. In addition, a 0.002-in. tantalum foil was inserted in the hole in the graphite to serve as a barrier to carburization of the tantalum wire wrap. As noted in the previous presentation, this technique was very successful, with no deleterious effects for either the thermocouple or the fuel particles. Other materials compatibility problem areas are discussed later in this paper.

C-III. BRAZING

Three areas of improvement have been accomplished in association with this topic. These include the brazing alloy, the method of brazing, and the brazing of backup thermocouples to the primary cans to increase thermal measuring effectiveness.

The selection of a braze material in the capsule construction is very important because the braze serves not only as a mechanical connection but also as a hermetic seal to contain gases, either control gases or fission product gases. A braze is also necessary to seal the multileads

of gas tubes and thermocouples penetrating the end caps for an instrumented capsule. A high-temperature stainless steel type braze, Nicrobraz 30, had served well with our capsules until dissimilar metals, such as the tantalum on the tantalum sheathed high-temperature thermocouples, were interjected into the design. Leaks in previous capsules and a laboratory investigation of a brazed mockup of the braze joint indicated a problem area existed. The high-temperature Nicrobraz 30 was too brittle. Cracks developed in the braze area because of thermal cycling effects and differential thermal expansion mismatch of the different materials. It was therefore necessary to investigate a braze material that would be quite ductile and yet compatible with the different materials to be brazed.

The materials of construction were tantalum, Inconel 600, and stainless steel Type 304. The high-temperature brazes investigated were Nicrobraz 30 (Ni-Cr-Si), Nioro (Ni-Au), copper-titanium, and copper. The tests included a mockup of the braze junction, any thermal treatments required, and a thermal cycling test to 1500°F with 11 cycles. Metallurgical evaluation of these brazed joints revealed the following:

1. Nicrobraz 30 did not produce a nonreactive ductile braze between tantalum-sheathed thermocouples and Inconel end caps. Cracks were evident in the braze area and were further increased and magnified by both thermal soak treatments and thermal cycling tests.
2. The Nioro braze reacted with the tantalum-sheathed thermocouples and produced porosity within the brazed area.
3. The copper-titanium braze reacted with the tantalum and did not appear to penetrate completely along the joint region. Several areas also revealed cracking and porosity.
4. The copper braze was excellent both before and after thermal cycling. No apparent reaction, cracking, or porosity was noted.

The method of brazing is important because of the many penetrations through an end cap. Induction brazing has been the standard method used. It was noted, however, that when multipenetrations of gas tubes and thermocouples were brazed by this method the braze temperature was not uniform across the end cap. This was the result of two conditions. First, the frequency of induction is 450 kc, which leads to skin heating effects and produces a radial temperature gradient. Second, because of the different materials in the braze area, stainless steel, Inconel, and tantalum, the mass of each material, and the conduction of the heat along the tube walls, the melting points of the braze materials were not reached concurrently. As the result of these conditions, the braze material tended to flow toward the hotter temperature, thereby pulling braze material toward the edge of the end cap and away from the braze joint. Also, the brazing of the last joint resulted in an overtemperature in the braze area for those joints that melted first. This higher temperature exposure is deleterious to the joint.

The above problem was corrected to some extent by using a radiation heat susceptor. The inductive fields heat this susceptor, which radiantly heats the area to be brazed and tends to shield the end cap and penetrating leads from the inductive currents. A more satisfactory solution was to braze these multileads in an end cap by vacuum resistance furnace brazing. Hydrogen furnace brazing could not be used for the copper brazing operation because of the effects of the hydrogen environment on the tantalum.

The third improvement associated with brazing involved the brazing of backup thermocouples to the sides of the primary can to provide a good thermal bond and thereby measure the primary can temperature. This feature is important to the safety of the experiment, first, because the operating temperature of the primary containment can is known, and, second, because it provides a good thermal detector backup to the fuel-measuring thermocouples once the initial relationship of temperature is established. Actually, two different techniques, welding and brazing, were used to attempt to make the metallurgical thermal bond. Welding the backup thermocouples

(Inconel-sheathed Chromel-Alumel thermocouples) to the primary can was successful, but metallurgical analysis indicated large grain size in the weld area. Brazing of the thermocouples proved very satisfactory. The braze alloy used was a eutectic nickel-palladium alloy, and the braze was accomplished by electron beam heating. Metallurgical investigation of the braze joint indicated no increase in grain size in the can or thermocouple.

C-IV. THERMOMETRY

One of the most important measurements when studying the high-temperature irradiation quality of a material is the precise measurement of the temperature in the fueled or nonfueled specimens during the irradiation tests. The selection of a proper thermocouple to make this measurement was based on the temperature of interest (1000°C to 1800°C), the compatibility of the thermocouple materials with their environment, and the commercial availability of a metallic sheath which could serve as a barrier or containment. We have been totally dependent on vendors to supply us with these high-temperature metallic sheathed thermocouples, and our experience has been generally poor. For this reason, we established some rigid pretesting procedures for the selection of thermocouples acceptable for incorporation into the capsules. This testing is done in our Capsule Irradiation Group. The tests include:

1. Gaging for proper dimensions.
2. Leak checking.
3. Electrical continuity measurements.
4. Resistivity measurements.
5. Insulator breakdown tests.
6. Calibration.
7. Thermal shock tests.

Compatibility tests were also initiated between the tantalum-sheathed, BeO-insulated, W-5% Re-W-26% Re type thermocouple as used initially in

our capsules and graphite. Some of these compatibility tests have been reported earlier. The results of these experimental tests assisted in the preparation of a rigid set of thermocouple specifications which we use to purchase our thermocouples and inspired the unique design as noted in the section on materials compatibility.

As mentioned above, these thermocouples must penetrate the primary fission product containment vessel in a fueled capsule. Therefore, the sheath material must serve as part of the primary containment vessel. In the fuel particle capsules, if this sheath breaks, fission product gases can enter and migrate along the inside of the thermocouples. Also, when the lead tube is cut after irradiation, these fission products can be released and, more damaging to our specific type of fuel, water or air can migrate back through the thermocouple. To prevent this occurrence a thermocouple sealoff junction was constructed. This sealoff is illustrated in Fig. C-1. The junction is constructed between the high-temperature thermocouple wires and the sheathed compensated lead wire. A cylindrical tube of silver solder is placed inside the junction and is not used until a seal is required after the irradiation test is complete. The desired sealoff operation is accomplished by a submersible furnace that heats up a section of the lead tube below the cutoff area and melts the sealoff assemblies. A sealoff assembly very similar to the thermocouple sealoff is used for all gas tubes.

As previously mentioned, thermal knowledge from the low-temperature Chromel-Alumel thermocouples outside the primary can was improved greatly by electron-beam brazing these thermocouples to the outside of the primary can. These thermocouples were placed in the transverse plane of the high-temperature thermocouple.

C-V. CAPSULE DOSIMETRY

The Capsule Irradiation Group, in cooperation with the Chemistry Department, is attempting to upgrade the accuracy and reliability of

FIG. C-1
THERMOCOUPLE SEAL-OFF ASSEMBLY

determining the neutron flux received by materials in irradiation capsules. The dosimeter wires that are presently used include a nickel-cobalt alloy, pure iron, and titanium. A problem with these dosimeters is that they monitor fast fluxes above the 3.0 MeV energies, whereas the majority of the neutron damage in materials is effected by neutron energies in the 0.1- to 3.0-MeV range. To determine these lower energy values, the neutron flux data as measured from the dosimeter wires must be extrapolated to these lower values. This extrapolation or correlation is poor. Further, error is compounded in the Ni-Co alloy because of the chemical separation necessary before counting techniques are applied. For the iron dosimeter, the influence of Fe^{59} affects the counting efficiency of the Mn^{54} formed from the Fe^{54} isotope.

Our approach is to upgrade the presently used dosimeters. For the fast flux problem, we have alloyed 0.10% Fe^{54} with vanadium. The Fe^{54} was alloyed with the vanadium to overcome the influence of Fe^{59} in the counting technique. The Fe^{54} was obtained from Oak Ridge National Laboratory and alloyed with high purity vanadium in the form of wire by the Stellite Division of Union Carbide.

To replace the Ni-Co alloy, a cobalt-vanadium alloy was compounded. The Co^{59} in vanadium will allow direct counting of Co^{60} without the chemical separation process required for the nickel alloy. Again, this reduces the timely and expensive separation cost and increases counting efficiency. The vanadium, in both cases, serves as a useful thermal neutron flux monitor for long-term irradiations, $>10^{21}$ nvt.

These dosimeters, along with the regular type dosimeter wires, were inserted into our most recent capsules presently undergoing irradiation tests. Results from these tests are not available at the writing of this paper.

C-VI. INSTRUMENTATION

The instrumentation for the General Atomic multigas-controlled, temperature-monitored irradiation capsule is the latest design that includes our own innovations as well as those required by the Engineering Test Reactor (ETR). This instrumentation is illustrated in Figs. C-2 and C-3. For discussion purposes, the instrumentation is separated into four groups: the gas purification system, the gas control system, the temperature monitoring system, and the main power and alarm system. The arrangement of these functional groups was designed to achieve ease of operation and installation, as well as safety.

The gas purification system accepts helium and argon gases from a supply, purifies each gas separately, and distributes it to each of the gas control systems. The gases are initially passed through two separate dehydrfilters and then enter a hot chip furnace. The active section of the hot chip furnace is composed of a titanium-uranium-titanium getter which operates at 720°C . Gas chromatographic analysis of this system during irradiation tests indicates impurity levels less than 2 ppm of O_2 , N_2 , H, or CO. Operation of the purification furnace is controlled from the instrumentation panel by a Variac Control with power meters and a temperature indicator-controller.

The gas control system for each particular cell and for the secondary containment is composed of two separate gas lines. Each of these gas lines, helium and argon, has a proportioning valve and flowmeter. The proportioning valves are interconnected and motor driven to the open or closed position. These motor driven valves proportion the gases from 100% He to 100% Ar, and they are energized by momentary switches. The motor is reduced to such a speed that gas changes in the capsule are made relatively slowly, thus eliminating instant surges of argon gas that could cause an overtemperature condition of the experiment. The proportion of gases entering each cell or containment is displayed on a position meter.

The temperature monitoring system includes the thermocouples in the capsule--tantalum-sheathed BeO-insulated W-5% Re-W-26% Re and

Fig. C-2--Instrumentation for five cell dual gas control capsule - front face

Fig. C-3--Instrumentation for five cell dual gas control capsule - back face

Inconel-sheathed MgO-insulated Chromel-Alumel--and the recording instruments. Each of the cells has three thermocouples of which one is coupled to a single point recorder. The other two fuel measuring thermocouples and the backup thermocouples are recorded on two separate multi-point recorders, one for the tungsten-rhenium thermocouples and the other for the Chromel-Alumel thermocouples. Cold junction compensation is made at these recorders with compensated lead wires directly from the thermocouples to the recorders.

The main power and alarm system is the heart of the instrumentation for providing the safety circuits and control. Included in this system are provisions for detecting low He and Ar inlet gas pressure, fuel or capsule overtemperature conditions, emergency cooling to place the capsule in a safe operating condition, and the associated horn and light display which notify the operators of trouble in the capsule. A red blinking light on the top of the console notifies the operators of trouble, and panel lights indicate the particular area of trouble. A white blinking light indicates the trouble was acknowledged; it does not go out until the problem has been corrected.

Installation of the instrumentation at the reactor site is simplified by designing the equipment so that all inlet and exit connections are made at the upper back plate of the consoles. This feature tends to ensure that proper connections are made without disturbing internal wiring or gas lines.

In conclusion, we have attempted to report on our approaches in the design of a high-temperature versatile irradiation test capsule with certain objectives, the operation of such an irradiation capsule in the ETR, and improvements or developments in specific problem areas. Hopefully, such experiences, which have resulted in a successful irradiation program at General Atomic, will assist others conducting irradiation experiments.

ACKNOWLEDGMENTS

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A CAPSULE DESIGN FOR THE IRRADIATION OF
EIGHT FUEL SPECIMENS AT EACH OF THREE TEMPERATURES*

Neil E. Miller

A part of the Coated Fuel Particle Program at Battelle-Columbus required the evaluation of the fission-gas-retention properties of coated fuels as a function of temperature to high burnup. The sweep-capsule experiment (described in the last meeting) was an excellent tool for this purpose for highly enriched fuels where high burnup could be realized in reasonable time periods at the BRR. For lower enrichment fuels obtaining these burnup levels in reasonable time periods required irradiations in the high flux regions of the ETR where the sweep capsule experiment was not practical. For this work a capsule was needed in which multiple specimens of a fuel material could be irradiated to essentially uniform burnup with each specimen at a different temperature followed by postirradiation evaluation of the coatings. For reasons of economy, it was desired to include several different types of fuel material in each capsule.

The capsule designed for this study is illustrated in the drawing, Figure 1. The capsule would accommodate eight fuel materials with one specimen of each type of fuel in three different temperature zones. For reasons of compatibility with the fuel coatings, strength, and nuclear properties, graphite was used as a fuel container and high-temperature structural material. The fuel sample holders were in the form of longitudinal sectors of right annular cylinders. Eight 45° sectors (or six 60° sectors) were assembled to form a cylinder and

*This work performed under AEC Contract W-7405-eng-92.

supported in a thin-walled graphite cup, as shown in Figure 2. Each sample holder had a series of longitudinal blind holes to hold the coated fuel particles. The holes were sized according to fuel particle diameter to obtain a stacking of 3 to 4 particles per layer. By encouraging each particle to be in contact with the graphite holder, more uniform particle temperatures could be achieved. After loading the specimen holder, the holes were sealed with a graphite cement which was baked subsequently in a vacuum at 1200 C.

The three specimen assemblies were positioned on the centerline graphite support tube inside the double-walled capsule. This assembly is shown in the photograph of capsule components, Figure 3. The centerline tube was supported at each end of the inner capsule and received radial support by 2-mil molybdenum-foil spacers between each specimen assembly.

Two centerline temperatures, near the expected maximum and minimum regions, were measured in each specimen assembly by thermocouples extending inside the support tube. Tantalum-sheathed, MgO-insulated W-5Re versus W-26Re thermocouples were used for the middle and lower specimens. Inconel-sheathed, MgO-insulated Chromel-Alumel thermocouples were used in the top specimen. The thermocouple sheaths were 8 feet long. At the point of connection to glass-insulated lead wire, the sheaths were sealed with an epoxy resin to prevent fission gas from leaking through a failed sheath into the capsule and lead tube. Backup thermocouples were used to measure inner shell temperatures directly radial from each centerline thermocouple. These were 0.040-inch-diameter thermocouples held in grooves on the outer surface of this shell.

The specimen assemblies, from top to bottom, operated at nominally 550, 1150, and 1300 C peak temperatures. These temperature zones were obtained by varying the heat transfer path from, and the heat generation in, each specimen assembly. The fuel loading in the lower specimen assembly was 10 per cent greater than the other two. Each specimen assembly had a different outside diameter to create increasingly larger gas annuli around each specimen from top to bottom. The inner shell had a step in the outside diameter to give a 0.007-in. gas gap adjacent to the top specimen assembly and a 0.010-in. gap along the lower two.

The temperature of the lower specimen assembly was controlled by adjusting the gas conductivity in the outer annulus. The conductivity could be changed by varying the mixture of nitrogen and helium flowing through the annulus. Obviously, this method required accepting a change in two assemblies to establish the temperature in the third.

The remainder of the capsule structure assembly followed a routine procedure. For double-walled capsules incorporating small gas annuli, it has been a practice to fabricate both capsule shells by a rather tedious process to achieve the desired tolerances. Each shell is made from 0.5-in. wall thickness tubing in the following steps: stress relieving, gun boring, rough I.D. honing, rough O.D. grinding, final I.D. honing, and final O.D. grinding. All capsule closure seals were made by Heliarc welding or nickel-base alloy brazing.

Operational Experience

Five capsules of this design were irradiated in the beryllium reflector region of the ETR. The first capsule was irradiated for one cycle to test the capsule design and the postirradiation evaluation techniques. The other four were planned for 3-cycle irradiations. Generally, the operation of the experiments was very good and resulted in the desired evaluation data.

Two problems were encountered. In the first two experiments, a degradation of the centerline W-Re thermocouples caused an indicated downward drift in temperature. The fact that this was not a real temperature change is illustrated by the plot in Figure 4. Thermocouples 2 and 3 show the indicated temperature drift through the three-cycle irradiation. Thermocouples 2A and 3A, the backup thermocouples, show that this gradual reduction did not occur in the adjacent capsule shell. In addition, centerline Thermocouple 1 showed that the drift did not occur in the low temperature specimen. The net result was an uncertainty in the temperature history of the two specimens. This problem was not evident in the latter experiments as shown by the temperature history for the 3-cycle irradiation, Figure 5.

After the first cycle of operation of the third capsule, a gas sample was taken from the lead tube by purging with helium through a small tube extending to the bottom of the lead tube. The result was a sudden high radiation level in the lead tube and the out-of-tank terminal box. The source of the radiation was identified as Ta¹⁸². It is believed that some reaction (probably oxidation) occurred with

the tantalum thermocouple sheaths exposed to the lead-tube atmosphere resulting in the production of a loose reaction product. This product was carried from the bottom of the lead tube by the purge operation. The capsule was discharged at that time. In the following capsules, each tantalum-sheathed thermocouple was encased in a close-fitting stainless steel tube from the capsule header to the seal at the top of the sheath.

Coating Integrity Evaluation

Early attempts to evaluate fission gas release in static capsule irradiations by containing the released gases within metal fuel clads for postirradiation measurements met with less than desired success. The primary problem appeared to be a degradation of the fuel coating materials by reactions with the metal container. In this series of experiments, no attempt was made to collect released gases. After the irradiation, the capsule was stored for three months to permit decay of essentially all fission gases except krypton-85. The specimen assemblies were then recovered and inserted in a second capsule designed for operation near 100 C. The encapsulation of these assemblies, of course, required remote assembly and welding in a hot cell. This capsule was re-irradiated in the BRR for a short period to produce a fresh supply of xenon-133. The specimen assemblies were then recovered and disassembled. The xenon-133 released from failed coatings was collected during the heating of the specimen to 1100 C. From the quantity of xenon-133 released, the fraction of coating failure was determined. This technique proved very effective and is believed to be more reliable than visual or metallographic examinations for detection of cracked coatings.

4.3.6
9.3.6

FIGURE 1. IRRADIATION CAPSULE ASSEMBLY DRAWING

10151

FIGURE 2. PHOTOGRAPH OF GRAPHITE SPECIMEN HOLDER ASSEMBLY BEFORE FILLING (Right) AND AFTER FILLING AND SEALING (Left)

10299

FIGURE 3. PHOTOGRAPH OF CAPSULE COMPONENTS SHOWING SPECIMEN ASSEMBLIES ON GRAPHITE SUPPORT TUBE

FIGURE 5. TEMPERATURE HISTORY OF CAPSULE 2

FIGURE 4. TEMPERATURE HISTORY OF CAPSULE 1

IRRADIATION CAPSULE DESIGN FOR OPERATION TO 2000 C*

William J. Zielenbach

INTRODUCTION

A phase of Battelle's AEC-DRD high-temperature reactor fuels research program has been concerned with developing an irradiation capsule concept capable of 1600 to 2000 C specimen cladding temperature emphasizing such factors as heat transfer, thermocouple location, structural integrity, etc. In the capsule-design study, selected concepts were narrowed down by analytical and experimental procedures to obtain a system with maximum utility and minimum compromise with respect to the following criteria:

- (1) Specimen exposures at desired temperature levels with fuel burnup rates up to 5×10^{19} fissions/cc per normal MTR cycle.
- (2) Capability for significant temperature measurement, i.e., measurement of specimen temperature or of temperatures that could be confidently extrapolated to them.
- (3) Provision for a useful range of specimen temperature control.
- (4) The inclusion of a significant number of specimens in a relatively small experiment.

The basic capsule design was arrived at by evaluating the limitations imposed by the specimen operating temperatures. Liquid metals suitable for cladding immersion are not available owing to unproven compatibility with other materials in the environment or unsuitable Temperature-vapor pressure characteristics. In the absence of liquid

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metals, specimens must be surrounded by an inert gas (or vacuum) through which the heat is transferred by conduction and/(or) radiation to the surrounding environmental enclosure. Consequently, large temperature gradients between the specimens and enclosure are necessary to achieve practical rates of fission heat generation. Meaningful temperature information could not be obtained from a finite-size thermocouple located in a large temperature gradient. Extrapolation of enclosure temperature to specimen temperature was ruled out owing to the great number of associated uncertainties.

SPECIMEN TEMPERATURE MONITORING

Satisfactory temperature monitoring in the capsule system emerged as the leading design problem and the intent of the program was to develop the most acceptable compromise situation. Figure 1 illustrates some of the conceptual cladding designs that were considered. Pertinent comments from the viewpoint of temperature monitoring for each configuration are given in the following paragraphs.

Figure 1-A depicts the reference specimen type which was immersed in a liquid metal in previous lower temperature capsules. However, the problem of locating thermocouples in a liquid metal or in a containment wall rendered this configuration unsuitable as a basic design concept for high-temperature operation. A compromise scheme is depicted in Figure 1-B in which the thermocouple is located in an axial in a thick cladding.

Obviously, the cladding and thermocouple temperatures would closely approach each other. The actual degree of restraint to fuel swelling that might accompany this unrealistically thick, although at 1800 to 2000 C inherently weak, cladding could not be assessed. The required data on high-temperature strength of refractory cladding materials are generally sparse and unreliable.

The second concept was modified to create the configuration shown in Figure 1-C. Ideally, the monitored temperatures and the thin cladding temperature could, by proper adjustments of heat transfer length between the two cladding diameters, be maintained at identical levels. The simple thermal radiation baffle around the thick section was needed to equalize temperatures; the baffle offset the radiative heat losses from the large surface area of the thick section. This concept as shown was discarded, however, owing to uncertainties created by the presence of additional radiative surfaces and the difficulties associated with installing a baffle in an already crowded system. A variation of this scheme was adopted for the final capsule design and is discussed in a later section.

Three other concepts were also considered as depicted in Figures 1-D, -E, and -F. The centerline thermocouple of Figure 1-D afforded the advantage of measuring fuel temperature. However, the thermocouple housing material and the thermocouple itself would operate at temperatures above the clad temperature and material compatibility problems could arise. Figure 1-E depicts two end-to-end specimens with a bare-wire thermocouple bead inserted between the cladding end caps. Measured temperatures would

not be far removed from maximum cladding temperatures since axial heat losses to the thermocouple position would be reduced considerably. Difficulties were anticipated in maintaining adequate dielectric insulation in the exposure zone and in sealing the containment shell against fission-gas seepage into the capsule lead tube. The Figure 1-F configuration involved placing a baffle around the void volume region of the cladding to maintain a more uniform temperature profile in the cladding. The thermocouple was to be located between the cladding and baffle or in the baffle; in either case the maximum clad temperature to thermocouple location gradient was too large.

SPECIMEN TEMPERATURE CONTROL

The companion problem to specimen temperature monitoring was provision for a useful range of specimen temperature control during in-pile operation. The general approach followed along the lines utilized extensively in other irradiation experiments at lower temperatures. The temperature of the specimen environment containment was regulated by the composition of a helium-nitrogen mixture flowing in the outer capsule annulus. The mixed gas variation provided an enclosure temperature variation of 300 C with a corresponding 200 C regulation of the specimen temperature.

In the case of 1800 C specimens, however, the specimen-to-enclosure heat transfer is largely radiative. The temperature control problem has two aspects:

- (1) The net effect from errors associated with estimating flux levels, emittances, etc., are relatively small when compared to the fourth power dependence of temperature on heat flux. Specimen temperatures would also be relatively insensitive to neutron-flux variations and changes in emissivities during the course of irradiation.
- (2) If specimen temperatures are significantly lower than predicted, the same problems interfere with increasing temperature to desired levels.

A reasonable trade-off between these two factors was necessary to assure adequate temperature control capabilities. The specimen enclosure would operate between approximately 550 and 1050 C, which in turn provides an upward response of 150 C in the cladding temperature.

HEAT TRANSFER STUDIES

Most of the cladding configurations mentioned previously were evaluated using a two dimensional, finite-difference digital computer program developed at Battelle to calculate temperature profiles in a combined axial-radial heat-flow situation. The program scope included heat transfer by conduction and radiation; both r-z and x-y flexibility was available for symmetric configurations. Physical inhomogeneities were handled by dividing the system being analyzed into as many as 100 zones of different thermal conductivity, heat-generation rate, emittance, and boundary conditions. A rectangular mesh of intersecting lines was imposed over a schematic model of the system and the interval between mesh lines was varied as necessary to divide the model into the desired zones.

Values for each constant and initial estimates of temperatures in each zone were assigned by including each mesh intersection in one of the up to 100 distinct zones. The computer calculated iteratively the approximate heat-flux imbalance at each mesh intersection using an over-relaxation factor to accelerate the temperature convergence.

Each configuration evaluated in the computer program assumed heat was transferred both axially and radially, by conduction and radiation, to the adjacent specimens (above and below) and to the enclosure (inner shell) wall. The heat generation rates for the fuel and cladding were based on an approximation of the effective neutron fluxes and gamma rates expected in a BRR irradiation. The temperature of the inner shell was set at 550 C and the results obtained for each configuration evaluated predicted peak cladding temperatures of 1700 C or higher could be expected. These solutions were obtained from idealized models. The program could not handle the asymmetrical effects of heat loss through thermocouples and support structure, or the partial baffle effect of several thermocouples passing through a specimen zone to specimens located nearer the bottom of the capsule. Examination of these effects, however, indicated they were negligible.

SPECIMEN THERMOCOUPLE LOCATION

The final cladding configuration shown in Figure 2 evolved from these studies. The sheathed thermocouple was located in an axial hole in a 0.120-inch thick section of cladding located around the lower 1/4 inch

of each 1-inch long fuel specimen. The remainder of the 0.210-inch diameter fuel specimen was restrained in a 0.02-inch thick cladding to allow fuel swelling. The void volume above the fuel was 5/16 inch long and the total clad length with end caps was 1-7/16 inches.

Since a relatively high neutron cross-section material had been selected as the prime cladding candidate, it was necessary to examine closely the changes in specimen temperature profile caused by the variation in cladding diameter and subsequent depression of the heat generation in the fuel adjacent to the thick cladding section. Nuclear mockup studies were performed for this purpose.

NUCLEAR MOCKUP STUDIES

The absolute specimen flux levels and clad perturbation effects were obtained from a BRR nuclear mockup experiment. A fuel core containing uranium of the desired enrichment was inserted in a 0.02-inch thick tube surrounded by a 0.100-inch thick collar of the cladding material. All fuel and cladding dimensions were accurately simulated. Uranium metal foils were located in circumferential grooves on the surface of the fuel pellet and in a centerline hole. The dimensions of the grooves and hole were determined to match the number of uranium-235 atoms removed from the pellet to the number in the fully enriched uranium dosimeter foils. This specimen was placed in a double-wall capsule to simulate the actual capsule configuration.

Analysis of the uranium foils located in the fueled core of the mockup indicated the following:

- (1) The specimen effective thermal neutron flux in the thin clad section was approximately 0.8×10^{13} nv in a reactor test cell in which the unperturbed flux is 4×10^{13} nv. Specimen burnup rates of approximately 2.2×10^{19} fissions per cc of fuel were expected in a 18-day BRR cycle.
- (2) There was a 40 per cent depression of this flux in the fuel located inside the thick collar.

DETAILS OF CAPSULE DESIGN

These results were used in the computer code to predict the expected BRR irradiation specimen temperatures assuming the specimen fission and gamma heat was transferred to the 0.08-inch thick inner containment and from this wall across a 0.0425-inch (cold-temperature dimension) helium gap to a Type 304 stainless steel outer shell in contact with the reactor process water. If the fission heat generation rate had been uniform, a maximum cladding temperature of 1740 C would be expected with a 160 C gradient to the thermocouple location. However, with the reduced local heat generation in the thermocouple region the clad temperature was reduced to 1700 C and the gradient increased to 180 C. The computation also indicated specimen temperatures could be increased to 1850 C when only nitrogen was flowing in the outer annulus; the gradient to the thermocouple position remained unchanged. This specimen temperature change was induced by an increase in temperature of the inner enclosure wall from 550 to 1050 C.

Examination of the effect of changes in cladding emissivity was interesting. If the cladding emissivity changed from 0.25, the value employed in the analysis, to 0.4, absolute specimen temperature levels would decrease approximately 100 C and the gradient to the thermocouple location would not be influenced appreciably.

Two BRR capsule irradiations will be initiated in the next two months. The capsule design as shown in Figure 2 is intended to accommodate six fueled specimens stacked axially in an array 8-5/8 inches long. This will assure the exposure of all specimens to flux no less than 85 per cent of the maximum. However, the first BRR capsule is being designed as a thermal-nuclear mockup with only three specimens. A sheathed thermocouple is located in the cladding of each specimen; three additional bare-wire thermocouples are attached to the exterior of the center specimen to obtain temperature profile data for verification of the computer program results. The bare-wire W-5Re/W-26Re couples are sleeved with BeO protection tubing in the hot zone and with 6 feet of alumina tubing from the header into the experiment lead tube. The entire length of ceramic above the header is encased in stainless steel tubing. The stainless tubing is brazed into two inner chamber seal disks located in the interior of the header and the fission gas seal is made by sealing the bare wires, ceramic, and tubing together with epoxy resin. Solid strand thermoelement wires are used in the protection tubing; however, the thermocouple beads are made with flexible stranded wire (ten 0.003-inch diameter wires twisted together). The bead itself is formed by tying the two stranded thermoelements in a knot and

the knot is held in place by a wire wrapped around the specimen. The flexible elements are also tied to loops fashioned in the solid wire where it emerges from the protection tubing in the exposure zone.

The specimens are secured and centered in a support structure of thin triangular plates and rods. The plates make point contacts with the inner shell to reduce conductive heat losses. Thermal radiation baffles are installed at each end of the specimen array to reduce axial heat losses. The structure is suspended from the capsule header on three 1/16-inch rods to maintain the relative positions of the specimens and thermocouples. This assembly is shown in Figure 3.

Sheathed thermocouples are located in axial grooves on the exterior of the inner shell. The thermocouple beads are positioned opposite the axial center of each specimen and serve as backup temperature monitors in the event of failure or erratic operation of the primary specimen thermocouples. The gas mixture enters the capsule at the top of the outer annulus. A small tube is installed in a groove on this shell to exhaust the helium-nitrogen mixture from the annulus. The inner shell is centered in the outer shell by a triangular disk attached to its end cap.

It is expected that the first experiment will be operated for up to three BRR cycles if performance is favorable. The second capsule containing six specimens is scheduled for irradiation to a higher burnup. The results of these two experiments will be used as the basis for capsule designs for several MTR irradiation in the the next fiscal year.

FIGURE 1. CLADDING CONCEPTS CONSIDERED FOR THERMO-
COUPLE LOCATION IN 2000°C IRRADIATION CAPSULES

FIGURE 2. FINAL SPECIMEN AND CAPSULE DESIGN

Figure 3 Specimen Assembly

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IVb. CAPSULE OPERATIONS - FISSILE MATERIALS (REPOSITIONING POWER CONTROL)

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DESIGN OF AN IN-PILE CAPSULE POSITIONING MECHANISM*

by

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ABSTRACT

A device capable of automatically and remotely adjusting the position of irradiation capsules in the NASA Plum Brook Reactor Facility was designed, fabricated, and tested by General Atomic. The latitude of movement covers a thermal neutron flux range of about a factor of 10.

This device, a Capsule Positioning Mechanism, (CPM), is mated to a 9" O.D. cylindrical vertical test tube facility, and consists of a hydraulic cylinder with its shaft directly coupled to a movable carriage supported by bearings on steel tracks.

Reactor primary water serves as the hydraulic fluid for the system and motion is initiated by signal from a temperature recorder monitoring selected capsule thermocouples. Fail safe automatic and manual control features provide fuel temperature control through CPM movement with or without personnel attendance.

I. INTRODUCTION

Under NASA sponsorship, a program of testing thermionic fuel materials for compatibility, dimensional stability, fission product releases and diffusion at specific temperatures and burnup was initiated. Two capsules were fabricated and irradiated in a GETR RAFT position under this contract. Subsequently it was decided that the next fuel irradiation should be conducted in the Plum Brook Reactor, a NASA Facility at Sandusky, Ohio. Temperature control during operation was one of the desired parameters. A device that could adjust the position of the capsules and hence, essentially maintain a constant flux during the cycle was required. Facilities for flux control of irradiation experiments were not available at Plum Brook Reactor Facility and as a consequence the device was provided by GA.

II. GENERAL DESCRIPTION OF THE REACTOR

The NASA Plum Brook Reactor is a light-water cooled and moderated reactor designed for operation up to 60 thermal megawatts. The core consists of a 3 x 9 array of MTR type, curved fuel elements, about 2 ft. high, and is surrounded by primary beryllium and secondary water reflectors.

The reactor is located in a vertical, cylindrical, stainless steel clad pressure vessel, 9 ft. in diameter and 32 ft. high. A cutaway perspective view of the reactor tank assembly, Figure 1, shows the core location and also the test facilities available. There are two 9" diameter horizontal through holes, three 6" diameter horizontal beam holes, a 42" diameter horizontal thermal column, two 9" diameter vertical test holes, and miscellaneous small test holes and rabbit tubes. We requested and were assigned one of the 9" vertical test holes in which to locate our irradiation capsules.

Figure 1. - Cutaway perspective drawing of reactor tank assembly.

III. DESIGN BACKGROUND

A number of nuclear calculations were performed, and with data made available by NASA, flux profiles of the test facility were obtained. This information together with detailed thermal calculations was employed to determine capsule temperatures for various fuel loadings. To check the calculated neutron flux magnitude and distribution, a gold foil irradiation was conducted in the Plum Brook Mock-up Reactor. The irradiation data confirmed that the flux was sufficient in the V-tube to achieve the required fuel temperatures and power densities at reasonable specimen enrichments. The flux was fairly uniform parallel to the face of the mock-up reactor core and as expected, fell off quite sharply perpendicularly away from the core. In the axial direction a region of 4" - 12" from the core bottom exhibited the most uniform flux profile. In order to irradiate the maximum number of samples at one time, it was decided to position in this region two capsules side by side, parallel to the reactor face. Therefore, a system or device was required that would utilize radial displacement of the capsules as indicated in Figure 2.

The preliminary design of the Capsule Positioning Mechanism (CPM) included exploratory effort on three significant concepts: 1) indirect cable drive, Figure 3; 2) direct cylinder drive and one piece V-tube, Figure 4; and 3) direct cylinder drive and a three piece V-tube, Figure 5. The first approach was undertaken because Plum Brook had required a separate coolant system for irradiation in a V-tube position. At our request, Plum Brook relaxed this "two coolant system" requirement and this resulted in a simplified direct drive hydraulic cylinder using reactor primary water as the hydraulic fluid. This concept required that a work platform be placed in the main reactor hatch and a man lowered about 7 feet into the reactor tank, as shown in Figure 6. At that level he could complete the assembly of the hydraulic cylinder to the V-tube after it was lowered through the 9" reactor tank nozzle. Because of the high radiation levels observed in the reactor vessel, the Plum Brook staff requested the procedure be changed to avoid exposure of personnel. The concept was then modified to its present design.

Fig. 2--Dual capsule arrangement

Fig. 3--Indirect cable drive concept

Fig. 4--Direct cylinder drive, one piece V-tube

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Fig. 5--Direct cylinder drive, three piece V-tube

Fig. 6--Preliminary concept for assembling
CPM to V-tube

IV. DESIGN DETAILS

The V-tube for this experiment is used only as a positioning and holding fixture for the Capsule Positioning Mechanism. There are three distinct parts to this assembly: 1) the pressure vessel penetration, 2) the center extension tube, and 3) the CPM holding fixture.

The upper portion of the tube, when installed, becomes part of the reactor pressure vessel and, as such, is designed and certified as a pressure vessel. The valves associated with the cylinder hydraulic system are externally mounted on this part of the V-tube. The central "extension tube" is provided with a slot through which the capsules may be discharged at the end of the test and removed from the reactor via a fuel discharge chute. The central and lower parts of the tube are fastened together with flanges and aluminum bolts. The purpose of this connection is to provide shorter sections to simplify the initial installation and ultimately the discharge and disposal.

The CPM consists of a carriage mounted on two stainless steel rails and driven by a hydraulic cylinder. The rails and cylinder are attached to a frame which in turn is mounted into the lower section of the V-tube. The rails are positioned in the frame using corrugated stainless steel tolerance rings designed to reduce small misalignments.

The carriage assembly consists of an aluminum block in which are mounted two stainless steel precision Thompson linear ball bushings which provide low friction movement of the carriage on its rails, and two 1-1/4" I.D. aluminum tubes that serve the dual purpose of holding the capsules and providing a coolant annulus. Total movement is 4-1/4" which covers a thermal neutron flux range of about a factor of 10.

The hydraulic cylinder is fabricated of 17-4 PH steel condition H-1100. It is built without seals at the shaft penetrations, and the clearance at these points is such that a good sliding fit is obtained with minimum leakage. Piston sealing and concentricity requirements are met by using a pair of expanding metal seal rings.

A schematic of the hydraulic arrangement is shown in Figure 7. In operation, primary system pressure is applied to both sides of the piston during a hold condition so that leakage around the piston cannot cause unwanted movement of the piston. When movement is desired, one side of the cylinder is opened to a hot drain at approximately atmospheric pressure, through a throttling valve to provide the desired rate of travel. The throttling valve can be bypassed to provide a higher speed of travel if desired.

Motion of the CPM is initiated by signals from a temperature controller monitoring selected capsule thermocouples. Solenoid valves in the hydraulic cylinder lines are controlled by switches in the recorder. Fail safe automatic and manual control features provide fuel temperature control through CPM movements with or without personnel attendance.

V. INSTALLATION PROCEDURE

The installation of the experiment is shown in Figures 8 through 12. The lower section with the hydraulic cylinder attached is lowered into the reactor tank through the 36" hatch by the overhead crane using a special "C" shaped lifting fixture. The tube is maneuvered in the reactor tank and is lifted up through the reactor tank nozzle a short distance. A temporary holding clamp is employed and the lifting fixture removed. The V-tube penetration assembly is lowered into position, hydraulic lines mated, and a coupling clamp installed. The temporary clamp is removed and the V-tube is lowered into a guide bracket at the core bottom until the nozzle flanges meet. After bolting down, and connecting the various hydraulic and electrical lines, the capsules are installed.

VI. TEST PROGRAM

The entire installation procedure was simulated at GA using the Engineering Test Tower and associated facilities. No difficulties were encountered although minor changes to the installation procedures were required. A water tank at atmospheric pressure was used for simulation of the reactor environment and a reservoir of pressurized water was provided to actuate the hydraulic cylinder. The CPM was operated through fifty cycles and

Fig. 7--Simplified hydraulic schematic

Fig. 8--Lower section being installed through hatch

Fig. 9--Lower section in place under reactor tank nozzle

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Fig. 10--Lower section temporarily supported on nozzle

Fig. 11--Penetration assembly being mated to lower sections

Fig. 12--Complete assembly in position

the linear differential transformer calibration was checked. Travel speeds as a function of applied differential pressure were obtained. The motion and its control were very satisfactory except at extremely slow travel speeds. Following the preliminary tests the system was disassembled for inspection from which no visible indications of wear or corrosion were evident.

The installation, test, and disassembly were repeated later to the satisfaction of NASA observers. Following disassembly the individual components were packaged for shipment.

VII. OPERATIONAL HISTORY

The experiment was installed in the Plum Brook Reactor during a scheduled shutdown in December 1965. No significant problems were encountered during the installation.

In operation it is necessary to provide for limiting the motion of the CPM toward the core in case of an accident, e.g., a line rupture, while at the same time permitting reasonable travel speeds for normal circumstances. A special two-way check valve was installed for this purpose. An erratic starting behavior was observed, and attributed to the combined effect of the throttled flow and in-leakage around the piston shaft. During the first irradiation cycle, the CPM at several times could not be moved in the "In" direction. This condition became more pronounced during the second cycle, and finally prior to the third cycle the CPM stuck and could not be moved in either direction.

After discharging the capsules intact and stored in the canal, the V-tube was removed from the reactor and the cylinder was inspected. The piston shaft had frozen onto one of the cylinder end plates and showed visible signs of galling. An appreciable amount of fine metal particles was also present in the cylinder. An inspection of the carriage showed that the tracks were slightly scored. Although the ball bushings still looked good, extreme difficulty was encountered in initiating motion by comparison with the pre-installation condition.

These various mechanical problems are currently being studied by both GA and NASA. In particular, a mechanical stop has been designed that would physically limit the travel of the CPM toward the core, thus allowing faster travel speeds for normal operation. The stop is positioned on the rails between the carriage and frame inside the V-tube wall, is inserted from the reactor hatch and can be changed if different travel distances are required.

33 68000

In-Core Vertical Adjustable Facility for Capsule
Irradiations in the NASA Plum Brook Reactor

by

T. L. Gregory

General Electric Company
Nuclear Technology Department
Pleasanton, California

Program Sponsorship

NASA Lewis Research Center
Nuclear Power Technology Branch

In-Core Vertical Adjustable Facility for Capsule
Irradiations in the NASA Plum Brook Reactor

by

T. L. Gregory

The power generation and temperatures, in an irradiation capsule containing nuclear fuel, can be controlled in a number of ways. One of the best is by moving the capsule with respect to the core, to change the flux, thus, increase or decrease the total thermal power generation.

A mechanical device has been designed for use in an in-core position in the NASA Plum Brook Reactor to allow a maximum of ten inches of vertical movement with respect to the core. Such a device has been built and is in use. It has proved an economical electro-mechanical device for remotely positioning a capsule in a core position.

This device was designed as part of a high-temperature fuel form irradiation under the sponsorship of NASA Lewis Research Center, Nuclear Power Technology Branch (Contract No. NAS 3-2544). Only the in-core repositioning facility will be described in this paper.

The most significant aspect of this device is that it permits in-core vertical capsule positioning through a linear motion pressure vessel penetration. Position is measured with respect to the core by using a position-indicating transducer mounted outside the pressure vessel on the drive unit. Figure 1 shows the equipment installed in the reactor vessel.

The device consists of four main assemblies: in-core facility tube, push-pull cable, pressure vessel seal, and drive unit. The movement can be adjusted before installation to cover any 10-inch travel in the 24-inch vertical fuel region. The facility tube is designed with a capsule mounting plate on an actuating rod. Figure 2 shows a cutaway drawing of the facility tube with a typical capsule assembly in place. The facility tube fits in a standard in-core PBR Beryllium "L" piece spaced to provide a 0.075-inch-cooling annulus. The downward flow of the coolant facilitates holding the tube in place, however, a suitable hold-down bracket is also provided. Figure 4 shows a plan view of the core and a typical installation. The vertical movement is achieved through the actuating rod mounted on linear ball bushings. The rod is attached to a ball-bearing, flex, push-pull cable.

The in-core facility tube and the ball-bearing, push-pull control cable before reactor installation is shown in Figure 5. The facility tube projects approximately 42 inches above the top core grid plate.

As shown in Figure 1, the push-pull cable is routed up the inside wall of the reactor pressure vessel and secured by instrumentation clamps. The external drive unit (Figure 3) consists of a linear-motion pressure vessel seal attached to a flange designed to fit a 4-inch reactor vessel penetration. The seal consists of a shaft through a double V-ring Teflon packing. The linear drive is powered by a reversible fractional horsepower (a-c phase shift) motor driving through a worm gear-screw-type mechanical actuator. The motor can be run in either direction by using a circuit with a 3-position switch, for moving the capsule up or down from a remote console. The switch is a spring-loaded-type with

center position off for operation safety. A linear-position transducer is mounted on an adjustable pad on the drive and is coupled to a key on the seal shaft (Figure 3). The transducer can be adjusted and calibrated to send a signal to a remote voltmeter calibrated to read in inches. The position of the capsule with respect to the reactor core can be measured to better than ± 0.100 inch.

Figure 6 shows the drive unit mounted on an out-of-pile test stand. The whole unit projects approximately 55 inches above the penetration flange.

The seal for the linear-motion pressure vessel penetration is a double V-ring-type with a bleed between the packing layers. After the seal has been operated for several cycles under pressure, the leakage rate through the bleed port is less than 50 milliliters per day. The leakage is drained to a hot sump. The seal was designed for 150 psi and up to 150°F. However, the seal was tested to over 1.5 times the design pressure. Another bleed is provided below the seal to allow venting of trapped air when the reactor vessel is pressurized. Figure 7 shows the parts of the disassembled drive unit. The parts of the seal can be seen in the center of the photo.

The device can be controlled from a remote console. In the PBR installation, the console is located outside the containment vessel.

This device has been used successfully in the Plum Brook Reactor for seven cycles to date and has been reliable; no maintenance problems have occurred.

Figure 5 In-Core Facility Tube and Flex Cable

Figure 6 Drive Unit and Pressure Vessel Seal

Figure 7 Disassembled Drive Unit

A DESIGN OF A FLUX POSITIONING CAPSULE FOR USE IN THE
VERTICAL THIMBLES OF THE CP-5 REACTOR*

by

W. N. Beck
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SUMMARY

A specially designed extension tube assembly permits remote positioning of an instrumented irradiation capsule within the vertical thimbles of the CP-5 reactor. The assembly has a total travel of 15 inches and is manually operated from the top of the reactor. The assembly has been used to control irradiation specimen temperature as well as a means of flux mapping the fuel tubes.

INTRODUCTION

The CP-5 reactor is a heavy water-moderated and cooled research reactor. It is fueled with cylindrical tubular elements of aluminum-uranium alloy jacketed with aluminum. The complete core loading consists of 17 vertical fuel elements, each element consisting of three concentric

*This work was performed under the auspices of the U.S. Atomic Energy Commission.

fuel tubes. (1) Some of these fuel elements are fitted with a central D_2O flow thimble so as to permit irradiations of instrumented capsules within the reactor core.

The fuel elements have an overall length dimension of 30 inches, however, the fueled portion of the tube measures approximately 24 inches.

Occasional measurements of the neutron flux profile in the fuel tubes showed flux peaking at the center with a gradual reduction toward the upper end of the tube (Figure 1). Several irradiations had been performed where the capsule assembly had been designed so that the specimens were located in an anticipated flux position. It was found that the flux values were subject to error or change due to (1) core loading, (2) control rod position, and (3) the addition or removal of absorber in adjacent thimbles.

A modification to the procedure of preselecting a fixed flux position was a technique of repositioning the capsule in the thimble by the use of shims which could be inserted under the shield plug collar. This procedure had its limitations in that the capsule could only be raised a total of four inches before the assembly interfered with the reactor top shield.

In order to overcome these limitations, a capsule assembly was designed which had a telescoping extension tube which permitted remote positioning of a capsule from the midplane of the reactor core to a position three inches above the fueled portion of the reactor tubes.

NEUTRON FLUX MEASUREMENT FOR THIMBLE VT-8

Fig. 1. Neutron Flux Measurement for Thimble VT-8

CAPSULE ASSEMBLY

The assembly constitutes three main sections: (1) shield plug assembly, (2) extension tube, and (3) capsule (see Figures 2 and 3). The shield plug duplicates the overall dimensions of a standard metallurgy plug⁽²⁾ but differs in the addition of a top connector housing and an enlarged spiral central tube which contains a flexible drive shaft and the thermocouple extension wires. The connector housing is necessarily a gas tight enclosure to prevent leakage of the reactor vessel helium gas cover. The electrical connector is a pressure seal type. The flexible shaft drive seals to its housing by an "O" ring. The shield plug casing is filled with paraffin and lead shot. Two boral disks at the bottom end serve as neutron absorbers.

The telescoping extension tube connects to the bottom of the shield plug. The telescoping action is accomplished by rotating a "Thompson" ball screw which moves a ball-nut. The screw is supported at both ends in bearing blocks while the nut is fitted in a block which attaches to the extension tube casing. The screw block rides on three guide tubes. These tubes also house the thermocouple extension leads. The three guide tubes join at the bottom in a drilled block which provides a common entry for the bundle of thermocouple extension lines. A packing gland prevents the thermocouple bundle from exerting tension on the leads which are inside of the extension tubes. Approximately 30 feet of spiralled thermocouple leads are housed in the 28-inch long chamber below the ball screw assembly. The bottom end of this chamber

Fig. 2. CP-5 Flux Positioning Capsule Assembly

Figure 3 Photograph of Assembly Components

is sealed with an aluminum fitting which is the attachment for the irradiation capsule. This fitting also has the packing gland which secures the bundle of thermocouple leads.

The irradiation capsule joins to the last mentioned seal tube with a pressure fitting. The capsule is a double chamber vessel, the top portion housing the thermocouple transition junctions and the lower section which is a sealed NaK-filled vessel which contains the specimens. For flux mapping purposes, a single fuel rod is utilized, as shown in Figure 4. Routine irradiations are performed with six specimens arranged in two tiers of three each.

The thermocouples are arranged in a single pin capsule so that the radial temperature distribution within the NaK bath may be accurately determined. With this information it is possible to calculate the heat generation of the specimen as well as the effective neutron flux.

PERFORMANCE CHARACTERISTICS

This capsule assembly was used to make repeated scans of the reactor fuel tubes under various operation conditions and core loadings. Positioning of the capsule was not permitted while the reactor was operating, consequently, the scans were made in successive steps. A plot of a UC-PuC fuel specimen jacket temperature versus position in a fuel tube is shown in Figure 5. It was observed that temperature control of a capsule had to be adjusted only when a change had been made in the core loading or if a transfer or replacement of absorbers had been made. The reactor control systems maintained a relatively

Fig. 4. Diagram of Single Pin Irradiation Capsule Vessel

Fig. 5. Effect of Capsule Position on Specimen Jacket Surface Temperature

uniform flux level under steady state operation. This type of capsule assembly is presently being operated in the reactor and one unit has been in operation for a period in excess of one year.

A postirradiation examination of a subassembly which had been in the reactor for eight months failed to show deterioration of any of the components or structural weakness. The whole upper section of this assembly, which included the screw drive mechanism, was refitted with a new casing and reassembled as another irradiation experiment.

ACKNOWLEDGMENT

The author would like to acknowledge the assistance of J. J. Minarik in the design of the assembly, R. J. Fousek in the testing and operation, and personnel of Reactor Operations Division without whose cooperation the experiments could not be performed.

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An Automated Power and Pressure Cycling System for
Elevated Temperature for Irradiation of Thin Clad Fuel Pins

by

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INTRODUCTION

The use of thin clad in water cooled reactors has two economic advantages. First and more obvious is the neutron savings compared to thick clad; second and perhaps more significant is the reduction in core size that is possible. To maintain the same water-to-uranium ratio, the lattice spacing is reduced for reduced clad thickness. For large cores, the required pressure vessel size is appreciably reduced.

In high temperature reactor systems, fast as well as thermal, the collapse of the clad onto the fuel may be avoidable. But whether the clad is pre-collapsed or allowed to collapse during operation, the fuel element designer must consider the mechanical behavior as well as the corrosion, nuclear, and other properties of the clad material.

Most published methods of analyzing the clad-fuel mechanical system are concerned with creep buckling, and strain fatigue. Attempts are made to separate these phenomena and to relate each to the differential thermal expansion of clad and fuel, and to the differential pressure across the clad thickness. Unfortunately, these attempts are not often satisfactory. Several in-pile and ex-pile experiments have been aimed at developing suitable mathematical models of creep, buckling, and strain fatigue. Some of these have been fairly successful, but a great deal of such work remains to be done.

A different approach is the brute force method of directly relating fuel element endurance to the operating variables - power cycling, thermal cycling, pressure cycling. From the standpoint of improving our sophistication as fuel element designers, the brute force method is a dead loss. Its sole recommendation is the speed with which practical data are obtained. The system described in this paper has not been built, but is designed to provide a flexible arrangement for performing endurance tests of fuel pins.

DESCRIPTION OF SYSTEM COMPONENTS

Fuel Pin and Inner Capsule

The fuel pin identified as Item 4 of Figure 1 may be considered typical of a thin clad test element. The fuel pellets are shown dished with inert (e.g. Al_2O_3) guard pellets at the ends of the column. The end plugs are welded in place in a low pressure weld box and the pin is then externally pressurized at elevated temperature to plastically collapse the clad. The nose on the upper end plug fits into a ceramic disk, Item 17 of Figure 1, which centers the pin and guides thermocouples around it. The lower end plug of the pin is centered by the inner capsule end plug. The roll pin, item 3 of Figure 1, joins these end plugs for ease of assembly and disassembly. The sheathed thermocouples, item 21 of Figure 1, are bundled and inserted up through the lead tube at the top of the inner capsule. The spacers, Items 5, 6, and 17 of Figure 1, are attached to the couples and this array inserted to the proper level in the inner capsule. The capsule is inverted and the fuel pin lowered until the bottom plug of the inner capsule is seated. After the welds are completed, the capsule is turned upright and the fluid, Item 16 of Figure 1, is added (through a fillhole, not shown) until the pin is completely submerged. The fluid may be water, liquid metal, steam, air, etc. The lead tube which contains the thermocouples passes up through a junction assembly located in the vicinity of the operating console. The lead tube transmits cyclic or steady gas pressure to the fuel pin and serves as a sampling tube to detect fission gas from the failed pin. This detection system signals end-of-test.

Outer Capsule

The pressure lead tube from the inner capsule penetrates the upper end plug of the outer capsule. This centers the top of the inner capsule within the outer one. At the base, centering is accomplished with a coil spring which also absorbs relative thermal expansion of the two capsules and serves as an absorber for hydraulic or handling shocks.

The tube which coils down the outside of the outer capsule acts as a flow guide

for cooling water moving past the capsule and conducts a N_2 -He mixture to the base of the assembly. This controlled (composition and flow) gas mixture provides a barrier of known thermal conductivity in the annular space between the inner and outer capsules. After it passes through the annulus, it is ducted to the off-gas system of the reactor facility.

The heat generated in the fuel is conducted through the assembly walls to cooling water flowing co-axially upward through a guide tube. The capsule assembly is held in the guide tube at the desired axial position by a clamp which attaches to the top of the tube. The clamp is operated with a remote handling tool.

"Raft" Facility

To provide power cycling, the capsule assembly (with cooling water flow guide) is placed in a facility which moves in and out of the neutron flux. One such arrangement, Figure 2, is the "raft" facility available at the General Electric Test Reactor. The raft depicted was redesigned from drawings supplied by G.E. to provide a greater range of travel and control of the motion by an hydraulic cylinder. A portion of the hydraulic-pneumatic system schematic is shown as an inset near the title block of Figure 2. The travel of the push rod and support block holding the raft tube and capsule assembly is limited by contact of the slide, Item 2, with the pneumatic valves, Item 7 in Figure 2. These pneumatic valves operate a master cylinder at the control console. The complete schematic and console design are given in Figure 3. The function sequence for this system is given in Appendix I. The system is pre-programed to slowly advance the raft toward the core and then quickly withdraw it. This is intended to simulate startup and scram.

Gas Pressurization and Thermal Control Systems

The thermal control system depicted in Figure 4 is patterned after one used at Idaho Falls. Similar and advanced systems are now in general use, so a detailed description is omitted from this paper. The system shown is for the simultaneous irradiation of two capsules. Clad temperatures of order $550^{\circ}C$ and pressures around 2500 psig are typical of the capabilities of the system.

The gas pressurizing system leading to the inner capsule is also for two units. The fission gas detection and capsule temperature monitoring equipment are not illustrated.

SUMMARY

A system has been designed to permit automatic cycling of temperature, pressure, and power, independently in fuel-bearing, instrumented capsules. Temperature and pressure levels are those of interest in fast and superheat reactors, as well as conventional water reactors.

Appendix I

Function Sequence of Raft Control System

Starting with raft in the position shown on Dwg. 540F064, (in the scram position) action can be started by turning on air supply.

Air flows in line E to inlet side of valve V3. Cylinder is in extreme right position and therefore has opened V3 (a limit switch). V2 is still closed, therefore air pressure is transmitted up line A and through V1 (which is open to Cyl. 1 port when handle is up).

V1 pressurizes pilot cylinders on valves PV1 and PV2. When PV1 is pressurized the Out and Ex ports are connected. Air in top portion of pressurizing tank T is then connected to atmosphere.

Simultaneously, inlet air and outlet of valve PV2 are connected, permitting line air pressure to pass through air control valve ACV and to air cylinder connection on intensifier I.

Air forces piston down and water on opposite side of piston flows through Water Flow Control valve WFC to water tank T. The second port of the piston is of a smaller diameter and operates in a small diameter cylinder. Intensified pressure developed in this portion flows through line C to the power cylinder connected to the "raft".

Slight initial travel closes valve V3 and "raft" continues to advance until valve V2 is opened. Travel rate is controlled by the settings on valves ACV and WFC.

With valve V2 opened, the air trapped in line A is released through the valve to exhaust. This also releases pressure on pilot control section of valves PV1 and PV2, causing these valves to move to their de-energized positions.

Valve PV1 now supplies air pressure above the water in tank T, and valve PV2 connects the outlet and exhaust ports.

Water pressure developed in tank T flows through line B through check valve portion of valve WFC and also to the left hand portion of PWR CYL forcing "raft" to travel to the right. Air in intensifier I is forced through check valve portion of valve ACV and exhausted through valve PV2.

"Raft" travel continues until stroke limit is reached. At this time valve V3 is operated and the cycling sequence is started again.

Travel to the left is slow as controlled by valve settings on valves WFC and ACV. Travel to the right is rapid and is limited only by the flow capacities of the check valves in WFC and ACV.

If, at any time in the operations of the raft, it is desired to "scram" the raft, depressing the handle on valve V1 will initiate this action and when raft reaches extreme right position it will remain there until handle on valve V1 is lifted. This is an emergency measure and is used to quickly withdraw raft and terminate cycling.

By substituting a solenoid operated valve for V1, a detrimental reactor function can be made to "scram" the raft and save the experiments being conducted.

Appendix I

MODIFIED "RAFT"
FACILITY AND HYDRAULIC
ACTUATOR

FIGURE 2

ITEM	DESCRIPTION	QTY	UNIT	REMARKS
1	GUIDE PLATE	1	PC	
2	GUIDE PLATE	1	PC	
3	GUIDE PLATE	1	PC	
4	GUIDE PLATE	1	PC	
5	GUIDE PLATE	1	PC	
6	GUIDE PLATE	1	PC	
7	VALVE	1	PC	
8	VALVE	1	PC	
9	VALVE	1	PC	
10	VALVE	1	PC	
11	VALVE	1	PC	
12	VALVE	1	PC	
13	VALVE	1	PC	
14	VALVE	1	PC	
15	VALVE	1	PC	
16	VALVE	1	PC	
17	VALVE	1	PC	
18	VALVE	1	PC	
19	VALVE	1	PC	
20	VALVE	1	PC	
21	VALVE	1	PC	
22	VALVE	1	PC	

A- B.O. FROM GENERAL ELECTRIC CORP.
B- B.O. FROM ROSS OPERATING VALVE CO. DETROIT, MICH.
C- B.O. FROM TONKINS JOHNSON CO. JACKSON, MICH.
D- B.O. FROM WILLIAMS CO. PHA, PA.
(PARKER LEAKPROOF TUBE FITTINGS)
E- B.O. FROM FLEXONICS, CAT NERT 50
150 ID. 1.000 O.D. SST. 1500 PSI NIPPLE ON ONE END
150 NPT BUTT WELD NIPPLE OTHER END.
TOTAL LENGTH 43.1 BT.

BILL OF MATERIAL					NO. REQ.		
QTY	ITEM	TITLE	DRAWING & BL OR FT.	MATERIAL SPEC.	GROUP SPEC.	U	F
1		HAND OPERATED VALVE	540F505			1	
2		AIR OPERATED VALVE	540F505			1	
3		AIR OPERATED VALVE	540F505			1	
4		COLORFLOW VALVE	540F505			2	
5		WATER TANK	540F505			1	
6		INTERMIFIER	540F505			1	
A 7		VALVE #A212-B				2	
A 8		MTG. BRACKET #MF-2				2	
9		PANEL	G74C-200			1	
B 10		CHANNEL RACK				1	
C 11		TEE #5BTX-D				3	
12		ANGLE	540F505			1	
13		1-8 X 1/2 HEX. HD. SST BOLT				4	
14		1-8 HEX. SST. NUT				4	
15		1" SST. LOCKWASHER				4	
C 16		UNION #5WBTX-D				2	
C 17		UNION #6WBTX-D				4	
C 18		UNION #5WBTX-D				4	
19		3/8-16 X 1/2 HEX. HD. SST BOLT				4	
20		3/8 SST. LOCKWASHER				4	
21		5/16-18 X 1/2 HEX. HD. SST BOLT				2	
22		5/16 SST. LOCKWASHER				2	
23		5/16-18 HEX. SST. NUT				2	
24		3/4-10 X 2 HEX. HD. SST. BOLT				4	
25		3/4-10 HEX. SST. NUT				4	
26		3/4 SST. LOCKWASHER				4	
C 27		TEE #6BCTX-D				2	
C 28		REDUCER 1/2 X 1/4 PTR-S				2	
C 29		CONNECTOR #5BCTX-D				2	
C 30		REDUCER #5BCTX-D				5	
C 31		CONNECTOR #6BCTX-D				1	
32		GAGE				1	

A- B.O. FROM REPUBLIC MFG. CO. CLEVELAND, OHIO
B- PAR-METAL TYPE "A" CHANNEL RACK #115-RX-195
B.O. FROM CAMERADIO CO. PGH., PA.
C- B.O. FROM WILLIAMS & CO. PGH., PA.
(PARKER LEAKPROOF TUBE FITTINGS)

540F135

REAR VIEW OF PANEL (PIPING)
(ALL PIPING ON BACK OF PANEL)

REAR VIEW OF PANEL

1
D. GOTTWITZ
IT 1, CHANGE

THERMAL CONTROL SYSTEM

GAS PRESSURE SYSTEM

BILL OF MATERIAL				NO. REQ.	
ITEM	DESCRIPTION	DRAWING # OR OR IT.	MATERIAL SPECIFICATION	QTY	UNIT
A 1	NEEDLE VALVE # R-352			1	
A 2	NEEDLE VALVE # R-354			1	
B 3	MIXING CONTAINER	540F135, GR		1	
B 4	ANALYZER # BECKMAN			1	
B 5	REGULATOR # PARKER			1	
A 6	FLOW METER			1	
7	PRESSURE GAGE (1000 PSI)			2	
8	PRESSURE GAGE (1500 PSI)			2	
C 9	SEL. SW. DP-DT # 143-201100			1	
C 10	SEL. SW. DP-DT # 143-201100			1	
11	SOL. VALVE # C-326211			4	
12	ORIFICE			2	
D 13	UNION TEE # 2 JBTX-D			9	
D 14	MALE ELBOW # 2 JBTX-D			40	
D 15	UNION CROSS # 2 JBTX-D			2	
C 16	SW. BOX # BOXO			1	
C 17	SW. BOX COVER FOR # 10			1	
C 18	PANEL	540F135		1	
C 19	CHANNEL RACK			1	

A. B. O. FR. HOKE INC., NEW JERSEY
B. B. O. FR. BECKMAN, FULLERTON, CALIFORNIA
C. B. O. FR. CAMERADIO, PGM, PA.
D. B. O. FR. WILLIAMS & CO., PGM, PA. (PARKER LEAKPROOF TUBE FITTING)

THERMAL CONTROL AND GAS PRESSURE SYSTEMS CONSOLE

FIGURE 4

UNLESS OTHERWISE SPECIFIED THE FOLLOWING TOLERANCES APPLY	
DIMENSIONS: ALL DIMENSIONS ARE IN INCHES UNLESS OTHERWISE SPECIFIED	
SURFACE FINISH: ALL SURFACES SHALL BE FINISHED TO THE FOLLOWING TOLERANCES	
MATERIALS: ALL MATERIALS SHALL BE OF THE FOLLOWING GRADES	
MANUFACTURER: WESTINGHOUSE ELECTRIC CORPORATION	
DIVISION: ATOMIC POWER DIV.	
LOCATION: PITTSBURGH, PA., U.S.A.	
SCALE: 1/4" = 1"	
SUB 1	

4.8.9

SURVEILLANCE OF IN-PILE TESTS
USING NEUTRON RADIOGRAPHY

by

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The development of materials suitable for use in and around a nuclear reactor core requires extensive in-pile experimental programs. In many instances, these experiments are hampered and results are limited because of the destructive analysis procedures required to determine the condition of the material. In the past few years neutron radiography has shown considerable promise as a means of nondestructively examining radioactive materials.⁽¹⁾

The examination of irradiated fuel capsules and fuel specimens using neutron radiography has been carried out at the Battelle Research Reactor (BRR) since early 1965. The manner in which this technique has been applied utilizes the reactor core as a neutron source and the pool water* as a biological shield, thereby precluding the fabrication of costly additional shielding.

* The BRR is a 2 Mw, pool reactor.

Neutron radiography consists of activating a metallic foil or screen by a well-collimated neutron beam after it has passed through the object under examination. The "image" of the object is formed on the screen in relation to the neutron absorption characteristics of the object. This image is then obtained on photographic film either directly using prompt neutron reactions with the film placed adjacent to the screen or by removing the activated screen and allowing it to decay adjacent to the film. Various suitable screen materials for both methods have been reported.⁽²⁾

The latter transfer method, utilizing 5 mil indium as the screen material, has been used at the BRR. In this manner, the radioactivity of the object does not fog the film and the film need not be placed in the water. The neutron beam is obtained using a 12 foot long, 1-1/2 inch I.D., aluminum tube placed horizontally in the reactor pool with one end within a few inches of the reactor core. The opposite end is fitted with a square, vertical channel to act as a guide for the fuel specimens. (See Figure 1.) The radioactive specimens are placed in a radiography fixture loaded with indium foil. This fixture slides in the vertical guide and is driven at a constant speed past the end of the collimating tube by a gearmotor mounted above the pool surface. This scanning method permits examination of long specimens without increasing the collimator cross section.

The specimen is then unloaded from the fixture remotely, the activated screen removed from the pool, and allowed to decay for 2 to 3 half-lives in a film cassette loaded with Kodak Type AA X-ray film. The specimens are then returned to the shipping cask and removed from the pool. For capsules irradiated on site and examined at intervals throughout a test, they need not be removed from the pool in order to obtain the radiograph.

Neutron radiography is used at the BRR for non-destructive examination of irradiated fuel specimens during the shutdown of the ETR cycle. The specimens are shipped to Battelle, unloaded from the shipping cask underwater, and placed in a radiography fixture. After radiography they are replaced in the cask and shipped back to Idaho in time for insertion for cycle startup. To date, approximately 80 clad fuel specimens have been examined. The majority of the specimens have been rods ranging in diameter from 0.4 to 0.6 inches and from 4 to 50 inches in length. Rods with estimated fission product activities of 60,000 curies have been successfully radiographed. Figure 2 is an example of this type of radiograph which has shown fuel pellet interfaces, cracks, fuel relocations, and swelling. It has been possible to radiograph up to 20 of these specimens on a one to two day turn around basis to facilitate interim examinations throughout the duration of the in-pile test.

Stainless steel irradiation capsules containing nuclear fuel have also been examined. Figure 3 is an example of a 1-1/8 inch O.D., 0.049 inch wall capsule containing a 1/4 inch fuel rod surrounded by NaK and a heat barrier.⁽³⁾ This capsule had been irradiated at steady state conditions, then was radiographed prior to a transient irradiation to better distinguish between effects caused by the steady state or transient phases of the test. The thermocouple connections were enclosed during the radiographic inspection to exclude the pool water so that the transient test could be instrumented.

Figure 4 is a doubly contained, 1 inch O.D. high burnup fuel capsule containing liquid sodium. The wall thickness of each containment is 0.050 inch with a 0.025 inch gap between. Information readily obtainable from radiographs of this type are sodium levels, fuel location, thermocouple location, dimensions, and overall capsule integrity. This information obtained at intervals throughout an in-pile test aids in determining the time at which changes begin to occur.

For capsules irradiated at the BRR, it is planned to obtain neutron radiographs of the capsule over each reactor shutdown. Since transportation is not involved, this can be accomplished with the lead tubes intact. For instrumented capsules which must be transported, the lead tubes must be capable of being disconnected. For non-instrumented capsules, transportation provides no obstacle and short turn around times at the radiographic facility can be utilized to return the capsule for reactor start up. The information obtained in this manner then provides a pictorial history of the in-pile capsule.

The nondestructive neutron radiographic technique has been a valuable tool in the analysis of fuel development programs in another area. The fuel specimens can be radiographed upon completion of the in-pile test and subsequent removal from the encapsulation. This examination will show more detail of the specimen, such as cracks, voids, etc., and may be used as a tool for determining the more advantageous post irradiation sampling locations in order to obtain the maximum amount of information from each in-pile test.

Fabrication of improved radiography apparatus is presently underway at Battelle. These devices will incorporate fixtures permitting the examination of irregular geometries and deformed specimens by utilizing a container from which water will be excluded. The necessity of machining a close fitting fixture for each sample will be eliminated. Plans and devices are also being made to examine irradiated specimens up to 10 feet in length. It is anticipated that these devices will enhance the quality of the neutron radiographs and extend the areas of application.

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Figure 1 Neutron Collimating Tube Showing Verticle Radiography Fixture Guide

Figure 2 Neutron Radiograph of Irradiated Fuel Specimens
Showing Pellet Interfaces, Cracks, and Fuel Annulus

Figure 3 Neutron Radiograph of Stainless Steel Fueled Capsule

Figure 4 Neutron Radiography of Doubly Contained Stainless Steel Irradiated Capsule

IN-PILE PROPERTIES OF REACTOR FUELS BY OSCILLATING TECHNIQUES*

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*Research sponsored by the U.S. Atomic Energy Commission under contract with the Union Carbide Corporation.

IN-PILE PROPERTIES OF REACTOR FUELS BY OSCILLATING TECHNIQUES*

R. M. Carroll and O. Sisman

INTRODUCTION

The application of the oscillating technique to measure the physical properties of reactor fuels while producing power is a new experimental tool which we feel shows great promise. The scope of this paper is too limited to give a full discussion of the applications of the oscillating experiments. Our intentions are to describe the experimental method, and show a few examples of how the data are applied.

The irradiation facility was originally conceived for the measurement of fission-gas release during irradiation and it has been operated as a steady state experiment for some time. The steady state experiments have been described in detail.^{1,2,3,4} It was found necessary to add the oscillating technique to the existing experiment in order to gain the additional degree of freedom needed to evaluate a proposed fission-gas release model.⁵ However it is really necessary to be able to perform the steady state as well as the oscillation experiments. The steady state measurements are in essence a special application of the oscillation experiments in which the oscillation frequency is zero. It is easy to see that, in the case of the fission-gas release experiments, much additional information is obtained by operating over a range of frequencies and studying the resultant effect on the amplitude and wave shape of the fission-gas release curve.

*Research sponsored by the U.S. Atomic Energy Commission under contract with the Union Carbide Corporation.

The experiments are designed to have the following features:

(1) a high neutron flux to obtain significant radiation damage in a reasonable time; (2) precise control and measurement of the specimen temperature; (3) measured and controlled neutron flux; (4) the ability to start and stop the irradiation without affecting the reactor operation; (5) all working parts accessible for repair; (6) the ability to operate unmanned, requiring only occasional adjustments; and (7) maximum flexibility so that many types of specimens may be tested.

DESCRIPTION OF EXPERIMENT

The fission-gas release experiment consists of a single-pass sweep capsule hung vertically in an air-cooled irradiation tube extending into the reactor core of the ORR (see Figs. 1, 2, 3, and 4). The specimen is heated by its own fission power which is regulated by moving the capsule into or out of the reactor by a water-powered piston. Thermocouples monitor the specimen temperature, which is controlled by air-cooling the surface of the capsule. A sweep gas, obtained from a bottled supply, usually passes over the specimen continuously, carrying fission gas outside the reactor where it is measured by gamma-ray spectrometry. Usually, mixtures of helium, hydrogen, and argon are used as sweep gas. A trace of argon allows the neutron flux to be measured; using an argon activation technique,⁶ and a small amount of hydrogen prevents the UO_2 from being oxidized.

The important parts of the experimental equipment are the positioning system and method of flux oscillation, the temperature control and oscillation system, and the method of handling the output data from the fission-gas experiments. These will be discussed

Fig. 1. Flow Schematic for Oscillation Experiments.

Fig. 2. Specimen Capsule Details.

Fig. 3. Fission Gas Release Experiment in the ORR.

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Fig. 4. Arrangement of Oscillation Experiment in the ORR Building

in some detail. Other construction and design details, including safety features, may be found in a previous paper.⁷

Positioning System

The entire in-pile portion of the sweep gas system may be moved within the air-cooled experiment tube over a 61-cm span by a piston powered from the building water supply. The positioner is a double-acting hydraulic cylinder, fixed directly above the movable section of the experiment so as to exert an axial thrust. Water is used as the powering fluid so that any leaks will not contaminate the reactor pool water. The building water supply drives the positioner with a force of 700 lbs, which is more force than necessary but results in very smooth movements. The positioner and all parts of the experiment in contact with the reactor pool water are made of stainless steel. A sliding seal in the loading plug (see Fig. 3) prevents cooling air from escaping and bubbling into the reactor pool.

The hydraulic positioner contains a slide wire in a pressurized tube (to insure against water leaks). As the piston moves, a contact, fixed rigidly to the piston, passes over the slide wire (see Fig. 5). Since this piston is directly coupled to the specimen capsule, the specimen position is sensed by setting a regulated voltage across the slide wire and measuring the voltage between the sliding contact and either end of the slide wire. This converts the 61-cm mechanical motion to a voltage ranging from 0 to 35 Mv.

The signal from the positioner goes to a recorder-controller which contains a reference slide wire. The desired position of the experiment is set on the reference slide wire and an error detector

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Fig. 5. Schematic Diagram of Experiment and Control Systems.

in the controller compares the positioner signal to the reference signal which drives the controller. The controller emits an electrical current ranging from 1 to 5 ma and this is converted to a proportional 3 to 15 psig pneumatic signal which operates a 4-way valve controlling the differential water pressure across the positioner piston.

The oscillating-position programmer consists of a slide wire having a sliding contact whose motion is controlled by the follower arm on a cam. The cam is mounted on a shaft which is coupled through a gear train to the shaft of a constant-speed motor. The period of the reference position is determined by the gear ratios of the gear train and by a timer that has an adjustable ratio of on-off operation. The timer effectively changes the speed of the motor driving the gear train by starting and stopping the motor within a 30-second cycle, thus varying the ratio of the time the motor is running to the time the motor is stopped.

Over small position changes, a sinusoidal position variation will create a sinusoidal neutron flux variation and thus a sinusoidal variation of fission rate in the fuel specimen. The movement of the capsule has no detectable effect on the reactor period and the capsule position is changed at will. We have operated the experiment for oscillating periods as short as 24 seconds and as long as 20 hours. The capsule position may be regulated to closer than 1 mm of any desired position; actually as close as one can read the recorder.

Temperature Controls

The temperature of the specimen is determined by the fission rate within the specimen and the air-cooling rate over the outside of the capsule. The temperature is controlled very precisely during steady state operation by manually adjusting the pressure control valve of the air supply (see Fig. 5). Slight drifts in neutron flux can cause the temperature to change slowly during weekend and overnight runs. The temperature is maintained constant during this time by a simple closed loop system to regulate the air flow (shown in Fig. 5). This system is essentially the same as that used in controlling the positioner except that air rather than water is regulated through the control valve.

A programmer identical to that used for the positioner is used to oscillate the temperature of the specimen. The reference temperature of the circuit shown in Fig. 5 is thus oscillated over a selected axis, amplitude, and frequency. The feedback system regulates the air flow to force the specimen temperature to duplicate the reference temperature.

The construction of the specimen capsule (see Fig. 2) is such that the generated fission heat has to be transferred to the cooling air stream through several layers of materials. The thermocouples which measure the specimen temperature thus have a rapid response to a change of fission rate but a slower response to changes of air cooling on the capsule exterior. The differences in the response rates make it difficult to maintain a constant specimen

temperature during the times when the neutron flux is being oscillated. Also, the rate of heating and cooling of the capsule depends upon the fission rate in the specimen. Thus, when the specimen position is being oscillated, the cooling rate at the top of the stroke is slower than the cooling rate at the bottom. This problem has been minimized by installing a secondary automatic control system to regulate the air pressure into the control valve in the air cooling system (see Fig. 6). In the secondary air control system, the position programmer is used to feed a signal to a manually-controlled pressure regulator. The combined settings of the manual air control and the proportional position signal fixes the upstream pressure of the control valve.

The air-cooling rate is thus fixed by two control systems; the temperature signal, which regulates the control valve opening, and the position signal, which sets the upstream pressure (which has a baseline selected by a manual valve). For example, if the specimen temperature is to remain at 1100°C while the specimen position is being oscillated, the pressure upstream of the control valve may be at 45 lbs/in.^2 at the highest flux position and at 30 lbs/in.^2 at the lowest flux position. The oscillation of 15 lbs/in.^2 on the air pressure is controlled by the compensating system connected to the positioner while the base 30 lbs/in.^2 is set by the manual control. Both the air pressure baseline and oscillation are adjusted by the experiment operator to give a desired result.

The compensated-air-control system is adequate for slow flux cycles, where the period is 30 minutes or more. However, for faster

Fig. 6. Block Diagram of Air Cooling System Compensated for Position Change.

cycles a feed-forward system is being constructed. This system, added to the existing system, will allow the air-cooling to change before the position changes and thus compensate for the different response times. Essentially, when a change signal is given to the positioner, it will be delayed, but the air cooling will change at once. The delay time of the positioner response will be adjustable.

Neutron Flux Measurements

The method of measuring neutron flux by using an argon activation technique has already been described.⁶ The neutron flux is measured while the capsule is stationary and then the capsule is oscillated about the measured flux position. A sinusoidal movement of the capsule will not necessarily produce a sinusoidal oscillation of flux since the neutron flux does not vary linearly with distance. However, we have found that sinusoidal oscillations with amplitudes of about 1 cm will produce essentially sinusoidal oscillations of neutron flux. This is determined by using a sweep gas containing argon and then measuring the wave shape of the argon oscillation as the capsule is oscillated.

During neutron flux oscillations, a sweep gas containing about 1% argon is used. The oscillation of the specimen in the reactor sets up waves of argon activity in the sweep gas. Since the time-position relation of the experiment is known and the time-activity relation of the argon can be measured, we can find the length of time required for the sweep gas to travel between the capsule and the analysis point.

Fission-Gas Analysis

All of the quantitative data are obtained from gas samples gathered in serum bottles. These bottles are positioned above a 3-in. x 3-in. NaI(Tl) scintillation detector in a graded (shield lined with cadmium and copper) lead shield. The detector output is fed to a preamplifier, and then to a model DD-2 linear amplifier, and the amplified pulse is fed to a 512-channel pulse-height analyzer. A beryllium absorber is placed between the NaI(Tl) crystal and the sample bottle to ensure that only gamma rays are measured by the scintillation detector. The amounts of the fission gases are calculated from the gamma-ray spectrum recorded by the pulse-height analyzer. These routine calculations are simplified by using the same size samples and a fixed counting geometry.

The concentration of fission gas in the sweep gas is inversely proportional to the flow rate of the sweep gas, and therefore the activity of the gas sample can be regulated by varying the sweep gas flow rate. The time between the release of the fission gas from the specimen and the time of sampling is determined by argon activation, as explained above.

During oscillation programs a continuous on-line analysis of the flowing fission gas is made. A U-shaped counting tube (see Fig. 1) is located near a shielded 3-in. x 3-in. NaI(Tl) scintillation detector. The analyzer is set to consider only a single gamma-ray energy, usually the 2.4 MeV gamma of ^{88}Kr , and to operate in a scalar-time mode wherein the analyzer collects the total number of counts in the selected energy range for a selected time and stores

this in successive channels of its memory. Since the time interval of each channel is the same (and is known), this counting process gives an activity-time record. These data are made quantitative by determining the atoms-per-second release corresponding to the activity at one or more points with serum-bottle samples.

Oscillation Data

During oscillations, the time-temperature or time-specimen position is measured and recorded by a Dymec integrating digital voltmeter driven by a Hewlet-Packard digital clock. A flip-flop circuit has recently been added to the system so that temperature and position data can be alternately measured in the thermal conductivity experiments. The data can be printed and/or punched into paper tape by a teletype punch. The tape contains a 4-digit time signal, a 5-digit data signal, and a single digit to show whether temperature or specimen position is being recorded.

The time-activity data obtained from the pulse-height analyzer may be transferred to an x-y plot, a digital readout, or a punch tape readout. The punch tape records the channel number and a 6-digit data signal. The channel number can be related to a time so, in essence, time-activity data are recorded.

APPLICATIONS OF OSCILLATING DATA

Fission-Gas Release

The oscillating experiments were conceived for the purpose of developing a mathematical model for the defect-trap theory for fission-gas release³ and for analyzing the mathematical expressions which contained too many variables to be analyzed by steady state experiments.

As we have said earlier, the purpose of this paper is to describe the experimental techniques and not the theory of fission-gas release. However, we would like to illustrate with a few qualitative observations the kind of results which the oscillating experiments will give that are not possible with steady state experiments.

During irradiation some fission-gas is released because of fissions on the surface of the fuel. At temperatures above 700°C, the fission-gas release increases exponentially with temperature and at lower temperatures the gas release is almost independent of temperature (Fig. 7). We have assumed that the amount of gas released by surface fissions is independent of temperature. However, since both processes occurred at the same time, steady state measurements provided no real way to prove that the amount of gas released by surface fissions was independent of temperature. In fact, there were some suggestions that only one process of gas release was involved (surface fissions) and that this process became temperature dependent.

During oscillations of fission rate, the number of surface fissions oscillate also. If the temperature of the specimen is maintained constant and there are two release processes, one temperature-dependent and the other flux-dependent, then the flux-dependent process should oscillate. The results of an experiment in which the neutron flux was oscillated while the specimen was maintained at 840°C is shown in Fig. 8. The double peak in the gas release curve shows that two processes are indeed occurring. The separation of the peaks shows that the processes have different time dependencies.

Fig. 7. Release Rate of ^{88}Kr from Fine Grain Specimen (Cl-12) of UO_2 .

Fig. 8 UO_2 Single Crystal, Cl-13-8-8, 10 hr Flux Cycle, 840°C.

The large peaks of Fig. 8 can be shown to be caused by surface fissions. If these peaks are temperature-dependent then a reduction of temperature would reduce the size of the peaks.

An oscillation of neutron flux while the specimen was at a lower temperature is shown in Fig. 9. The temperature of 575°C was selected because it is low enough so that the gas release is mostly caused by surface fissions. Therefore the gas release of Fig. 9 shows only single waves, caused by the oscillations of fission rate. However the single waves are nearly as big as the large peaks of Fig. 8. If the gas release caused by surface fission was temperature-dependent, then the waves of Fig. 8 would be more than twice as large as they actually are. This result clearly shows that the fission-gas release caused by the surface fissions is not temperature-dependent and that two distinct release processes are occurring at temperatures over 700°C.

Thermal Conductivity

Thermal conductivity is one of the most difficult of in-pile measurements. Mainly, it is difficult to measure the specimen temperature in a high thermal gradient. We are making thermal conductivity measurements using a hollow cylinder of single crystal UO_2 (Fig. 10) in an insulated capsule. The specimen is oscillated in the neutron flux and the temperature response of the specimen is measured only along the axis, where thermal gradients are small.

When the fission rate in the specimen is oscillated the temperature response has a time lag (phase shift) that depends on the

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Fig. 9 UO₂ Single Crystal, CI-13-9-29, 3 hr Flux Cycle, 575°C.

Fig. 10. A Hollow Cylinder of Single Crystal UO₂ in an Insulated Capsule.

specific heat and thermal conductivity of the specimen. This phase shift is established very precisely by averaging a series of oscillations. When the frequency of the oscillation is changed, the amplitude of the temperature response is also changed, again depending on specific heat and conductivity. The higher frequencies result in lower amplitude temperature responses. By shifting the axis of cooling rate or position, the temperature responses can be measured as a function of fission rate and temperature.

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V. CIRCULATING LIQUID METAL EXPERIMENTS

DESIGN AND FABRICATION
OF A LIQUID METAL TEMPERATURE CONTROLLED
IRRADIATION CAPSULE*

by

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DESIGN AND FABRICATION
OF A LIQUID METAL TEMPERATURE CONTROLLED
IRRADIATION CAPSULE

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I. INTRODUCTION

The Fast Neutron Breeder Reactor Program has been initiated to develop a new reactor concept which will improve the overall economical potential of nuclear power stations through improved utilization of nuclear fuel resources. The primary coolant considered for the reactor concept is liquid sodium.

Recent studies have indicated that sodium environments at temperatures to 1200 F may be feasible with resulting fuel clad temperature of 1450 F. In addition, it has been determined that the sodium environment itself may affect the mechanical properties of fuel cladding and reactor structural materials. It is well known that neutron irradiation in thermal reactors cause marked changes in these properties and it is possible that even greater changes may be associated with irradiation in fast neutron reactors.

To determine the effects of irradiation upon the mechanical and physical properties of reactor structural materials, it is necessary to separate the effects of the irradiation environment from the effects of the neutron irradiation. To accomplish this for sodium cooled reactors, irradiations must be conducted in an environment typical of the reactor coolant whereby velocity and mass flow effects are considered. Since such sodium environments are not readily available for irradiation testing, especially at elevated temperatures to 1450 F, a temperature controlled, liquid metal capsule,

utilizing convective flow was designed and fabricated for use in the Engineering Test Reactor. This work is in support of a program to determine the effects of irradiation upon reactor metals and is sponsored by the Division of Reactor Development of the Atomic Energy Commission.

II. CAPSULE DESIGN AND FABRICATION

The primary purpose of the capsule was to provide first a flowing liquid metal environment past an assembly of corrosion and mechanical testing coupons and second to provide a means of controlling the maximum temperature of the coolant in the capsule. Both of these objectives were to be accomplished in a simple, direct, and economical manner and thus out-of-reactor liquid metal pumps, heat exchangers, heaters, etc., were not considered.

The capsule may be divided into three portions, as shown schematically in Figure 1. The inner portion contains the specimen chamber and a vacuum insulation layer. The next layer is an annulus through which the liquid metal coolant can flow, and a third portion may be considered as the capsule inner and outer containment walls and the temperature control gas annulus between these walls.

Flow past the specimens was accomplished by removing the energy generated in the specimens by gamma heating in the reactor by a combination of convective and conductive heat transfer. By placing a vacuum insulating layer around the specimen chamber, the heat generated in the specimens must be transferred axially up the capsule. The resulting decrease in coolant density causes the coolant to rise in the capsule. The cooler return annulus surrounding the specimen chamber completes the convective circuit shown in Figure 1.

LIQUID METAL ENVIRONMENTAL CAPSULE

Figure 1

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Schematic of a Liquid Metal Environ-
ment Capsule

Heat is also allowed to leave the capsule by conductive heat transfer radially through the capsule walls. A temperature drop is associated with each of the capsule walls and the gas annulus separating them. The temperature drop across the gas annulus is, of course, much greater than the temperature decreases associated with the capsule walls because of the relative thermal conductivity of the materials. By varying the gas composition within this annulus a wide variation in capsule temperatures can be achieved.

Figure 2 shows the as-built construction and assembly drawing for the liquid metal capsule. Components of the capsule were fabricated from AISI 304 stainless steel to the tolerances specified on Figure 2. Assembly of the components was accomplished by tungsten inert gas welding using no filler metal, followed by radiographic examination and helium leak testing to insure weld reliability.

The specimen chamber shown as part 2 on Figure 2 provides a volume 1.620-inches in diameter and 15.75 inches in length for the insertion of test specimens. Space between the two cylinder walls was evacuated through a small hole which was subsequently welded closed by electron beam welding in the vacuum. This vacuum layer provides the necessary thermal barrier for convective flow within the capsule. Figure 3 shows the specimen container components prior to assembly.

In order to monitor the temperature distribution within the capsule, four chromel-alumel thermocouples sheathed in 1/16th-inch diameter 304 stainless steel were positioned at the bottom, top, and midsection of the specimen chamber. One additional thermocouple was placed in the midsection of the coolant return annulus. It was also necessary to conduct both

Figure 2: As-Built Drawing and Assembly of a Liquid Metal Environment Capsule

Figure 3 Specimen Container Components Prior to Assembly

Figure 4 Experimental Apparatus for Induction Brazing Capsule Seals

argon and helium gas into the capsule and provide for exiting these gases through a single tube. Stainless steel tubing 1/8-inch in diameter was used for this application.

To conduct the gas tubes and thermocouples into the capsule it was necessary to penetrate into the capsule through the end caps parts 4 and 7, as shown in Figure 2. To seal the thermocouple sheathing and tubing to these end capsules an induction brazing technique was developed using a commercial brazing compound Coast Alloy 52. The experimental brazing setup is shown in Figure 4. The end caps were prealigned and positioned prior to the brazing operation. Small depressions were machined into the end caps around each penetration hole to contain sufficient brazing compound. The entire end cap was heated by induction to a temperature of about 1950 F, or slightly over the melting point of the brazing compound. After brazing the resulting seals were helium leak tested to insure no leaks were present.

Subsequent to the fabrication of the components and assembly of the specimen chamber, parts were assembled as shown in Figure 2. The as-fabricated components are shown in Figure 5 prior to assembly.

Actual assembly of the capsule components consisted of welding the end cap, part 6, to part 12. This assembly when welded to the end cap, part 5, acts as a pedestal supporting the specimen assembly and provides a chamber into which the temperature control gas flows prior to exiting the capsule. This chamber is also designed to hold the liquid metal volume should a leak occur in the region of the specimen chamber. The specimen assembly itself, shown in the foreground of Figure 5, is then inserted in the specimen chamber. The thermocouple and gas tube assembly previously described is inserted into the specimen chamber. The inner liquid metal tube and exit gas tube were then

welded to the end caps, parts 5 and 4. The pedestal tube was then welded to the end cap, part 5, completing the interior assembly. Finally, this outer tube, part 8, was placed around the entire assembly and welded to the top and bottom end caps, parts 6 and 7. The final assembly is shown in Figure 6.

Temperature control of the capsule is attained by varying the mixture of gas flowing in the gas annulus. Because of their relatively high and low thermal conductivities, respectively, helium and argon gas were used. Figures 7 and 8 show the gas flow and electrical control systems used in the temperature control system. To insure that a relatively pure gas enters the capsule, it is first passed over heated copper turnings and thence through a molecular sieve. Either pure helium, pure argon, or a mixture of the two gases can be manually metered into the system depending upon the desired temperature in the capsule.

Safety circuits have been designed into the system whereby warning signals are activated if a low gas pressure is sensed. In addition, if the capsule temperature exceeds that set on the temperature recorder, a signal is activated, and solenoid valve V-4 opens an auxiliary helium line which is at higher pressure than the primary helium and argon gas line. Helium will then flush through the capsule lowering its temperature.

III. HEAT TRANSFER AND FLUID FLOW ANALYSIS

The analysis leads to the analytical determination of the maximum temperature to be expected during irradiation of the capsule in the ETR where the average gamma heat generation is assumed to be 13 watts per gram. The technique employed in these calculations is summarized as follows:

Figure 5 The As-Fabricated Components of the Liquid Metal Environment Capsule Prior to Assembly

Figure 6 Assemble of the Liquid Metal Environment Capsule Prior to Irradiation

Figure 7

5.1.12

NOTES

1. K5 Unlatches K4 thus Changes the Position of (K4-1) and (K4-2)
2. He Pressure Sw-NC at Atmospheric Pressure
3. Hi-Temp. Dump Valve - NC Solenoid Valve
4. Recorder CAM-1-N0 to Close on Temp. Rise Before CAM-2
5. Recorder CAM-2-N0 to Close on Temp. Rise After CAM-1
6. TC Leads to tie Directly into Recorder

Figure 8 ETR GEH-22 Experiment Console Electrical Schematic

1. Determination of the heat generated in each portion of the capsule from the 13 watts/gram gamma heat.
2. Make an initial assumption concerning the sodium mass flow rate.
3. On the basis of this initial assumption, calculate the heat transfer coefficients for sodium and container walls.
4. Evaluate the temperature distribution as a function of position in the capsule and calculate the average temperatures and sodium density.
5. Determine the friction factors and compute the associated pressure drops.
6. Compare the pressure differential available with the calculated pressure drop and repeat the analysis with a new assumed mass flow rate until they compare.

For the convenience of the reader, equations used in this analysis are listed below.

Heat Transfer to Sodium in Specimen Chamber

Assume uniform heat generation in test specimens and in liquid sodium.

Temperature rise of sodium

$$\Delta T = \frac{\sum q_i}{\dot{m} C_p}$$

T = Temperature, °C (1)

$\dot{m}$ = mass flow rate, g/sec

C_p = Specific heat

q_i = Total heat generation of each material in specimen chamber

Heat Transfer from Sodium to Water

$$-\dot{m} C_p \frac{dT}{dX} = u \left(\frac{dA_3}{dX} \right)_{\text{surface}} (T_3 - T_4) - q'_3 A_3 \quad (2)$$

u = overall heat transfer coeff. from sodium to water

q'_3 = effective heat generation for sodium in return

= heat gen in sodium + heat gen in outer vacuum container wall
length of return

= watt
cm

A_3 = cross section area of Na in return, cm^2

$\frac{dA_3}{dX}$ = Surface area of inner container wall/unit length

Flow Calculations

$$(\bar{p}_3 - \bar{p}_2)l = \frac{G_2^2}{2\rho_2 g} \left[\left(\frac{\xi l}{d} \right)_2 + \text{Entrance and Exit losses} \right] + \frac{G_3^2}{2\rho_3 g} \left[\left(\frac{\xi l}{d} \right)_3 + \text{Entrance and Exit losses} \right]$$

ρ = density

(3)

l = length of specimen chamber and return

G = specific mass flow rate, $\frac{\text{gm}}{\text{cm}^2 \text{sec}}$

g = gravity

ξ = friction factor

d = hydraulic dia. of regions

The results of the calculations given in detail in Appendix A of this report are shown in Figure 9. The maximum calculated capsule temperature is 1210 F (655 C). The maximum temperature drop across the test specimens was estimated at approximately 620 F (343 C).

IV. IRRADIATION TESTING PERFORMANCE

The capsule containing 18 tensile specimens of austenitic stainless steel was inserted into the Engineering Test Reactor (ETR) M-7 position at the beginning of cycle 78. Prior to reactor startup, all instrumentation and gas lines were connected and pure helium circulated through the gas annulus. Upon reactor startup the capsule temperature was monitored as the reactor power level increased to a maximum of 175 MW. The results of these observations are given in Figure 10. As shown in Figure 10, a uniform temperature rise was observed with increasing reactor power level. At the maximum reactor power level of 175 MW the maximum steady state temperature was 1060 F.

To determine the effective temperature control capabilities of the capsule, the gas composition was varied at a constant reactor power level of 175 MW from pure helium to pure argon. The change in capsule temperature level distribution is shown in Figure 11. As the volume percent argon was increased a linear increase in temperature was observed in the capsule at each position being monitored. At 100 percent argon the maximum capsule temperature monitored was 1350 F. This is an increase of 290 F over the temperature recorded for pure helium and represents a substantial temperature control range in which to conduct irradiation experiments.

V. CONCLUSIONS

As a result of the design and developments described in this report, a liquid metal capsule capable of wide variations in temperature control has been fabricated and successfully operated in the ETR-M7 irradiation facility. This represents the first irradiation of this type of capsule in

Figure 9

5.1.16

5.1.17

ETR START-UP DATA
LIQUID METAL TEMPERATURE CONTROLLED IRRADIATION CAPSULE
(Pure Helium)

TEMPERATURE VS GAS COMPOSITION ETR LIQUID METAL TEMPERATURE CONTROLLED IRRADIATION CAPSULE

Figure 11

which the major goals of the irradiation were successfully met. The capsule reached its design temperature, exhibited a wide range of temperature control capability, and provided a flowing liquid metal environment to mechanical test specimens being irradiated within its specimen chamber.

Future capsules of this type using sodium coolants for the test environments are now being fabricated. It is hoped that by use of such irradiation capsule facilities an insight into the effects of irradiation and environment for liquid sodium cooled fast breeder reactor systems can be achieved.

APPENDIX A

HEAT TRANSFER AND FLUID FLOW ANALYSIS FOR LIQUID METAL TEMPERATURE CONTROLLED IRRADIATION CAPSULE - GEH 22-2

The analysis leads to the analytical determination of the maximum temperature to be expected during irradiation of the capsule in the ETR where the average gamma heat generation is 13 watts per gram.

I. The equations used in this analysis are listed below.

Heat Transfer to Sodium in Specimen Chamber

Assume uniform heat generation in test specimens and in liquid sodium.

Temperature rise of sodium

$$\Delta T = \frac{\sum q_i}{\dot{m} C_p}$$

T = Temperature, °C

$\dot{m}$ = mass flow rate, g/sec

C_p = Specific heat

q_i = Total heat generation of each material in specimen chamber

Heat Transfer from Sodium to Water

$$-\dot{m} C_p \frac{dT}{dX} = u \left(\frac{dA_3}{dX} \right)_{\text{surface}} (T_3 - T_4) - q'_3 A_3$$

u = overall heat transfer coeff. from sodium to water

q'_3 = effective heat generation for sodium in return

= $\frac{\text{heat gen in sodium} + \text{heat gen in outer vacuum container wall}}{\text{length of return}}$

= $\frac{\text{watt}}{\text{cm}}$

A_3 = cross section area of Na in return, cm^2

$\frac{dA_3}{dX}$ = Surface area of inner container wall/unit length

Flow Calculations

$$(\bar{p}_3 - \bar{p}_2) \ell = \frac{G_2^2}{2\rho_2 g} \left[\left(\frac{\xi \ell}{d} \right)_2 + \frac{\text{Entrance and Exit losses}}{\quad} \right] + \frac{G_3^2}{2\rho_3 g} \left[\left(\frac{\xi \ell}{d} \right)_3 + \frac{\text{Entrance and Exit losses}}{\quad} \right]$$

ρ = density

ℓ = length of specimen chamber and return

G = specific mass flow rate, $\frac{\text{gm}}{\text{cm}^2 \text{sec}}$

g = gravity

ξ = friction factor

d = hydraulic dia. of regions

II. Heat Generation Assuming Gamma Heating at β watts/gm

A. Heat Gen in Test Specimens

$Q = q \text{ weight}$

= $13 \frac{\text{watts}}{\text{gm}} (402 \text{ gm})$

= 5220 watts

B. Inner vacuum container wall

$Q = q w = \left(13 \frac{\text{watts}}{\text{gm}} \right) (92.6 \text{ cc}) \left(7.98 \frac{\text{gm}}{\text{cc}} \right)$

= 9620 watts

C. Specimen Holder

$$Q = q w = \left(13 \frac{\text{watts}}{\text{gm}}\right) (153 \text{ gm}) = 1990 \text{ watts}$$

D. NaK Alloy

$$Q = (472 \text{ cm}^3) \left(0.703 \frac{\text{gm}}{\text{cm}^3}\right) \left(13 \frac{\text{watts}}{\text{gm}}\right) \\ = 4310 \text{ watts}$$

E. Total heat input to NaK during travel in the test section is therefore

$$\begin{array}{r} 5220 \\ 9620 \\ 1990 \\ 4310 \\ \hline 21,140 \text{ watts} \end{array}$$

F. Heat Generation in Return Section/cm

1. Outer Vacuum Sheel

$$Q = \dot{q} w = \left(13 \frac{\text{watt}}{\text{gm}}\right) 103 \text{ cc} \left(7.98 \frac{\text{gm}}{\text{cc}}\right) \\ = 10,700 \text{ watt}$$

2. NaK Alloy

$$Q = \dot{q} w = 193 \text{ cc} \left(0.703 \frac{\text{gm}}{\text{cc}}\right) \left(13 \frac{\text{watt}}{\text{gm}}\right) \\ = 1760 \text{ watt}$$

G. Effective heat generation of NaK per unit volume including heat generation of S.S. Containers

$$\frac{\text{Effective Heat}}{\text{unit vol}} = \frac{10,700 + 1760 \text{ watts}}{193 \text{ cc}} = 59.4 \frac{\text{watts}}{\text{cm}^3}$$

II. Temperature Difference Between Inner Container Wall and Water Coolant

Note: This will give the effective temperature of the water. The heat generation in the outer walls may then be neglected. This is possible because the equations are linear and solutions may be added together.

A. Internal Heat Generation for Flat Plate

$$T = -\frac{qy^2}{2K} + C_1y + C_2$$

Let $T = T_{\max}$ at center

$$\frac{dT}{dx} = 0 \text{ at center}$$

$$\therefore C_2 = T_{\max} \text{ and } C_1 = 0$$

$$T_m - T_s = \frac{\dot{q}S^2}{2K}$$

K = thermal conductivity = $0.16 \frac{\text{watts}}{\text{cm } ^\circ\text{K}}$ for S. Steel

S = wall thickness = 0.165 cm

$\dot{q}$ = heat gen in wall/unit volume

B. ΔT Across Inner Wall

$$T_m - T_s = \frac{\left(13 \frac{\text{watts}}{\text{gm}}\right) \left(7.98 \frac{\text{gm}}{\text{cc}}\right) (0.165 \text{ cm})^2}{2 \left(0.16 \frac{\text{watts}}{\text{cm } ^\circ\text{K}}\right)} = 8.9 ^\circ\text{K}$$

C. ΔT Across the Wall

$$\Delta T = \frac{qx}{K} = \frac{13 \frac{\text{watts}}{\text{gm}} \left(7.98 \frac{\text{gm}}{\text{cm}^3}\right) (0.165 \text{ cm}) (0.00508)}{\left(1.59 \times 10^{-3} \frac{\text{watts}}{\text{cm } ^\circ\text{C}}\right)}$$

$$= 54.8 ^\circ\text{C}$$

D. ΔT across Outer Wall

$$\Delta T = 8.9 + \frac{13 \frac{\text{watts}}{\text{gm}} \left(7.98 \frac{\text{gm}}{\text{cm}^3}\right) (0.165 \text{ cm})^2}{0.16 \frac{\text{watts}}{\text{cm } ^\circ\text{C}}}$$

$$= 26.7 ^\circ\text{C}$$

- E. Total temperature difference across wall due to its internal heat generation

$$\Delta T = \frac{55}{9} = \frac{27}{91}^{\circ}\text{C}$$

This has the effect of raising the water temperature of $120^{\circ}\text{F} = 49^{\circ}\text{C}$ to $140^{\circ}\text{C} = T_4$.

Note the outer surface of the vessel is assumed to be at the water temperature. The film drop is described later in this analysis.

III. Thermal Properties of NaK

A. Thermal Conductivity = $K = 0.21 \frac{\text{watt}}{\text{cm}^{\circ}\text{C}}$

B. Density $\rho = 0.703 \text{ gm/cc}$

C. Specific Heat $C_p = 0.88 \frac{\text{watt-sec}}{\text{gm}^{\circ}\text{C}}$

D. Viscosity = $\mu = 0.15 \times 10^{-2} \frac{\text{gm}}{\text{cm-sec}}$

- IV. Calculations to determine the hydraulic diameter and area of regions 2 and 3 shown below.

$$d_3 = D_o - D_i = (2.222 - 2.00) 2.54$$

$$d_3 = 0.564 \text{ cm}$$

$$A_3 = (D_o^2 - D_i^2) \frac{\pi}{4} = (2.222^2 - 2.000^2) \frac{\pi}{4} \left(2.54^2 \frac{\text{cm}^2}{\text{in.}^2} \right)$$

$$= 4.75 \text{ cm}^2$$

$$d_2 = \frac{4A_2}{\rho} = \frac{[1.62^2 - 7(0.3)^2] \pi}{\pi [1.62 + 7(0.3)]} = 0.538 \text{ in.}$$

$$d_2 = 1.37 \text{ cm}$$

$$A_2 = [1.62^2 - 7(0.3)^2] \frac{\pi}{4} \left(6.45 \frac{\text{cm}^2}{\text{in.}^2} \right)$$

$$A_2 = 10.1 \text{ cm}^2$$

V. Calculation of the Overall Heat Transfer Coefficient

A. Assume a mass flow rate of 70 gm/sec

$$G_3 = \frac{\text{in.}}{A_3} = \frac{70 \text{ gm/sec}}{4.75} = 14.75 \frac{\text{gm}}{\text{cm}^2 \text{ sec}}$$

$$G_2 = \frac{\text{in.}}{A_2} = 6.94 \frac{\text{gm}}{\text{cm}^2 \text{ sec}}$$

$$\text{Re}_3 = \frac{G_2 d_2}{\mu} = 5550$$

$$\text{Re}_2 = \frac{G_2 d_2}{\mu} = 6340$$

B. Change in NaK temperature while in bottom reservoir

$$\Delta T = \frac{q}{\dot{m} C_p} = \frac{44.6 \text{ gm} \cdot 13 \frac{\text{watt}}{\text{gm}}}{70 \frac{\text{gm}}{\text{sec}} \left(0.88 \frac{\text{watt-sec}}{\text{gm } ^\circ\text{C}} \right)}$$

$$= 9.44 ^\circ\text{C} \quad \text{Use } 10 ^\circ\text{C}$$

Use same temperature for top reservoir

C. ΔT in test section

$$\Delta T = \frac{\dot{q}}{\dot{m} C_p} = \frac{21,140 \text{ watts}}{70 \frac{\text{gm}}{\text{sec}} \left(0.88 \frac{\text{watt sec}}{\text{gm } ^\circ\text{C}}\right)}$$

$$= 343 ^\circ\text{C}$$

Heat transfer coefficient for NaK Alloy correlation for $10^4 < \text{Re} < 6.5 \times 10^5$

$$\text{Nu}_m = 4.5 + 0.014 (\text{Re Pr})^{0.4}$$

Ref. "High Temperature Carriers," A. V. Chechetkin, eq. 9.5.

Since this is out of the range of these Reynolds numbers, a conservative value for Na would be 4.5.

$$h_3 = \frac{4.5K}{d_3} = \frac{4.5 \left(0.21 \frac{\text{watt-sec}}{\text{gm } ^\circ\text{C}}\right)}{d_3}$$

$$h_3 = 1.68 \frac{\text{watt}}{\text{cm}^2 ^\circ\text{C}}$$

D. Evaluate overall heat transfer coefficient "u" for two outer walls, the gap, and Na film coeff.

Walls

$$h \cong \frac{K}{X} = \frac{0.16 \frac{\text{watts}}{\text{cm } ^\circ\text{K}}}{0.165 \text{ cm}} = 0.97 \frac{\text{watt}}{\text{cm}^2 ^\circ\text{C}}$$

He Gap

$$h = \frac{K}{X} = \frac{0.09 \frac{\text{Btu}}{\text{ft hr } ^\circ\text{F}}}{(0.00508 \text{ cm}) 57.8 \frac{\text{Btu}}{\text{ft hr } ^\circ\text{F}} / \frac{\text{watt}}{\text{cm } ^\circ\text{F}}}$$

$$= 0.307 \frac{\text{watt}}{\text{cm}^2 ^\circ\text{C}}$$

Therefore,

$$\frac{1}{u} = \sum \frac{1}{h} = \frac{2}{0.97} + \frac{1}{0.307} + \frac{1}{1.68} = 5.92$$

$$u = \frac{1}{5.92} = 0.169 \frac{\text{watt}}{\text{cm}^2 ^\circ\text{C}}$$

$$\left(\frac{dA_3}{dX}\right)_{\text{surface}} = \pi \text{ dia} = \pi(2.22) \left(2.54 \frac{\text{cm}}{\text{in.}}\right) = 17.7 \text{ cm}$$

VI. Heat Transfer from Sodium to Water.

Temperature change of NaK = heat transferred away + heat added due to heat generation.

$$-\dot{m} C_p \frac{dT_3}{dX} = u \left(\frac{dA_3}{dX}\right) (T_3 - T_4) - q'_3 A_3$$

$$\frac{dT_3}{dX} + \frac{u}{\dot{m} C_p} \frac{dA_3}{dX} T_3 = \frac{u}{\dot{m} C_p} \frac{dA_3}{dX} T_4 + \frac{q'_3}{\dot{m} C_p} A_3$$

Solving

$$T_3 = C_i e^{ax} + T_4 + \frac{q'_3 A_3}{u \frac{dA_3}{dX}}$$

Where $a = -\frac{u}{\dot{m} C_p} \frac{dA_3}{dX}$

At $X = 0$, $T_3 = T_{3i}$, which is unknown at this point.

$$\therefore C_i = T_{3i} - T_4 - \frac{q'_3 A_3}{u \frac{dA_3}{dX}}$$

$$\therefore T_3 = \left(T_{3i} - T_4 - \frac{q'_3 A_3}{u \frac{dA_3}{dX}}\right) e^{ax} + T_4 + \frac{q'_3 A_3}{u \frac{dA_3}{dX}}$$

$$\frac{q'_3 A_3}{u \frac{dA_3}{dX}} = \frac{59.4 \frac{\text{watts}}{\text{cm}} (4.75 \text{ cm}^2)}{0.169 \frac{\text{watt}}{\text{cm}^2 \text{ } ^\circ\text{C}} (17.7 \text{ cm})} = 94.2 \text{ } ^\circ\text{C}$$

$$a = -\frac{u}{\dot{m} C_p} \frac{dA_3}{dX} = \frac{-0.169 (17.7 \text{ cm})}{70 \frac{\text{gm}}{\text{sec}} \left(0.88 \frac{\text{watt sec}}{\text{gm } ^\circ\text{C}}\right)} = -0.0488/\text{cm}$$

$$e^{ax} = e^{-0.0488 (40.6)} = e^{-1.98} = 0.138$$

$$T_4 + \frac{q_3 A_3}{\mu \frac{dA_3}{dX}} = 140^\circ + 94^\circ = 234^\circ \text{C}$$

Assume $T_{3i} = 655^\circ \text{C}$

$$T_3 \text{ at exit of return} = (655 - 234)(0.138) + 234$$

$$T_{3e} = 292^\circ \text{C}$$

In Summary,

Using $T_3 = 421 e^{-0.0488x} + 234$

$$T_{3Av} = \frac{\int_0^L T_3 dx}{L} = \left[\frac{-421}{0.0488} e^{-0.0488x} + 234x \right]_0^L$$

for $L = 40.6 \text{ cm}$

$$\begin{aligned} T_{3Av} &= \frac{\frac{-421}{0.0488} (0.148) + 234 (40.6) + \frac{421}{0.0488}}{40.6} \\ &= \frac{-1280 + 9520 + 8640}{40.6} = \frac{16,880}{40.6} \\ &= 414^\circ \text{C} \end{aligned}$$

Average Temperature of Test Specimen Side

$$T_{2Av} = \frac{302 + 645}{2} = 473^\circ \text{C}$$

VII. Density Calculations

Using the equation for density

$$\rho = 0.859 - 2.58 \times 10^{-4} (T-100) \text{ gm/cm}^3$$

$$T = ^\circ\text{C}$$

$$\rho = \text{gm/cm}^3$$

$$\begin{aligned}\rho_{3\text{Av}} &= 0.859 - 2.58 \times 10^{-4} (473 - 100) \\ &= 0.778 \text{ gm/cm}^3\end{aligned}$$

$$\begin{aligned}\rho_{2\text{Av}} &= 0.859 - 2.58 \times 10^{-4} (473 - 100) \\ &= 0.763 \text{ gm/cm}^3\end{aligned}$$

VIII. Pressure head available to circulate NaK

$$\begin{aligned}\Delta\rho &= (\rho_{3\text{Av}} - \rho_{2\text{Av}}) \ell = (0.778 - 0.763)(40.6) \\ &= 0.609 \text{ gm/cm}^2\end{aligned}$$

for friction pressure losses

$$\xi = \frac{0.316}{\text{Re}^{0.25}}$$

$$\xi_3 = \frac{0.316}{(5550)^{1/4}} = 0.0365$$

$$\xi_2 = \frac{0.316}{(6340)^{1/4}} = 0.0355$$

$$\Delta\rho = \frac{G^2}{2g\rho} \left(\frac{\xi \ell}{d} + \text{Entrance and Exit losses} \right)$$

Assume 1 for Entrance and Exit losses

$$\begin{aligned}\Delta\rho_2 &= \frac{(6.94)^2}{2(980)(0.763)} \left[\frac{(0.0355)(40.6)}{1.37} + 1.0 \right] \\ &= 0.067 \frac{\text{gm}}{\text{cm}^2}\end{aligned}$$

$$\Delta p_3 = \frac{(14.75)^2}{2(980)(0.778)} \left[\frac{0.0365(40.6)}{0.564} + 1.0 \right]$$

$$= 0.52 \text{ gm/cm}^2$$

Total pressure drop through friction

$$\Delta p = 0.52 + 0.067 = 0.587 \text{ gm/cm}^2$$

The problem has effectively converged for a mass flow rate of 70 gm/sec of NaK.

IX. Temperature difference between sodium and test specimens

A. For internal heat generation in a cylinder

$$T_{\text{max}} - T_{\text{surface}} = \frac{GR^2}{4K_f}$$

Assume 2 dimensional heat flow

G = power generation function volume

R = radius

K = conductivity

$$G = 13 \frac{\text{watt}}{\text{gm}} (7.98 \text{ gm/cc})$$

$$= 104 \frac{\text{watt}}{\text{cc}}$$

$$R_1 = \frac{0.20}{2} (2.54) = 0.254 \text{ cm}$$

$$R_2 = \frac{0.375}{2} (2.54) = 0.475 \text{ cm}$$

$$K = 13 \frac{\text{Btu}}{\text{ft hr } ^\circ\text{F}} \cdot 1.73 \times 10^{-2} \frac{\text{watts/cm } ^\circ\text{K}}{\text{Btu/ft hr } ^\circ\text{F}} = 0.225 \frac{\text{watts}}{\text{cm } ^\circ\text{K}}$$

$$T_{\text{max}} - T_s = \frac{104 \frac{\text{watt}}{\text{cm}^3} R^2}{4 \left(0.225 \frac{\text{watt}}{\text{cm } ^\circ\text{K}} \right)} = 115 \frac{^\circ\text{C}}{\text{cm}^2} R^2$$

Position 1.

$$T_m - T_s = 115 \frac{^{\circ}\text{C}}{\text{cm}^2} (0.254)^2 = 7.4 ^{\circ}\text{C}$$

Position 2.

$$T_m - T_s = 115 \frac{^{\circ}\text{C}}{\text{cm}^2} (0.475)^2 = 26 ^{\circ}\text{C}$$

B. ΔT across boundary layer of NaK

Using $\text{Nu} = 4.5$

$$\begin{aligned} h &= 4.5 \frac{\text{K}}{\text{d}} = \frac{4.5}{1.37 \text{ cm}} \left(0.21 \frac{\text{watt}}{\text{cm}^{\circ}\text{C}} \right) \\ &= 0.69 \frac{\text{watts}}{\text{cm}^2 ^{\circ}\text{C}} \end{aligned}$$

Volume of test specimen/cm

$$V = R^2 \pi$$

$$V_1 = (0.254)^2 \pi = 0.202 \text{ cm}^3$$

$$V_2 = (0.425)^2 \pi = 0.708 \text{ cm}^3$$

ΔT across boundary layer

$$\Delta T = \frac{q}{hA}$$

$$= \frac{GR^2 \pi}{h2\pi R}$$

$$= \frac{GR}{2h}$$

$$\Delta T_1 = \frac{104 \frac{\text{watt}}{\text{cc}} (0.254 \text{ cm})}{2 \left(0.69 \frac{\text{watts}}{\text{cm}^2 ^{\circ}\text{C}} \right)} = 19.1 ^{\circ}\text{C}$$

$$\Delta T_2 = \frac{104 \frac{\text{watt}}{\text{cc}} (0.475)}{2 \left(0.69 \frac{\text{watt}}{\text{cm}^2 ^{\circ}\text{C}} \right)} = 35.8 ^{\circ}\text{C}$$

$$(T_{\max} - T_{\text{NaK}})_1 = 19.1^\circ\text{C} + 7.4^\circ\text{C} = 26.5^\circ\text{C}$$

$$(T_{\max} - T_{\text{NaK}})_2 = 35.8^\circ\text{C} + 26^\circ\text{C} = 61.8^\circ\text{C}$$

X. Water Film Coefficient

$$\frac{hD}{k} = (0.023) \left[\frac{\rho VD}{\mu} \right]_f^{0.8} \left[\frac{\mu Cp}{k} \right]_f^{0.3}$$

$$h = \frac{(0.023)(0.39)}{0.015} \left[\frac{(0.0151)(26.2)(60.5)(3600)}{0.839} \right]^{0.8} [2.15]^{0.3}$$

$$= 7637 \frac{\text{Btu}}{\text{hr-ft}^2\text{-}^\circ\text{F}}$$

XI. Temperature Difference Across Water Film

$$\Delta T = \frac{q}{hA}$$

$$= \frac{(144)(190,000)}{(7637)(\pi)(2.5)(16)} = 28.5^\circ\text{F}$$

$$\Delta T = 16^\circ\text{C}$$

XII. Maximum Specimen Surface Temperature

A. Temperature at specimen surface

$$T_s = 705 - 44 = 661^\circ\text{C}$$

B. NaK Alloy temperature at specimen surface

$$T_{\text{NaK}} = 661 + 17 = 688^\circ\text{C}$$

DESIGN, DEVELOPMENT AND OPERATION OF
A NaK FILLED NATURAL CONVECTION
CIRCULATING CAPSULE

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DESIGN, DEVELOPMENT AND OPERATION OF A NaK FILLED NATURAL CONVECTION CIRCULATING CAPSULE

The Fast Ceramic Reactor (FCR) program has been undertaken by the Atomic Products Division of the General Electric Company under the sponsorship of the Atomic Energy Commission. The objective of the program is to develop a fast, ceramic-fueled, sodium-cooled reactor.

One series of capsules to be developed under this program will study the potential for fission product plugging as well as fuel and clad performance in a high-power fuel element of reference length ($\sim 20''$ fuel and $12''$ blanket). The irradiation of the first capsule in this series had to be terminated well before the desired burnup was achieved when an abnormally high temperature on the fuel cladding was recorded. Subsequent Radioactive Materials Laboratory (RML) examinations revealed that the fuel cladding had failed. It appeared that the failure was initiated by gas blanketing and burnout of the cladding. The gas blanketing apparently resulted from helium entrapped during the initial NaK loading, which was unable to rise through the annulus due to the small radial dimensions of the annuli. The experience with this capsule called for a basic modification in capsule design to prevent gas blanketing and burnout of the fuel in subsequent high burnup, fission-product-plugging tests having fuel of reference length.

CAPSULE DESIGN

The remainder of this paper will cover the design, development and operation of subsequent capsules in this series. (Fig. 1)

One obvious solution to the problem encountered in the first capsule was to enlarge the radial dimension of the liquid metal annulus surrounding the fuel element. However, enlarging the annulus would amplify the coolant convection problems which had been minimized with the thin-bond annuli. It was decided to capitalize upon, rather than resist, the convection tendency by designing a natural convection-circulating capsule. The basic design criteria, listed in approximate order of decreasing importance, are summarized as follows:

1. Radial dimensions of bond annulus around fuel to be enlarged to reduce gas entrapment, gas blanketing, and fuel burnout problems.
2. Design to minimize gas entrapment and liquid-metal bonding problems during filling operations.
3. Cladding and fuel temperature profiles to approximate those in a large fast ceramic reactor.
4. Provide continuous fission gas pressure monitoring.
5. Provide FCR reference-type continuous spacer tube support of fuel element.
6. Design for ease of post-irradiation disassembly and examination.
7. Provide simple, integral diffusion cold trap to reduce oxide concentration in coolant to reference values.

Sodium, the FCR primary coolant medium, was preferred as a capsule heat transfer medium. However, because of its high melting point (208°F), sodium freezes during capsule shutdown. Freezing is directional and the sodium surrounding the fuel is last to solidify, hence, sodium shrinkage voids (about 3% shrinkage) form around the fuel. These voids furnish potential for fuel burnout unless provisions are made to melt the sodium and remove the shrinkage voids prior to startup. Due to the difficulties involved in providing for sodium melting in the capsule design, the decision was made to use NaK as the coolant since it is molten at room temperature (melting point 10°F).

The fuel elements being investigated in this series of capsules consist of a mixed oxide fuel column (25 w/o PuO₂ and 75 w/o UO₂) topped by a UO₂ blanket. Two lengths of fuel are being used, one to simulate the high power center fuel element in an FCR and the other to simulate the peripheral element. The length of the fuel in the high power element is 20 inches, while in the peripheral element it is 23 inches. The

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FIGURE 1. CONVECTION-CIRCULATING CAPSULE

added length in the peripheral element was necessary to lower the temperature at the fuel blanket interface by having it operate in a lower flux region. In both cases the fuel is clad in 1/4-inch O.D. x .015-inch wall type 347 stainless steel tubing.

One of the design criteria called for the fuel and cladding temperature profiles to approximate those in a large fast ceramic reactor. In order to determine the temperatures accurately, it was necessary to set up differential equations describing the temperature gradient in the axial (x) direction as a function of the heat generated in the fuel, the heat transmitted radially and the specific heat of the NaK (Fig. 2). The equations are:

$$\frac{dT_1}{dx} = \frac{\phi(x)}{WC_p} - \frac{\pi d_1 U_1}{WC_p} (T_1 - T_4)$$

$$\frac{dT_2}{dx} = -\frac{\pi d_2 U_2}{WC_p} (T_2 - T_3)$$

$$\frac{dT_3}{dx} = -\frac{\pi d_3 U_2}{WC_p} (T_2 - T_3) - \frac{\pi d_3 U_3}{WC_p} (T_3 - T_w)$$

$$\frac{dT_4}{dx} = -\frac{\pi d_1 U_1}{WC_p} (T_1 - T_4) + \frac{\pi d_4 U_4}{WC_p} (T_4 - T_w)$$

- where
- T_1 = Bulk NaK temperature inner annulus (fuel region)
 - T_2 = Bulk NaK temperature inner annulus (blanket region)
 - T_3 = Bulk NaK temperature outer annulus (blanket region)
 - T_4 = Bulk NaK temperature outer annulus (fuel region)
 - $\phi(x)$ = Power generation as function of axial distance (x)
 - W = Flow rate
 - d = Diameter
 - U = Overall heat transfer coefficient
 - C_p = Specific heat

These equations are solved simultaneously to obtain the expression defining the temperatures at any point (x). The temperature equations are of the form

$$T_1 = C_1 e^{m_3 x} + C_2 e^{m_4 x} + A_1 \sin \frac{\pi x}{L_1} + B_1 \cos \frac{\pi x}{L_1} + D_1$$

$$T_2 = C_3 e^{m_1 x} + C_4 e^{m_2 x} + T_w$$

$$T_3 = K_{12} C_3 e^{m_1 x} + K_{13} C_4 e^{m_2 x} + T_w$$

$$T_4 = K_{10} C_1 e^{m_3 x} + K_{11} C_2 e^{m_4 x} + A_4 \sin \frac{\pi x}{L_1} + B_4 \cos \frac{\pi x}{L_1} + D_4$$

The value of the constants "C", "M", "A", "B", "D", and "K" are functions of the NaK flow rate. It is necessary, therefore, to determine the steady-state flow before the temperatures defined above are meaningful. The steady-state flow is obtained by first assuming a flow and evaluating the constants in the temperature equations. With the temperatures defined, the density variations around the loop can be determined and, from this, the "driving head" calculated. The "driving head" obtained in this manner must equal the "loss head" for the assumed flow. If it does not, a new flow is assumed and the procedure repeated. The loop bulk NaK and the fuel element temperatures for a 23-inch fuel pin are shown in Fig. 3 and 4.

No attempt is made here to define all of the terms used in these expressions; however, a computer program developed to solve the problem is presented in the appendix. (It should be noted that the constants K_{10} , K_{11} , K_{12} and K_{13} are located in L0, L1, L2 and L3, respectively.)

The chimney (Fig. 1) is insulated in order to develop a high differential in NaK density between the inner and outer NaK annuli due to the temperature differential and, thereby, create substantial circulation. This high thermal resistance in the chimney was obtained through the use of thin-walled concentric tubes with a gas-filled intervening annulus. This annular design permits high thermal resistance without substantial reduction in neutron flux. The chimney was designed in two sections. The conductivity of the chimney in the power-producing lower region of the capsule was raised through the use of helium gas and a stainless steel wire gauze spacer. The increased conductivity of the chimney allows a sizeable portion of the total heat to be removed by radial conduction. The fuel clad temperature profile is strongly dependent upon the ratio of heat leaving by convection to that leaving by radial conduction and so the design of the chimney is very critical. Since little power is produced in the blanket section of the fuel element, it is desirable to minimize radial conduction in the top end of the chimney. This is accomplished by the use of argon in the annulus. Annular spacing in this upper section was maintained through the use of a spiral-wrapped wire between the two tubes.

The outer wall of the capsule consists of two concentric tubes with the gas-filled annular space between them fixed with a wire gauze spacer. This gas annulus contains an adjustable binary gas (helium-argon) which can be changed during operation to regulate the temperature level of the circulating NaK. The relative NaK temperatures within the capsule are determined by power generation rate, the chimney conductivity and the hydraulic resistance to flow. The absolute temperature of the NaK is determined by the conductivity of the outer wall of the capsule and which is controlled by the binary gas.

A diffusion cold trap (Fig. 1) has been incorporated in the lower portion of the capsule. The purpose of the cold trap is to remove any oxides which may be formed during operation. This is to be accomplished by maintaining the cold trap temperature at less than 200°F. Since the

FIGURE 2. CONVECTION-CIRCULATING CAPSULE

FIGURE 3. FCR-B3B PREDICTED LOOP TEMPERATURE

FIGURE 4. FCR-B3B PREDICTED TEMPERATURE VERSUS REFERENCE TEMPERATURE

solubility of Na_2O in NaK at this temperature is less than 50 ppm, it is anticipated that during operation the oxides will diffuse to the cold trap and be deposited there. This will result in relatively high purity NaK coming into contact with the fuel clad.

CAPSULE DEVELOPMENT

Before fabrication of the capsule could begin, the effect of the wire gauze on the conductivity of a gas annulus had to be determined. Heat transfer tests were performed in which a solid stainless steel cylinder was drilled to accept an electrical heater as well as several thermocouples. This cylinder was then wrapped with the stainless steel wire gauze. Finally, the gauze was enclosed by a piece of thin-walled stainless steel tubing with thermocouples attached to its outer surfaces. The ends of the test piece were insulated to minimize end effects. The power input to the electrical heater and the temperature drop across the gas-gauze gap could be measured and from this the effective thermal conductivity determined.

As would be expected, the effect of the wire gauze on the conductivity of the gas gap varied with the composition of the gas in the annulus. The "improvement factor" ($h_{\text{Gas} + \text{Wire Gauze}}/h_{\text{Gas}}$) for argon and helium was approximately 4 and 1.5, respectively. These values are for a .004 inch annulus at approximately 1 inch diameter. It was found that the "improvement factors" stated above could be obtained only if the wire gauze was in good contact with the walls of the annulus. This required that the annulus be sized such that the screen would be in compression when the capsule reached operating temperature.

The complexity of the convection and heat-transfer problems made it desirable to mock up the convection system for out-of-pile tests. A mock up of the system was built and operated with an electrical heater. The results of these tests verified the design. Based upon the analyses and out-of-pile tests, it appeared that the convection-circulating capsule would develop the required temperature profile.

The initial calculations of the steady-state flow and the bulk NaK temperatures were done by hand. These calculations were quite lengthy and any small change in capsule parameters required a complete recalculation. It was decided to use a computer to solve the problem so that parameter changes and operational characteristics would be easily analyzed. The problem was set up for solution on a computer using "BASIC", the elementary algebraic language designed for use with the Dartmouth Time Sharing System. The program that has been developed will solve for and print out the value of the steady-state flow as well as the bulk NaK temperature at five places within the loop for any set of capsule parameters desired.

The computer program developed to solve the natural convection-circulation problem is presented in the appendix.

CAPSULE OPERATION

The capsule is being irradiated at the General Electric Test Reactor (GETR) in a vertically and radially adjustable facility tube (V-RAFT) (Fig. 5). The V-RAFT allows the capsule to be positioned such that the required peak power (21 KW/ft) can be attained and maintained throughout the life of the capsule. In addition, the location of the peak relative to the capsule can be maintained at the desired elevation. Both the vertical and the radial adjustments are necessary to maintain a reasonably constant fuel element temperature profile.

The initial approach to power for this type of capsule is performed as follows:

1. The reactor is brought to power with the capsule backed out to a low flux position.
2. When the reactor power has stabilized, the capsule is adjusted vertically to the proper elevation as determined by the elevation of the peak flux.
3. The capsule is then moved in radially until the required total power, as determined by water calorimetry, is obtained.
4. The required temperatures are obtained by adjustment of the argon-helium gas ratio in the outer annulus.
5. Further vertical adjustment is made as necessary to keep the peak flux at the required elevation relative to the capsule and is based on the reactor rod bank.
6. Further radial adjustment is made as necessary to keep the temperatures within the required limits.

Once the capsule is brought to power and the temperatures adjusted with the binary gas system, water calorimetry will be used only as a check on the power. Future radial adjustments will be made based on the observed temperatures.

The adequacy of the design and development work is demonstrated by the agreement between the predicted and observed temperatures shown in Fig. 6. In this figure the observed temperatures are indicated as points corresponding to the appropriate thermocouple reading while the predicted temperatures are plotted as a continuous line. In both cases, the temperatures are for the capsule operating at rated power.

FIGURE 5. V-RAFT WITH CAPSULE IN POSITION

FIGURE 6. FCR-B3B PREDICTED VERSUS OBSERVED

CONCLUSION

The natural convection circulating capsule concept is practical for the irradiation of fuel elements requiring a liquid metal environment. The use of the convection circulating approach will minimize the problem of gas blanketing and cladding burnout which can be quite troublesome in a thin-bond annuli. In addition, the flowing liquid metal is much more prototypical of most reactor applications than is the stagnant liquid metal version. Since the fuel element temperature can be predicted quite accurately, the use of the natural convection circulation concept has proved to be quite useful in experiments utilizing liquid metals as coolants.

APPENDIX A

NATURAL CIRCULATION LOOP ANALYSIS PROGRAM (UTILIZING "BASIC", THE ELEMENTARY ALGEBRAIC LANGUAGE FOR USE WITH THE DARTMOUTH TIME SHARING SYSTEM)

This program has been developed to solve for and print out the value of 1) the steady-state flow rate (flow), 2) the bulk temperature at the bottom of the fuel column (T inlet), 3) the bulk temperature at the fuel blanket interface (T Top of Fuel), 4) the bulk temperature at the top of the chimney (T exit), 5) the bulk temperature at elevation of the fuel-blanket interface but in the outer annulus (T TOF (outer)) and 6) the bulk temperature at the mid-point of the fuel element (T 1/2). In addition, the value of the steady-state flow equation (F(W)) is printed out to indicate the degree of convergency.

In operation, the capsule parameters are entered as "DATA" in lines 500-531. The order is defined in the "READ" statements, lines 10-45, and the nomenclature given in the "REM" statements, lines 601-639. Any value of flow may be entered (1000 lb/hr being a convenient entry). The program will adjust the flow until the convergency criteria has been met and will then print the temperatures, the steady-state flow and the value of the function F(W). In addition, the final values of the "LET" statements, lines 50-430, will be printed out so that values for various constants may be retrieved for use in calculating the temperatures at any point in the loop.

For the purpose of these calculations, the flux profile is approximated by a sine curve. The peak power occurs at the mid-point of the fuel column and is represented by " $\emptyset$ PEAK". The power at the upper and lower ends of the fuel is represented by " $\emptyset x=0$ ".

```

10 READ W1, W2, W3, W4, W5
15 READ W6, W7, W8, W9, W0
20 READ X1, X2, X3, X4, X5
25 READ X6, X7, X8, X9, X0
30 READ Y1, Y2, Y3, Y4, Y5
35 READ Y6, Y7, Y8, Y9, Y0
40 READ Z1, Z2, Z3, Z4, Z5
45 READ Z6, Z7, Z8, Z9, Z0
49 LET S=0
50 LET A5=W1*X3*X6/X9
55 LET A6=W1*X4*X7/X9
60 LET A7=W1*X5*X8/X9
65 LET A8=W1*Y0*Z1/X9
70 LET K1=W1/X0
75 LET K2=A8/Y9
80 LET K3=A5*A8/(Y9*Y9)
85 LET K4=(A5+A8)/Y9
90 LET K5=(Y2-Z2)/(Y9*X9)
95 LET K6=(Y2-Z2)*A5/(Y9*Y9*X9)
100 LET K7=Z2/(Y9*X9)
105 LET K8=Z2*A5/(Y9*Y9*X9)
110 LET K9=A5/(Y9*Y9*X9)
115 LET M1=Y5/(Y7*Y9)*(A7+SQR(A7*A7+Z3*A6*A7))
120 LET M2=Y5/(Y7*Y9)*(A7-SQR(A7*A7+Z3*A6*A7))
125 LET M3=Y5/(Y7*Y9)*(A8+SQR(A3*A8+Z3*A5*A8))
130 LET M4=Y5/(Y7*Y9)*(A8-SQR(A3*A8+Z3*A5*A8))
135 LET L0=M3*Y9/A5+Y5
140 LET L1=M4*Y9/A5+Y5
145 LET L2=M1*Y9/A6+Y5
150 LET L3=M2*Y9/A6+Y5
155 LET A1=(-K4*K5*(-K1*K1-K3)-K1*K1*K2*K5)/((-K1*K1-K3)2+(K2*K1)2)
160 LET B1=(K1*K5+K2*K1*A1)/(-K1*K1-K3)
165 LET D1=(-K4*K7-K3*Y6)/(-K3)
170 LET A4=(-K6*(-K1*K1-K3))/((-K1*K1-K3)2+(K2*K1)2)
175 LET B4=K2*K1*A4/(-K1*K1-K3)
180 LET D4=(-K3*Y6-K8)/(-K3)
185 LET E0=EXP(M1*X0)
190 LET E1=EXP(M2*X0)
195 LET E2=EXP(M3*X0)
200 LET E3=EXP(M4*X0)
205 LET E4=EXP(M1*Y1)
210 LET E5=EXP(M2*Y1)
215 LET E6=-B1-D1+B4+D4
220 LET E7=B1-D1+Y6

```

```

225 LET E8=B4-D4+Y6
230 LET E9=E3*L2*Z4*E0*(Y5-K9)*E5
235 LET F0=L1*E3*(Y5-L2)*E4*E1*Z4
240 LET F1=L1*E3*E0*Z4*(Y5-L3)*E5
245 LET F2=(Y5-L2)*E4*L3*Z4*E1*E3
250 LET F3=E2*L2*Z4*E0*(Y5-L3)*E5
255 LET F4=L0*E2*(Y5-L2)*E4*E1*Z4
260 LET F5=L0*E2*E0*Z4*(Y5-L3)*E5
265 LET F6=E2*L3*Z4*E1*(Y5-L2)*E4
270 LET F7=E7*L2*Z4*E0*(Y5-L3)*E5
275 LET F8=E8*(Y5-L2)*E4*E1*Z4
280 LET F9=E8*E0*Z4*(Y5-L3)*E5
285 LET G0=E7*L3*Z4*E1*(Y5-L2)*E4
290 LET G1=E9+F0-F1-F2
295 LET G2=F3+F4-F5-F6
300 LET G3=F7+F8-F9-G0
305 LET G4=(Y5-L0)*G1-(Y5-L1)*G2
310 LET C1=Y5/G4*(E6*G1-(Y5-L1)*G3)
315 LET C2=Y5/G4*((Y5-L0)*G3-E6*G2)
320 LET G7=(Y5-L0)*E3*E8
325 LET G8=E2*L1*E3*E6
330 LET G9=L0*E2*(Y5-L1)*E7
335 LET H0=L0*E2*E3*E6
340 LET H1=E2*(Y5-L1)*E8
345 LET H2=(Y5-L0)*L1*E3*E7
350 LET H3=G7+G8+G9-H0-H1-H2
355 LET C3=Y5/G4*(Y5-L3)*E5*H3
360 LET C4=-Y5/G4*(Y5-L2)*E4*H3
365 LET T1=C1+C2+B1+D1
370 LET T2=C1*E2+C2+E3-B1+D1
375 LET T3=C3*E4+C4*E5+Y6
380 LET T4=L2*C3*E0+L3*C4*E1+Y6
385 LET T5=C1*EXP(M3*X0/Y7)+C2*EXP(M4*X0/Y7)+A1+D1
390 LET I1=Z5/(W3*W9*W5*W5)
395 LET I2=W4*X0+I1*Y9*Y9*X0
400 LET I3=C1/M3*(EXP(M3*X0)-Y5)+C2/M4*(EXP(M4*X0)-Y5)+D1*X0+Y7*A1/K1
405 LET P1=I2-Y4*I3
410 LET I4=Z6/(W3*W0*W6*W6)
415 LET I5=W4*(Y1-X0)+I4*Y9*Y9*(Y1-X0)
420 LET I6=C3/M1*(EXP(M1*Y1)-EXP(M1*X0))+C4/M2*(EXP(M2*Y1)-EXP(M2*X0))
425 LET P2=I5-Y4*(I6+Y6*(Y1-X0))
435 LET I7=Z7/(W3*X1*W7*W7)
440 LET I8=W4*(X0-Y1)-I7*Y9*Y9*(X0-Y1)
443 LET R1=L2*C3/M1*(EXP(M1*X0)-EXP(M1*Y1))
444 LET R2=L3*C4/M2*(EXP(M2*X0)-EXP(M2*Y1))
445 LET I9=R1+R2
450 LET P3=I8-Y4*(I9+Y6*(X0-Y1))

```

```

455 LET J0=Z8/(W3*X2*W8*W8)
460 LET J1=(Z4*W4+J0*Y9*Y9)*X0
465 LET J2=L0*C1/M3*(Y5-EXP(M3*X0))+L1*C2/M4*(Y5-EXP(M4*X0))
470 LET P4=J1-Y4*(J2-Y7*A4/K1-D4*X0)
475 LET J3=Y9*Y9/(W3*Z9)*(Y5/(W5*W5)+Y5/(W8*W8))
480 LET J4=P1+P2+P3+P4+J3
481 LET S=S+1
482 PRINT J4,Y9
483 IF S>=2 THEN 492
484 LET S9=Y9
486 LET S4=J4
488 LET Y9=Y9-25
490 TO TO 50
492 LET S3=Y9
494 LET Y9=S9-S4*(Y9-S9)/(J4-S4)
495 IF ABS(Y9-S3) 5 THEN 532
496 LET S4=J4
498 LET S9=S3
499 GO TO 50
500 DATA
505 DATA
510 DATA
515 DATA
520 DATA
525 DATA
530 DATA
531 DATA
532 PRINT
533 PRINT "T INLET=";T1,"T TOP OF FUEL=";T2
534 PRINT "T EXIT=";T3,"T TOF(OUTER)=";T4
535 PRINT "T L/2=";T5
536 PRINT "FLOW=";S3," ", "F(W)=";J4
537 PRINT
538 PRINT
539 PRINT
540 PRINT A5,A6,A7,A8,K1
541 PRINT K2,K3,K4,K5,K6
542 PRINT K7,K8,K9,M1,M2
543 PRINT M3,M4,L0,L1,L2
544 PRINT L3,A1,B1,D1,A4
545 PRINT B4,D4,E0,E1,E2
546 PRINT E3,E4,E5,E6,E7
547 PRINT E8,E9,F0,F1,F2
548 PRINT F3,F4,F5,F6,F7
549 PRINT F8,F9,G0,G1,G2
550 PRINT G3,G4,C1,C2,G7
551 PRINT G8,G9,H0,H1,H2

```

```

552 PRINT H3,C3,C4,T1,T2
553 PRINT T3,T4,T5,I1,I2
554 PRINT I3,P1,I4,I5,I6
555 PRINT P2,I7,I8,F1,R2
556 PRINT I9,P3,J0,J1,J2
557 PRINT P4,J3,J4
600 GO TO 900
601 REM W1=P1
602 REM W2=VM(MEAN KINEMATIC VISCOSITY)FT2/HR
603 REM W3=2G(832 X 106)FT/HR2
604 REM W4=P0(REF. DENSITY AT 0 DEGREES F)LB/FT3
605 REM W5=ARI(CROSS-SECTIONAL AREA)FT2
606 REM W6=AR2 FT2
607 REM W7=AR3 FT2
608 REM W8=AR4 FT2
609 REM W9=DH1(HYDRAULIC DIA.) FT
610 REM W0=DH2 FT
611 REM X1=DH3 FT
612 REM X2=DH4 FT
613 REM X3=D1(DIA.-HEAT TRANSFER) FT
614 REM X4=D2 FT
615 REM X5=D3 FT
616 REM X6=U1(OVERALL H.T. COEFF)BTU/HRFT2 DEGREES F
617 REM X7=U2 BTU/HRFT2 DEGREES F
618 REM X8=U3 BTU/HrFT2 DEGREES F
619 REM X9=CP(SPECIFIC HEAT)BTU/LB DEGREES F
620 REM X0=L1(FUEL LENGTH)FT
621 REM Y1=L(TOTAL LENGTH)FT
622 REM Y2=0PEAK(PEAK POWER)BTU/HRFT
623 REM Y3=64
624 REM Y4=E(CHANGE IN DENSITY)LB/FT3 DEGREES F
625 REM Y5=1
626 REM Y6+TW(WATER TEMP.)DEGREES F
627 REM Y8=3
629 REM Y9=W(FLOW)LB/HR
630 REM Y0=D4(DIA.-HEAT TRANSFER)FT
631 REM Z1=U4(OVERALL H.T. COEFF)BTU/HRFT2 DEGREES F
632 REM Z2=0/X=0 (POWER /X=0)BTU/HRFT
633 REM Z3=4
634 REM Z4=- 1
635 REM Z5=F1/PM1(FRICTION FACTOR/MEAN DENSITY)
636 REM Z6=F2/PM2
637 REM Z7=F3/PM3
638 REM Z8=F4/PM4
639 REM Z9=PM(MEAN DENSITY)LB/FT3
900 END

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THE IRRADIATION OF LIQUID PLUTONIUM FUELS
IN A THERMAL REACTOR*

By

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INTRODUCTION AND SUMMARY

This paper describes an in-pile irradiation program which has been in progress at the Los Alamos Scientific Laboratory (LASL) since January 1964. The purpose of this program is to obtain preliminary irradiation data on the liquid plutonium alloy fuels and tantalum containers that will be used in the Fast Reactor Core Test Facility (FRCTF).^(1,2) Thus, the irradiations are performed under operating conditions and environment simulating those expected in the FRCTF; i.e., fuel-container interface temperatures are maintained above the fuel freezing point of approximately 440°C and below 725°C during all phases of reactor operation and shutdown. The irradiations are accomplished in LASL's 5.0 MW MTR-type reactor, the Omega West Reactor (OWR).⁽³⁾ The technique developed employs a natural convection sodium loop located in the OWR core where the fast and epithermal neutron flux components are highest. A tantalum fuel capsule containing 24 g of Pu-Co-Ce fuel (6.2 g Pu/cc) placed in the loop operates at an average specific power of 150 W/g plutonium. With the present reactor operating schedule, 1% burnup is achieved in about 10 weeks.

* Work done under the auspices of the United States Atomic Energy Commission.

I. IN-PILE TEST ASSEMBLY

A. General

Figure 1 shows the in-pile test assembly which consists of two separate parts; the outer part, the environmental cell, is semipermanently installed in the reactor and contains the convecting sodium loop, thermocouples, heater units, and a gas annulus for heat removal and temperature control. The inner part, an experimental insert, contains the test fuel and additional thermocouples; it is installed and removed as a unit from the environmental cell.

During nuclear operation, sodium is heated by the fuel and heaters and convects upward within an insulating flow divider. It then descends outside the flow divider, transferring heat through a gas annulus to the OWR water coolant stream. The heaters are thermostatically controlled to stabilize temperatures during periods of nuclear operation; during shutdown, they keep the fuel above its melting point of approximately 440°C.

B. Environmental Cell

The environmental cell is a 2-in. o.d., 22-ft long stainless steel assembly containing a 13-in. long convecting sodium loop in its lower section. The convecting sodium is doubly contained and consists of 400 g of hot-trapped sodium. A stainless steel flow divider, containing an argon-filled gas gap, minimizes radial heat conduction between the convective legs of the sodium loop. The top and bottom of the convective sodium channels are rounded to smooth the reversal of the sodium flow.

A heat leak gas annulus is formed by the primary sodium container and the outer tube of the cell. The gap is 0.020-in. thick over the lower nine inches and 0.012-in. thick over the top four inches at operating temperatures. The reduction in gap thickness at the top increases the convective sodium flow. A continuously flowing gas mixture (helium and nitrogen) is admitted to the annulus at the top and exhausted at the bottom via a tube passing through the outer leg of the sodium stream.

Fig. 1. In-Pile Test Assembly

The cell contains seven electrically powered heaters. Six of these are supported on the interior wall of the flow divider and are used for temperature control; the other is wound on the noninsulating upper section of the flow divider and is used only during sodium melting operations. The heater units (Fig. 2) were obtained commercially and are 27-ft long, Inconel-sheathed, MgO-insulated, Nichrome V cored, and have nickel lead wires. The heater section is swaged down to 0.093 in. o.d. over its 50-in. long active length. The six heaters are normally operated at a power of 5.0 kW, which gives a heat flux of 60 W/in.^2 of sheath area. (The manufacturer suggests an operating range of 45-100 W/in.^2 for the heaters if immersed in water or higher conductive media.) Prior to installation, the lower six feet of each heater are given a thorough radiographic and dye penetrant examination to determine sheath and heater integrity. Also, lead and insulation resistances are carefully measured. The 0.250-in. o.d. and 0.188-in. o.d. sections of each heater sheath are sealed to the cell with metal compression-type seals.

A multiple thermocouple unit, consisting of three stainless-steel-sheathed, chromel-alumel thermocouples swaged in an 0.125-in. o.d. stainless steel sheath, is located in the outer convecting sodium stream. The thermocouples are 0.031-in. o.d., MgO-insulated, and are ungrounded. A fourth sheathed thermocouple is located in the heat leak gas exhaust tube; an O-ring fixture at the cell top provides a sliding seal for this thermocouple so that the position of the junction is adjustable; this thermocouple can be used to obtain the axial temperature distribution in the outer convecting sodium stream.

Helium cover gas, at a pressure slightly above atmospheric, is maintained over the convecting sodium and in the upper sealed annular region of the environmental cell.

5.3.5

Fig. 2. Electrical Heater Unit

C. Insert

Figure 3 shows an insert typical of those which have been installed and operated in the cell. The insert is 28 feet long and contains a test fuel element in its lower section. The tantalum fuel element (0.420-in. o.d., 0.020-in. wall, 8-in. long) is filled to a height of about two inches with Pu-Co-Ce fuel (24 g Pu). The closed space above the fuel is filled with helium at atmospheric pressure. The fuel element is placed in an 0.575-in. o.d., 18-in. long tantalum container to provide secondary containment for the fuel. Sodium is used in the resulting annulus to provide a thermal bond. The junctions of eight 30-ft long chromel-alumel thermocouples are located at various axial positions in the annulus and are sealed at the top end of the insert by a multiple-penetration, compression-type fitting.

A Zircaloy tube located above the capsule is used for introducing the bonding sodium into the annulus. This tube is prefilled with hot-trapped sodium and installed in the insert during assembly. After final leak testing and radiography, the insert is evacuated and brought to a temperature of about 140°C to melt the sodium in the Zircaloy tube. During operation, this tube also continuously hot traps the bonding sodium.

To monitor the integrity of the fuel capsule, a helium sweep stream flows down the annulus between the 3/4-in. o.d. tube and the 3/8-in. o.d. tube, passes through the volume above the capsule, and is exhausted via the 3/8-in. o.d. tube through a molecular sieve. If gaseous fission products were released, they would be delayed for several hours in the sieve and detected by a nearby scintillation detector. The helium sweep stream is obtained from a diffusion cell to minimize contamination of the bonding sodium.

Fig. 3. Typical Experimental Insert

The 3/4-in. stainless steel tube is attached to the secondary tantalum container with a threaded joint which is sealed by a silver gasket. This joint provides a separation point during removal of the experimental insert from the environmental cell when the irradiation has been completed.

Two containers which act as gas locks are used in connection with the insert and remain in place at the top of the environmental cell during an irradiation experiment. These maintain a helium atmosphere over the convecting sodium during installation and removal of the insert from the cell and confine any activity which might be released by a fuel capsule. A double O-ring fixture on the top container provides a sliding seal for the 3/4-in. tube. A similar fixture seals the 3/8-in. o.d. tube to the 3/4-in. tube.

D. Control Circuitry and Instrumentation

The six heater units are electrically separated into two groups of three, each having identical control circuits. Figure 4 is a simplified schematic of the control circuit for one group. A variable autotransformer supplies power to the heaters from a 208 V source. The input to the transformer is controlled by mercury relays which are actuated by temperature control instruments. A wattmeter indicates the power drawn by the heaters. Each heater is separately fused and is fed by its own isolation transformer to prevent arcing in case of a heater lead-to-sheath short. Indicator lamps are incorporated between the heater leads and ground to give visual warning of insulation resistance deterioration.

5.3.9

Fig. 4. Simplified Heater Control Circuit

Six of the thermocouples are connected to recording-only equipment; the remaining six thermocouples are connected to indicating- and recording-type control equipment. High and low temperature alarms obtained from these instruments are fed into the main reactor alarm system. If the temperatures exceed the control set-point, a 2-out-of-3-type network will scram the reactor.

Other annunciator circuits indicate the existence of high or low flow in the sweep gas and heat leak gas flowmeters, high or low radiation level from the sweep gas activity monitor, and loss of instrument power.

II. DESIGN AND PERFORMANCE

A. Thermal

This section of the paper discusses the thermal design and performance characteristics of the convecting sodium loop which has just been described and the neutronic characteristics of OWR fuel irradiation experiments.

Some desirable features of this design (Fig. 1) are as follows:

1. High heat fluxes, and resulting high temperature gradients, are restricted to the immediate neighborhood of the fuel. Consequently, heat fluxes across the gas gap are low, permitting larger gaps and giving a low film drop to the reactor coolant stream.
2. The heaters need not be confined to a small space, nor is thermal conduction through them a factor. Hence, commercially available heaters may be used, and they may be operated conservatively.
3. A wide range of control over temperature and power operating characteristics is available at any time through variation in the composition of the gas mixture flowing in the heat leak annulus.

During reactor operating periods, the heat source (fuel) is near the bottom of the rising leg of the convection loop, while the major heat sink (thinner gas annulus) is at the top of the descending leg. These locations favor a high convective flow, which is desired to minimize the vertical temperature gradient in the fuel sample. During periods of reactor shutdown, the heat source (electrical heaters) is not as well localized, so the convective flow is reduced.

The calculated temperatures around the convecting loop are shown in Fig. 5 for an assumed $\frac{5}{1}$ kW fission or electrical source. The corresponding turbulent convective flows are 0.90 gal/min for fission heat and 0.66 gal/min for electrical heat. Measured temperatures and convective flow, based on power and temperature measurements, are consistent with these calculations.

Fig. 5. Temperatures in Convecting Sodium Loop

The vertical temperature rise along the fuel during the above nuclear operation is about 80°C, and during electrical operation it is much less (about 30°C). The 80°C figure is comparable to that expected for the liquid plutonium fuel elements of proposed reactor designs, so these designs are well simulated in the present test samples. This temperature rise would increase, less than linearly, with test sample power; at a power several times present levels, the temperature at the top of the fuel would become excessive for long-term operation.

The effect of varying the composition of the He + N₂ gas mixture flowing through the heat leak annulus is shown in Fig. 6. No dependence on rate of flow was observed for pure gases or for a mixture over the range tested, 0 to 40 cc/min. The thermocouple indicating approximately the mean temperature in the descending (outer) leg was held at 540°C during these measurements. Early design calculations for pure N₂ and for pure He at a mean temperature of 487°C are indicated. A power range of about 3 to 12 kW can be spanned.

During operation, the gas mixture is adjusted to give approximately the desired conditions, after which fine control is maintained by the thermostatically controlled electrical heaters. The only automatic change in gas mixture is the rapid introduction of N₂ under certain emergency conditions to bring power requirements within the range of the electrical heaters (5 kW).

At full reactor power, the radiation heating produced in the loop containing a fuel capsule with 24 g plutonium is 5.6 kW -- 2.0 kW gamma heat and 3.6 kW fission heat. Gamma heat (1 W/g) was measured by placing a stainless steel dummy capsule in the sodium loop and observing the difference in electrical power required to maintain a constant average temperature drop across the heat leak annulus with the reactor at zero power and at 5.0 MW. The fission heating was measured in a similar manner. The value obtained for the fission heating by this method agrees within 15% with the values obtained by radiochemical analysis.

Fig. 6. Effect of Gas Mixture on Power

B. Neutronic

Before starting detailed design and construction of the irradiation experiment, preliminary neutronic calculations indicated that useful data could be expected from such irradiations. The calculations were made using a simplified one-dimensional cylinder model of the OWR reactor with a cylinder of fuel along the axis. Material compositions were realistically specified in 65 mesh intervals which were closely spaced within the fuel and in its neighborhood. The DTK code used was of the Carlson⁽⁴⁾ type, based on transport theory in S_4 approximation. The Hansen-Roach⁽⁵⁾ 16-group cross section library was used.

Flux distributions for the present design, containing fuel of 5.7 g Pu/cc, is indicated in Fig. 7. Here the thermal flux and the fast flux (combined three top groups, energy over 0.9 MeV) are shown. The thermal flux is depressed by about a factor of five at the fuel surface.

The corresponding power distribution is indicated in the upper curve of Fig. 8. The lower curve is for a dense fuel sample surrounded by a thick cadmium neutron filter. This calculation was compared with experimental results obtained in an early (1960) experiment in OWR, using LAMPRE I fuel (Pu-Fe eutectic, 16 g Pu/cc). This comparison was used to obtain a normalizing factor for use in this and other calculations. The two curves drawn represent the normalized power distributions, while the points shown were obtained by radiochemical analysis of solid specimens of fuel which were irradiated for short times at low power. Samples for the latter were taken at various radii by machining on a lathe. The horizontal lines on the points indicate the radial extent of the samples where it is significant. The experimentally determined edge-to-average ratio for the 6.2 g Pu/cc fuel was 5.1, while the calculated value for the 5.7 g Pu/cc fuel, the nearest calculation available, was 5.7. Respective average specific powers were 143 and 158 W/g Pu. The lower curve was normalized to the experimental value of 35 W/g Pu. The edge-to-average ratios were 1.4 (calculated) and 1.5 (experimental).

Fig. 7. Flux for 5.7 g Pu/cc Fuel in Present Design

Fig. 8. Radial Power Distributions

A calculational study was made of the effect of varying the thickness of a natural boron filter for the case of 3.0 g Pu/cc fuel. Results (Fig. 9) indicate that a serious loss in average power is involved for even a moderate improvement in edge-to-average ratio.

The edge-to-average ratio without filtering is a function of the plutonium content of the fuel, as shown in Fig. 10. Analysis of the related fuel temperature distributions indicated that, for liquid fuels in the range from 3 to 8 g Pu/cc fuel, the power distribution of unfiltered test samples was satisfactory. Comparison was made to samples with the same average power density but of uniform radial power generation. Both central temperature rise and the force inducing fuel convection were found to be about 40% (for 8 g Pu/cc fuel) to 50% (3 g Pu/cc fuel) of those for the uniform distribution. Hence, it was decided to omit neutron filters in the design. Actually, the high power density at the tantalum wall may be considered advantageous in overtesting certain interface effects such as fission product recoil damage.

The average power density of Fig. 10 increases rather slowly with the plutonium content of the fuel. For the denser fuels, it appears to consist of a constant part, about 1000 W/cc, plus a linear part having a slope of about 42 W/g Pu. These are probably attributable to nonpenetrating and to penetrating components of the flux, respectively.

Figures 9 and 10 are for a preliminary design of the irradiation experiment. The final design, used for Figs. 7 and 8, contains more tantalum and gives lower power densities -- presumably at the expense, primarily, of the nonpenetrating component of flux.

Fig. 9. Effect of Natural Boron Neutron Filters

Fig. 10. Effect of Plutonium Content of Fuel

III. LOOP OPERATION AND HANDLING

A. General

There are two environmental cells installed in the OWR at the present time. The first cell was installed in March 1965, and the second in December 1965. Each cell has been operated with a variety of inserts as follows:

1. A nonfueled insert containing a stainless steel capsule to measure gamma heating,
2. A solid-fueled insert to measure the fission power distribution,
3. A liquid-fueled insert for system checkout.

Other irradiations included a fourteen-week irradiation to achieve 1% burnup. Presently one cell contains a fuel element which has achieved 2% burnup to date, and the other cell contains a fuel element which has achieved 1% burnup.

The electrical and gas control systems have been designed so that the loop operation requires very little manual attention during all phases of reactor operation and shutdown. Daily readings of the system parameters are taken and weekly checks of the control and alarm circuitry are performed.

B. Operation

When the reactor is shut down, nitrogen is kept in the gas annulus and about 3.0 kW of electrical heat is required to maintain the system at operating temperature. When the reactor is at full power and the cell is loaded with 24 g Pu of the 6.2 g Pu/cc alloy, about 5.6 kW of radiation heating (2.0 gamma heat + 3.6 fission heat) is produced in the loop. An additional 2.5 kW of electrical heat is used to stabilize the system temperature. A gas mixture consisting of helium and nitrogen ($\approx 75\%$ He) is required to remove this 8.0 kW of heat. A gas system automatically changes the heat leak mixture during reactor startup and shutdown (Fig. 11). During shutdown, all solenoid valves are de-energized and the preset gas mixture is vented through solenoid valve V-1. As the reactor power is increased, less electrical heat is required to maintain temperature. When reactor power is 3.0 MW, 3.4 kW

Fig. 11. Heat Leak Gas Control System

of radiation heat is produced in the loop and no electrical power is required. At this point, the solenoid valves are manually energized, and the preset gas mixture begins to flow through the heat leak annulus, causing a 5.0 kW electrical heat demand. Also, the nitrogen dump line is filled to a pressure of 20 psig. As the reactor power is raised to 5.0 MW, less electrical heat is required until at full reactor power, about 2.5 kW of electrical heat is used for thermostatic control. When the reactor power drops below 3.0 MW, the solenoid valves are automatically de-energized, resulting in the gas mixture being vented and nitrogen from the dump line sweeping out the helium-nitrogen mixture from the annulus. The nitrogen enters the annulus in about three seconds. Valve V-5 is closed when de-energized to prevent continual nitrogen flow during periods of shutdown.

The operation of the control system has proven to be entirely satisfactory and reliable. The heater control circuitry holds the system to within $\pm 5^{\circ}\text{C}$ during nuclear operation at power or while the reactor is shut down. There are twelve mercury plunger-type relays in the two systems; three of these have stuck in the off position and have had to be replaced. This is not unusual, considering that some of them switch 15 A currents ≈ 20 times per minute. The heater controls are divided into two independent groups as mentioned earlier. When each of the above failures occurred, the remaining group continued to operate, and there was little change in temperature.

There are fourteen heaters in the two systems. Some of these have operated at temperature for over a year, and all are still working. None has shown any significant change in heater resistance or heater-to-sheath insulation resistance. There are four thermocouples in each environmental cell, and each of the inserts has had up to eight thermocouples. Of the total of about 65 thermocouples used, all worked without trouble, except one which failed before installation of the insert into the environmental cell.

The nuclear reaction, $\text{Na}^{23}(\text{n},\text{p})\text{Ne}^{23}$ produces significant amounts of Ne^{23} in the system. During normal operation, the gas flow rate is adjusted so that the Ne^{23} , which has a half-life of 38 sec, decays to an undetectable level before reaching the monitor at the top of the reactor. However, this reaction permits a check of the sweep gas monitoring system. To make such a check, one increases the sweep gas flow. A response on the radiation monitor indicates proper operation. A standard source can also be used to check and calibrate the system.

C. Handling

The in-pile test assembly, as previously described, consists of two parts; a semipermanent environmental cell and a replaceable insert containing the test capsule.

To install the insert, the gas-tight container assembly (Fig. 3) is attached to the top of the cell with an O-ring seal. The containers are pumped out, filled with helium, and opened to the cell. The insert is then lowered into the cell through the sliding seal located at the top of the container assembly (Fig. 1). The convecting sodium temperature is held at approximately 250°C during these operations. After the insert has been satisfactorily installed, the system temperature is increased to melt the fuel.

A minimum of two days cooling time is usually allowed before an insert is removed from the cell. There have been up to 2500 curies of gamma activity in inserts that have been removed. Prior to insert removal, two split shields of depleted uranium are assembled around the gas-tight containers at the top of the cell. These are shown in Fig. 12. The lower shield is nine inches outer diameter and four inches thick; the upper shield is five inches outer diameter and two inches thick. With the sodium liquid, the removal process

Figure 12 Shields for Insert Removal

is started by raising the 3/8-in. o.d. inner tube two feet, pulling it through the upper seal (Fig. 3). This removes the thermocouples from the region of the fuel element. The insert is then raised until the tantalum secondary container is positioned in the lower container. In this position, the hexagonal top of the tantalum engages a mating socket in the container. Valves at the top of the cell and the bottom of the lower container are then closed. Next, the steel tubing is unscrewed from the tantalum secondary container and raised into the top shield. The valves between the two containers are then closed. The unactivated part of the steel tubing is now cut away, leaving the radioactive parts in two shielded, gas-tight containers which are separated and removed. Plastic bags would be used to seal the ends of the gas-tight containers, if the sweep gas indicated the presence of activity. No irradiated environmental cells have been removed from the reactor thus far. However, a uranium shield and other necessary components exist for this eventuality.

Figure 13 shows the radiation fields which can be expected through four inches of uranium from the fission product-tantalum source. Somewhat higher fields than this were actually experienced due to scattering from beams coming from the unshielded ends of the large gamma shield.

Fig. 13. Radiation Fields from Fission Products & Tantalum

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VI. SPECTRAL CONTROL AND DOSIMETRY TECHNIQUES

**A COMPARISON OF FUEL BURNUP
DETERMINATIONS**

by

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INTRODUCTION

Two irradiation programs have been selected for comparison of burnup determinations. In both programs, a total of 22 specimens were available with burnups ranging from 16,000 Mwd/MT fuel to 114,000 Mwd/MT fuel. The earlier program involved the irradiation of (Pu+U)C fuel samples irradiated in the GETR during the period of November 1962 through September 1964. The second program utilized the MTR to irradiate UO₂ cermet fuels during the period of September 1963 through October 1964. All capsules functioned satisfactorily and remained in the reactor until design burnup was achieved.

DESCRIPTION OF IRRADIATION PROGRAM

The earlier program consisted of 12 specimens in six capsules which were irradiated in the GETR; five of the capsules were in pool positions and one in an in-core location. Peak burnup achieved was 114,000 Mwd/MT fuel. The specimens were 0.190-in. diameter clad pellets of (Pu_{0.2}U_{0.8})C. The second program consisted of 10 specimens (0.250-in. diameter) in five capsules which were irradiated in the MTR to burnups of 38,000 Mwd/MT fuel. Of the 10 specimens, eight were columbium UO₂ cermets with 20% U²³⁵ enrichment. Two of the specimens, in Capsule 71, were UO₂ pellets.

METHODS OF BURNUP DETERMINATIONS

Two methods were used to estimate burnup during the irradiation of the capsules in the GETR. A calorimetric method, using a heat meter built into the capsules, from which burnup was calculated, and burnup based upon an estimate of neutron flux at the specimens. The flux estimate was made by the test reactor physics personnel. At the end of the irradiation, two additional burnup determinations were made, isotopic analysis of the fuel samples and an estimate based upon the integrated flux dose received by the flux monitor wires.

The burnup of the second set of specimens irradiated in the MTR was determined by two methods: the capsule calorimetry method during irradiation and isotopic analysis of the fuel at the end of irradiation.

A brief description of the methods of burnup determination follows.

Calorimetric Method

Curves of total heat generation rate vs Δt from reactor water coolant to each fin root thermocouple are prepared from startup data. Fig. 1 is a cross section of a typical capsule and shows the thermocouple locations. The data for these calibration curves are obtained by increasing reactor power a step at a time. At each step, a known amount of electrical heat is added and the temperature changes are observed. This permits determination of the fission heat generation rate at each step and makes it possible to construct total heat generation rate (neutron + gamma) curves vs temperature covering the range from zero to full reactor power.

A digital computer program is used to determine the two-dimensional axisymmetric temperature distribution in the capsules. The results are used to calculate correction factors for the heat meter. The program permits variation of thermal properties and different heat generation rates in the capsule components.

Fig. 1 — Irradiation Capsule Cross Section

Several simulation problems for each capsule design show that the heat flux lines do not have the same shape for electrical heating and fission heating in the specimen. Consequently, the temperature drop across the heat meter is different for the same linear heat generation rate in the electrical heaters and in the specimens. Generally, the temperature drop is larger for electrical heat than for fission heat. The computer results are used to calculate correction factors for the heat meter. With the correction factors and the calibration curve obtained for the heat meter during reactor startup, it is possible to determine specimen power during the irradiation cycle.

A computer code for data reduction has been prepared and the appropriate calibration data are fed in at the beginning of each cycle. During that cycle, temperature data are used to calculate total heat generation rate at all the fin root thermocouple locations. Gamma heat is subtracted and burnup is calculated from daily heat generation values. When more than one burnup value is available per specimen, an average value is used.

Burnup Determination by Reactor Operator

Estimates of specimen burnup were supplied by GETR physics personnel for each irradiation cycle. The burnup estimates were based upon calculated neutron flux values at the specimen locations. The following procedure was followed for each cycle of irradiation for all capsules in the GETR. A two-dimensional diffusion theory code on the entire reactor was run to determine the unperturbed flux at the capsule locations. A one-dimensional transport theory code was run to determine the ratio of average flux in the fuel sample to the unperturbed flux. The fuel content for each calculation is adjusted to account for the fuel depletion during the previous cycles of operation.

Radiochemical Analysis of Monitor Wires

Thermal neutron dosimetry data were obtained by radiochemical analyses of nickel wires containing 0.094 w/o cobalt. Analyses were performed on 1-in. sections which were adjacent to the respective specimens.

An average specimen neutron flux value was then calculated from the value of the flux wires. The distance from the monitor wire to the fuel sample was known and the neutron flux attenuation factor was calculated. Specimen burnup was then calculated using the value of thermal neutron flux at the specimen location.

Isotopic Analysis

Isotopic analysis was performed on each fuel specimen to determine burnup. The samples were taken at the midplane of each specimen in the form of transverse cross-sectional slabs. Uranium burnups were calculated both on the basis of depletion of U^{235} and formation of U^{236} and on the change in U^{236} to U^{235} ratio. Plutonium burnup was calculated both on the basis of Pu^{239} disappearance and on net changes in Pu^{240} , Pu^{241} , and Pu^{242} .

COMPARISON OF DATA

Tables 1 and 2 list the burnup data for the two programs. Fission data have been converted using 200 Mev/fission, and the values in all cases are midplane burnups for each specimen. The isotopic values were used as the standard and all values compared to those. The best agreement with isotopic values are those based upon the GE flux estimates. Capsule 65, containing specimens 65T and 65B, was located in an in-core position, and at the time of irradiation GE personnel indicated that the in-core flux values were not as well known as the pool positions. The in-pile calorimetric method shows the next best agreement with isotopic analysis. Agreement is better on the average for the lower burnup spec-

TABLE 1 — COMPARISON OF BURNUP VALUES FOR $(\text{Pu}_{0.2}\text{U}_{0.8})\text{C}$ FUEL SPECIMENS IRRADIATED IN GE TRFissions per cm^3 — (Mwd/MT Fuel)

Specimen	Isotopic Analysis	Flux Wire Measurement	% Variation from Isotopic	Calorimetric Measurement	% Variation from Isotopic	GE Flux Measurement	% Variation from Isotopic
61T	7.25×10^{20} (21,700)	10.7×10^{20} (32,050)	-47.7	8.97×10^{20} (26,850)	-23.7	7.91×10^{20} (23,700)	- 9.2
61B	7.69×10^{20} (24,200)	9.38×10^{20} (29,450)	-21.7	7.32×10^{20} (23,000)	+ 5.0	7.90×10^{20} (24,800)	- 2.5
64T	7.47×10^{20} (23,400)	7.38×10^{20} (23,150)	+ 1.1	6.12×10^{20} (19,200)	+17.9	5.45×10^{20} (17,100)	+26.9
64B	7.60×10^{20} (22,800)	8.15×10^{20} (24,400)	- 7.0	5.88×10^{20} (17,600)	+22.8	5.41×10^{20} (16,200)	+28.9
62T	10.4×10^{20} (31,300)	15.8×10^{20} (47,300)	-51.1	14.2×10^{20} (42,500)	-35.8	11.2×10^{20} (33,500)	- 7.0
62B	11.2×10^{20} (35,100)	14.4×10^{20} (45,100)	-28.5	10.7×10^{20} (33,500)	+ 4.6	11.15×10^{20} (35,000)	+ 0.3
65T	12.6×10^{20} (37,700)	16.6×10^{20} (49,800)	-32.1	11.9×10^{20} (35,600)	+ 5.6	17.2×10^{20} (51,500)	-36.6
65B	15.7×10^{20} (49,300)	16.1×10^{20} (50,700)	- 2.8	12.05×10^{20} (37,900)	+23.1	18.4×10^{20} (57,700)	-17.0
63T	37.6×10^{20} (112,700)	38.4×10^{20} (115,000)	- 2.0	30.8×10^{20} (92,100)	+18.3	36.8×10^{20} (110,000)	+ 2.4
63B	36.3×10^{20} (114,000)	38.4×10^{20} (115,000)	- 1.0	24.2×10^{20} (76,000)	+33.3	36.8×10^{20} (110,000)	+ 3.5
66T	33.7×10^{20} (101,000)	38.9×10^{20} (116,500)	-15.3	23.2×10^{20} (69,500)	+31.2	30.4×10^{20} (91,000)	+ 9.9
66B	32.0×10^{20} (100,500)	—*	—	23.4×10^{20} (73,400)	+26.9	30.4×10^{20} (91,000)	+ 9.5

*Flux wire not recovered.

TABLE 2 — COMPARISON OF BURNUP VALUES FOR UO₂
CERMET FUEL SPECIMENS IRRADIATED IN MTR

<u>Capsule</u>	<u>Specimen</u>	Calorimetric Avg. Burnup, <u>Mwd/MT</u>	Isotopic Avg. Burnup, <u>Mwd/MT</u>	<u>% Variation from Isotopic</u>
67*	80 v/o	22,750	29,000	+22
67*	70	25,600	32,000	+20
68*	80	12,770	16,026	+20
68*	70	15,130	16,053	+ 6
69	80	22,990	22,400	- 3
69	70	24,890	28,850	+14
70	80	34,760	34,100	- 2
70	70	35,100	38,500	+ 9
71	Top	37,500	38,300	+ 2
71	Bottom	35,800	35,500	- 1

*Burnup calculated from only three power determinations per cycle. All others based upon daily power determinations.

imens. The numbers indicate that additional work is required on the higher burnups to 100,000 Mwd/MT. The computerization of data reduction did not occur until after the majority of the irradiations had been completed. At first, in computing the burnup for each cycle, three values of heat generation rate were used: the beginning rate, the end rate, and an average rate. This same method was also used for the first two capsules, 67 and 68, in Table 2. With the irradiation of Capsule 69, data reduction was carried out on the computer and burnup was calculated using heat generation rates determined daily. An appreciable improvement for Capsules 69 through 71 over previous capsules is seen in the agreement between isotopic analysis and the calorimetric method.

The determination of a heat generation rate depends upon a thermocouple indication. The second program, irradiated in the MTR, included improvements in thermocouple manufacture and placement and the malfunction rate was much

lower in this program than in the earlier program. The effect of improved thermocouple reliability on the burnup determination has not been evaluated.

CONCLUSIONS

Based upon the data available, the calorimetric method of determining burnup during irradiation is a workable one and shows good agreement with isotopic results to burnups of 40,000 Mwd/MT fuel. Based upon the improvement shown from the first to the second program, it seems reasonable to say that the calorimetric method would be equally good at higher burnups.

The data also indicate that reactor operators can predict burnup in good agreement with isotopic analysis, performed at the end of irradiation.

The variation in flux wire dosimetry suggests that the money spent for wire analysis might be better allocated to improvement of one of the other methods of burnup determination.

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EFFECT OF REACTOR CORE CHANGES ON NEUTRON SPECTRA^(a)

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EFFECT OF REACTOR CORE CHANGES ON NEUTRON SPECTRA

INTRODUCTION

Calculations of neutron spectra⁽¹⁾ are made for reactor positions in which graphite irradiation programs are conducted. These spectra are required to determine:

Effective activation rates for flux monitor materials placed in the irradiation capsules.

Differences in flux between the monitors and actual samples being irradiated.

Correlations of radiation data obtained in different irradiation facilities in various reactors.

A study was made to evaluate the effect on the neutron spectra, flux intensity, and carbon damage index caused by changes in material compositions in the radiation test positions or adjacent facilities.

DISCUSSION

Several core loadings of the General Electric Test Reactor (GETR) were analyzed. The GETR (Figure 1) is a light-water-cooled and moderated reactor. The 36 in. long reactor core is contained inside a 24 inch ID pressure vessel. The inner capsule positions D7, E7, and F7 have been used for irradiations of graphite and are the positions of interest. These positions contain a 3 in. by 3 in. filler piece with an axial 2.2 in. diameter hole accommodating a 2.030 in. diameter irradiation capsule.

A two dimensional transport theory physics code, 2DXY, was used to calculate the neutron spectra. The geometry for input to the calculations was a

21 by 21 matrix of one-inch squares at the mid-plane cross section of the reactor. Thus, in the 3 in. by 3 in. irradiation positions nine separate solutions of spectral detail, flux intensity, effective nickel and iron cross sections, and damage index were calculated. Of these nine solutions the one for the central area containing the graphite samples was of primary interest. Changes caused by the various core loadings discussed below are those occurring in this central area.

The nine core loading arrangements studied were based on various loadings used or proposed for use in the GETR in the vicinity of the D7, E7, and F7 positions used for irradiation of graphite. The core arrangement considered as the basic one is shown in Figure 2. In this basic core arrangement there are experimental capsules in C6, C8, G6, and G8; fuel in D6, E6, F6, D8, and F8; and control rod fuel in C7 and G7. In D7 and F7 the graphite capsules are positioned in beryllium filler pieces and in E7 in an aluminum filler piece.

The variations to the basic core arrangement are as follows:

1. Replacement of the aluminum filler piece in E7 with an aluminum guide tube (a 2-1/2 in. OD schedule 80 aluminum pipe).
2. Replacement of the control fuel with the control rod poison in C7 and G7 (the poison is 1 wt.% boron in stainless steel).
3. Replacement of the control fuel with 75% fuel and 25% poison in C7 and G7.
4. Replacement of the graphite capsule in E7 by a cobalt capsule in a beryllium filler piece (the cobalt capsule is a 1-1/8 in. diameter non-instrumented capsule).
5. Replacement of the graphite capsule in E7 by the same cobalt capsule in an aluminum filler piece.
6. Replacement of the fuel in E6 with the same cobalt capsule in a beryllium filler piece.
7. Replacement of the fuel in E6 with the same cobalt capsule in a beryllium

filler piece plus the replacement of the control fuel in C7 and G7 with the control rod poison.

8. Replacement of the graphite capsules in D7 and F7 with cobalt capsules in beryllium filler pieces.

For each of these variations the change in neutron spectra over the range of energies from 0.18 to 10 MeV were compared to the basic core arrangement. Since the results for the symmetrical positions D7 and F7 were nearly identical, only those for D7 are considered here. A typical result is shown in Figure 3 for variation 4. As would be expected the spectra at the center of E7 changed quite drastically since the graphite capsule from the basic arrangement was replaced by a cobalt capsule. However, the neutron spectra on the graphite samples in the center of D7 showed very little change and the flux intensity decreased slightly.

A shift in neutron spectra is shown on Figure 4 which presents the spectra for variation 6. The spectra in both D7 and E7 are softened when the fuel in E6 is replaced with a cobalt capsule. At the same time the fast flux intensity increases. Shifts in spectra in the graphite samples were also noted for variations 1, 2, 3, 7, and 8.

In the eight variations the flux intensity in D7 and E7 was compared with the flux intensity calculated for the basic core arrangement. Both the absolute fluxes and the ratio of the flux to that for the basic core are presented in Table I. Thermal fluxes in the graphite varied from -28 to +8% of those for the basic core while fast fluxes, $E > 0.18$ MeV, varied from -14 to +20%. For E7 the largest effects on fast flux are for changes in fuel to control rod poison (variation 2) and for material changes within the position (variations 1, 4 and 5). For D7 the largest effects are for changes in fuel to poison (variation 3) and fuel to cobalt in E6 (variation 6).

Corresponding calculations for the effective nickel and iron cross sections showed that they changed from -11 to +7% for these core variations.

To determine if these calculated changes in spectra, flux intensity, and effective cross sections were of serious consequence in analyzing irradiation damage to the graphite, calculations of a carbon damage index⁽²⁾ were also made for each core variation. The damage index (K) in displacements per cm³ is:

$$K = \frac{\int_0^{\infty} \Sigma_s(E) N(E_n) \phi(E) dE}{\int_{.18 \text{ MeV}}^{\infty} \phi(E) dE}$$

where $\Sigma_s(E)$ is the group averaged elastic scattering cross section for carbon.

$N(E_n)$ is the number of displacements per neutron collision as a function of neutron energy.

$\phi(E)$ is the group averaged flux.

The displacement model used for $N(E_n)$ is the deHalas model.⁽³⁾

The damage index in displacements per cm³ and the damage per unit flux are presented in Table II. The calculated values as well as the ratios to the basic core loading are given. The gross displacements vary from -14 to +19% of the basic core, which is nearly the same as the fast flux changes of -14 to +20% (Table I). However, the displacements per unit flux vary from only -0.8 to +0.9% of the basic core. This indicates that even with variations in spectra and intensity, fluxes based on neutrons having energies greater than 0.18 MeV will accurately indicate the displacements in graphite and that it is possible to correlate damage in differing reactor loadings if neutron spectra are calculated.

TABLE I
FLUX INTENSITIES

Core Arrangement	Thermal Flux ϕ_{th} ($nv \times 10^{-14}$)		Ratio of Thermal Fluxes*		Fast Flux $\phi(E > 0.18 \text{ MeV})$ ($nv \times 10^{-14}$)		Ratio of Fast Fluxes*	
	E7	D7	E7	D7	E7	D7	E7	D7
Basic	1.632	1.767	1.000	1.000	1.485	1.231	1.000	1.000
1	1.741	1.276	1.067	0.722	1.677	1.238	1.129	1.006
2	1.585	1.269	0.971	0.718	1.779	1.259	1.198	1.023
3	1.390	1.173	0.851	0.664	1.456	1.062	0.980	0.863
4	1.166 ⁽¹⁾	1.778	0.715 ⁽¹⁾	1.006	1.332	1.206	0.897 ⁽¹⁾	0.980
5	0.946 ⁽¹⁾	1.650	0.579 ⁽¹⁾	0.934	1.620 ⁽¹⁾	1.266	1.091 ⁽¹⁾	1.028
6	1.606	1.266	0.984	0.716	1.596	1.095	1.075	0.889
7	1.763	1.852	1.080	1.048	1.361	1.193	0.916	0.969
8	1.358	0.918 ⁽¹⁾	0.832	0.520 ⁽¹⁾	1.438	1.168 ⁽¹⁾	0.968	0.949 ⁽¹⁾

* Compared to the basic core loading.

⁽¹⁾ Cobalt capsule.

DAMAGE INDEX

Arrangement	Displacements Production				Damage Index			
	$D_c = \int_0^{\infty} \Sigma_s(E) N(E_n) \phi(E) dE$				$D_c / \phi(E), (E > 0.18 \text{ MeV})$			
	E7		D7		E7		D7	
	(1)	(3)	(1)	(3)	(2)	(3)	(2)	(3)
Basic	4463.5	1.000	3737.5	1.000	3005	1.000	3035	1.000
1	5008.7	1.120	3726.2	0.997	2986	0.994	3010	0.992
2	5305.4	1.187	3796.8	1.016	2982	0.992	3015	0.994
3	4359.0	0.975	3217.7	0.861	2995	0.997	3031	0.999
4	4059.5 ⁽⁴⁾	0.908 ⁽⁴⁾	3678.2	0.984	3047 ⁽⁴⁾	1.014 ⁽⁴⁾	3048	1.004
5	4853.4 ⁽⁴⁾	1.086 ⁽⁴⁾	3839.2	0.973	2995 ⁽⁴⁾	0.997 ⁽⁴⁾	3034	1.000
6	4782.7	1.070	3333.5	0.892	2997	0.997	3044	1.003
7	4124.5	0.922	3638.6	0.973	3032	1.009	3050	1.005
8	4333.6	0.969	3556.7 ⁽⁴⁾	0.951 ⁽⁴⁾	3014	1.003	3046 ⁽⁴⁾	1.004 ⁽⁴⁾

- (1) Displacements $\text{cm}^{-3} \text{sec}^{-1} \cdot (x 10^{14})$
 (2) Displacements cm^{-1}
 (3) Ratio to Basic Core Loading
 (4) Cobalt Capsule

SUMMARY

Several variations of the loading of the GETR core in positions adjacent to the D7, E7, and F7 positions in which graphite capsules are irradiated were analyzed. In general, changes in materials within the irradiation position perturb the neutron spectra, change the fast flux intensity, effect the effective cross sections for flux monitoring materials, and change the number of displacements occurring. However, changes made in the adjacent positions produce smaller perturbations, with the exception of drastic changes, such as replacing control fuel with poison. The results point out that with knowledge of the neutron spectra and effective cross sections that the flux, $E > 0.18$ MeV, provides a very accurate indication of the damage to the graphite even when large variations in reactor loadings occur. Of even greater significance as far as irradiation testing is concerned, the results show that once the detailed neutron spectra and effective cross sections are determined for an experiment they will continue to be valid even though large variations in core loadings are made in adjacent reactor positions.

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RADIAL LOCATIONS OF GETR EXPERIMENTAL FACILITIES

FIGURE 2
BASIC CORE ARRANGEMENT

FIGURE 3
REPLACEMENT OF GRAPHITE CAPSULE IN E7 WITH A COBALT CAPSULE
(VARIATION 4)

FIGURE 4
REPLACEMENT OF FUEL ELEMENT IN E6 WITH A COBALT CAPSULE
(VARIATION 6)

A GAMMA THERMOMETER

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A Gamma Thermometer

Introduction

Part of a Westinghouse Atomic Power Division program to determine the irradiation embrittlement of certain steels involves the use of a device to measure the gamma heating in the periphery of the Saxton Reactor core. We have called the device a Gamma Thermometer. The principle of operation is similar to that of equipment developed some years ago at the Savannah River Laboratory and instrumentation presently used in the Halden Boiling Water Reactor.^{/1}

The author has dealt elsewhere^{/2,3} with engineering methods of estimating gamma heat in certain reactor positions. However, the use of a measuring device made of the material of interest in the exact core location is more accurate than any estimation method. If the gamma heating is known, the experimenter can design for it. Several years ago an in-pile creep experiment^{/5} failed because the specimen melted in one of the grips; a region where the local heat rating had been underestimated.

Design

The Gamma Thermometer is shown as sections B-B and D-D in Figure 1. The central pin, item 13, is an 1/8" diameter 3" long rod of the material from which the irradiation specimens are made. An inch from one end is a section of slightly larger diameter which is force fitted into a stainless steel collar. The 1-5/8" diameter collar, item 7, provides a path for heat generated in the pin to be conducted to the coolant. The coolant, pressurized water at 288°C, flows mainly by thermal convection up the tube, item 4, and out through slots in the side.

Three thermocouples*, item 12, are brazed into the collar at equally spaced

*Chromel-alumel, stainless steel sheathed, ungrounded junction.

points around it. Penetrations in the collar permit the thermocouples to be attached at each end of the pin and at the center of the enlarged section of the pin. This arrangement makes it possible to measure the temperature difference, as well as the actual temperatures, over a 1" long pin and a 2" long pin. Even failure of all but one couple, after full reactor power is reached, would not make the device unworkable. The upper and lower caps, items 6 and 8, are welded to the collar after the thermocouples are attached. A hole joining the two cavities makes it possible to evacuate the device through a small side hole and seal it by electron beam welding. The upper cap is attached to a support tube.

The thermocouple sheaths serve as flow spacers where they parallel the outside of the upper cap. The couple leads are taken inside of the support tube to protect them from hydraulic vibration in the space between the top of the core and the vessel head penetration. Violent cross currents make it necessary to provide a very sturdy connection between the vessel head and the gamma thermometer. The remainder of the drawing depicts the dummy fuel assembly in which the device is located and the conoseal joints at the vessel head penetration which makes it possible to insert and remove the gamma thermometer without removing the head of the reactor vessel.

Principle of Operation

The steady state, one-directional heat conduction in the gamma thermometer is described by equation (1):

$$T_O - T_L = \frac{R_\gamma l^2 \bar{\rho}}{8K} \quad (1)$$

where T_O = temperature at free end of pin, °C

T_L = temperature at midpoint of collar, °C

R_γ = mean gamma heat over length, watts/gram

$\bar{\rho}$ = mean density over length, g/cc

K = thermal conductivity at mean temperature of pin, W/cm-°C

l = distance between thermocouples measuring T_O and T_L , cm

The assumption is that there are not heat losses from the sides of the constant cross section pin.

The two legs of the pin provide a check on the heat loss assumption. The ratio of the temperature differences, $T_O - T_L$, in the two legs should be 4:1 since one leg is twice as long as the other.

At temperatures above 550°C, the radiation heat losses become significant and the ratio of temperature differences drops below 4:1. The extent of the deviation from 4:1 can be used to estimate the radiant heat loss. Once a relationship is established between each of the temperatures and reactor power, any two of the thermocouples can fail without destroying the usefulness of the thermometer.

Calibration

Prior to insertion of the gamma thermometer in the reactor, it may be used to estimate the effective thermal conductivity of the pin material. This is done by rapidly chilling the thermometer from its maximum expected operating temperature. The device is heated until all thermocouples read the same, then the device is withdrawn from the furnace and flooded with tap water until the pin is nearly down to room temperature. A Visicorder is used to continuously monitor the transient. The differential rate of cooling of the two positions on the pin leg - the free end and the collar - is fitted to a mathematical model of the transient cooling process. The objective is to find a value of thermal conductivity appropriate to the differential temperature expected over the length of the pin leg in-pile.

The model* of cooling is expressed:

$$\frac{U(o, t)}{U_o} = 1 - \sum_{n=0}^{\infty} (-1)^n \left\{ 2 \operatorname{erfc} \left[\frac{ml}{2\sqrt{kt}} \right] - \frac{l}{U_o\sqrt{\pi}} \int_{\frac{ml}{2\sqrt{kt}}}^{\infty} F \left[t - \frac{m^2 l^2}{4k\lambda^2} \right] e^{-\lambda^2} d\lambda \right\} \quad (2)$$

where $m = 2n + 1$, $n = 0, 1, 2, \dots, \infty$

$U(o, t)$ = temperature difference, at time t , between free end of pin and tap water, $^{\circ}\text{C}$

U_o = $U(o, t)$ at $t = 0$, $^{\circ}\text{C}$

t = time since start of cooling, sec.

k = $\frac{K}{\rho C_p}$, thermal diffusivity, $\frac{\text{cm}^2}{\text{sec}}$

K = thermal conductivity, $\text{w/cm-}^{\circ}\text{C}$

ρ = density, g/cc

C_p = heat capacity, $\text{w-sec/g-}^{\circ}\text{C}$

l = length of pin leg, cm

$\operatorname{erfc}[]$ = complementary error function

λ = a dummy variable in the convolution integral

$F[t]$ = temperature difference, at time t , between juncture of pin and collar, and tap water, $^{\circ}\text{C}$

A cooling curve is shown in Figure 2. $F(t)$ is expressed as a fraction of the initial temperature difference, U_o . A straight line has been fitted arbitrarily to the data to establish the constants in an exponential decay law:

$$\frac{F(t)}{U_o} = \exp [-0.0371 t] \quad (3)$$

where $t = 0$ occurs 15.5 seconds after the water is turned on (the capsule body has a high thermal inertia).

* Derivation in appendix.

In this test, $U_o = 416.6 - 21.0 = 395.6^\circ\text{C}$.

Figure 3 gives the temperature response of the free end of the pin, expressed as $\frac{U(o,t)}{U_o}$, versus the same time scale as Figure 2. Equation (3) is also plotted in Figure 3.

A computer program has not yet been prepared to solve the rather slowly converging equation (2). When it is, the following data will be inserted and the curve which appears to best fit the data will be selected:

$$\begin{aligned}\rho &= 7.8 \text{ g/cc} \\ C_p &= 0.4609 \text{ w-sec/g-}^\circ\text{C} \\ l &= 2.54 \text{ cm}\end{aligned}$$

Using K-values centering around $0.4256 \text{ w/cm-}^\circ\text{C}$, the temperature ratio, $U(o,t)/U_o$, will be computed for t-values between 0 and 65 seconds. This is the time range over which equation (3) for $F(t)$ appears to be valid.

At the present time insufficient data have been accumulated with the gamma thermometer to make any quantitative statement regarding the accuracy or precision.

In-Pile Operation

In addition to measuring the gamma heating at reactor startup, the continued operation of the thermometer indicates how the local gamma flux changes during the life of a given core loading. For compact cores or batch loaded reactors, this variation can be significant. Longer term irradiation over many fuel loadings permits measurement of thermal conductivity changes in the pin material. Thermal conductivity is expected to decrease with accumulated fast neutron exposure since electrical resistivity increases, and, according to the Wiedemann-Franz law, the product of electrical resistivity and thermal conductivity at a given temperature should be constant.

With some modification, the gamma thermometer could be used to measure electrical as well as thermal properties of a variety of pin materials in-pile. Whether

or not the Wiedemann-Franz law should be obeyed* in a nuclear environment is not clear, but if the thermal conductivity does decrease, the temperature gradient - for the same gamma flux - will rise in the pin. The implication of this for pressure vessels and core internals may be serious.

Besides being useful as a research tool, the gamma thermometer can be used in more prosaic tasks, such as mapping the gamma flux distribution in the core. In large reactor cores, especially those which follow the load, the gamma thermometer may be useful in assessing any xenon (and power) spatial oscillations.

Summary

A gamma thermometer has been constructed to perform a number of tasks in addition to the measurement of in-core gamma heat. Among these other measurements are the transient thermal properties of reactor structural material, the effect of irradiation on thermal conductivity, and spatial power oscillations in large reactor cores.

*This law is only approximate for most metals, largely because it ignores the lattice component of thermal conduction.

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APPENDIX

Derivation of the Transient Temperature Expression, Equation (2) in the Text

The cooling of a laterally insulated bar is analogous to heating it. The analysis of the heated bar is given by Churchill⁴; our cooling analysis parallels it.

At the free end of the bar (pin), there is no heat transfer. The bar is initially at a uniform temperature. The clamped end (collar) of the bar decreases in temperature in an arbitrary fashion. In a one-dimensional coordinate system, $X = 0$ at the free end, $X = 1$ at the clamped end. Thermal diffusivity is k . The temperature at point, x , and time, t , is defined $U(x,t)$. The partial differential equation and boundary conditions are expressed:

$$\text{D.E.: } \frac{\partial U(x,t)}{\partial t} = k \frac{\partial^2 U(x,t)}{\partial x^2}, \quad (0 < x < 1, t > 0)$$

$$\text{B.C.: } \begin{cases} U(x,0) = U_0, & (0 < x < 1) \\ \frac{\partial U(0,t)}{\partial x} = 0, & (t > 0) \\ U(1,t) = F(t), & (t > 0) \end{cases}$$

Laplace transforms:

$$\text{D.E.: } sU(x,s) - U_0 = \frac{\partial^2 U(x,s)}{\partial x^2}$$

$$\text{B.C.: } \frac{\partial U(0,s)}{\partial x} = 0, \quad U(1,s) = f(s)$$

General solution:

$$\begin{aligned} U(x,s) &= \frac{U_0}{s} + C \cosh x \sqrt{\frac{s}{k}} \\ \text{for } \frac{U(0,s)}{x} &= 0 \end{aligned}$$

$$\text{At } U(1,s) = f(s) = \frac{U_0}{s} + C \cosh l\sqrt{\frac{s}{k}}$$

$$C = \frac{f(s) - \frac{U_0}{s}}{\cosh l\sqrt{\frac{s}{k}}}$$

Solution:

$$U(x,s) = \frac{U_0}{s} + \left(f(s) - \frac{U_0}{s} \right) \frac{\cosh x\sqrt{\frac{s}{k}}}{\cosh l\sqrt{\frac{s}{k}}}$$

$$\text{or, } U(x,s) = \frac{U_0}{s} + f(s) \frac{\cosh x\sqrt{\frac{s}{k}}}{\cosh l\sqrt{\frac{s}{k}}} - \frac{U_0}{s} \frac{\cosh x\sqrt{\frac{s}{k}}}{\cosh l\sqrt{\frac{s}{k}}}$$

Inverse transforms:

$$L^{-1} \left(\frac{U_0}{s} \right) = U_0$$

$$\text{write } q = \sqrt{\frac{s}{k}} :$$

Expand:

$$\frac{\cosh x q}{\cosh l q} = \sum_{n=0}^{\infty} (-1)^n \left(\exp [-q (ml - x)] + \exp [-q (ml + x)] \right)$$

$$\text{where } m = 2n + 1$$

Note that:

$$L^{-1} \left(\frac{1}{s} e^{-\alpha\sqrt{s}} \right) = \text{erfc} \left(\frac{\alpha}{2\sqrt{t}} \right)$$

for $\alpha \geq 0$

Therefore:

$$L^{-1} \left(\frac{1}{s} \exp [-q (ml \pm x)] \right) = \text{erfc} \left(\frac{ml \pm x}{2\sqrt{kt}} \right)$$

Appendix

Then:

$$U(x,t) = U_0 - U_0 \sum_{n=0}^{\infty} (-1)^n \left\{ \operatorname{erfc} \left[\frac{ml-x}{2\sqrt{kt}} \right] + \operatorname{erfc} \left[\frac{ml+x}{2\sqrt{kt}} \right] \right\} + L^{-1} \left\{ f(s) \frac{\cosh x q}{\cosh l q} \right\}$$

As before:

$$\frac{\cosh x q}{\cosh l q} = \sum_{n=0}^{\infty} (-1)^n \left\{ \exp [-q (ml-x)] + \exp [-q (ml+x)] \right\}$$

and,

$$L^{-1} \left\{ e^{-\alpha \sqrt{s}} \right\} = \frac{\lambda}{2\sqrt{\pi t^3}} \exp \left[-\frac{\alpha^2}{4t} \right]$$

where

$$\lambda = \frac{ml+x}{\sqrt{k}}$$

then,

$$L^{-1} \left\{ \frac{\cosh x q}{\cosh l q} \right\} = \frac{1}{2\sqrt{\pi t^3}} \sum_{n=0}^{\infty} (-1)^n \left\{ \left(\frac{ml-x}{\sqrt{k}} \right) \exp \left[-\frac{(ml-x)^2}{4kt} \right] + \left(\frac{ml+x}{\sqrt{k}} \right) \exp \left[-\frac{(ml+x)^2}{4kt} \right] \right\}$$

Which is combined with $f(s)$ by convolution:

$$L^{-1} \left\{ f(s) \frac{\cosh x q}{\cosh l q} \right\} = \int_0^t \frac{F(t-\tau)}{2\sqrt{\pi k\tau^3}} \sum_{n=0}^{\infty} (-1)^n \left\{ (ml-x) \exp \left[-\frac{(ml-x)^2}{4k\tau} \right] + (ml+x) \exp \left[-\frac{(ml+x)^2}{4k\tau} \right] \right\} d\tau$$

Appendix

Change variables:

$$\text{Let, } \lambda_1^2 = \frac{(ml - x)^2}{4 k \tau}$$

$$\text{and, } \lambda_2^2 = \frac{(ml + x)^2}{4 k \tau}$$

Split up integrals, letting,

$$\tau_1 = \frac{(ml - x)^2}{4 k \lambda_1^2}$$

$$d\tau_1 = \frac{-2 (ml - x)^2 d\lambda_1}{4 k \lambda_1^3}$$

$$\text{integral limits } \begin{cases} @ \tau_1 = 0, \lambda_1 = \infty \\ @ \tau_1 = t, \lambda_1 = \frac{ml - x}{2 \sqrt{kt}} \end{cases}$$

$$\frac{1}{\sqrt{\tau_1^3}} = \left(\frac{2 \sqrt{k} \lambda_1}{ml - x} \right)^3 = \frac{8 k^{\frac{3}{2}} \lambda_1^3}{(ml - x)^3}$$

Then substitution leads to:

$$\begin{aligned} & \sum_{n=0}^{\infty} (-1)^n \int_0^t \frac{F(t - \tau_1)}{2 \sqrt{\pi k \tau_1^3}} (ml - x) \exp \left[\frac{-(ml - x)^2}{4 k \tau_1} \right] d\tau_1 \\ &= \sum_{n=0}^{\infty} \frac{(-1)^n 2}{\sqrt{\pi}} \int_{\frac{ml - x}{2 \sqrt{kt}}}^{\infty} F \left[t - \frac{(ml - x)^2}{4 k \lambda^2} \right] \exp [-\lambda^2] d\lambda \\ & \qquad \qquad \qquad m = 2n + 1 \end{aligned}$$

and a similar expression for τ_2 .

Combining these expressions and setting $x = 0$:

$$\frac{U(0,t)}{U_0} = 1 - \sum_{n=0}^{\infty} (-1)^n \left\{ 2 \operatorname{erfc} \left[\frac{m l}{2 \sqrt{k t}} \right] - \frac{4}{\sqrt{\pi} U_0} \int_{\frac{m l}{2 \sqrt{k t}}}^{\infty} F \left[t - \frac{m^2 l^2}{4 k \lambda^2} \right] e^{-\lambda^2} d\lambda \right\}$$

$$m = 2n + 1$$

SECTION A
SCALE 2:1

SECTION B-B
SCALE 2:1

SCALE 2:

— HOLE TO BE WELDED
AFTER EVACUATION

FIGURE

		BILL OF MATERIAL				REV. 0001	
ITEM	QTY	ITEM	QUANTITY DRAWING	REVISION DATE	BY	CHK	
1	1	TUBE SUPPORT	1				
2	1	FLUTE	1				
3	1	CAPSCREW	1				
4	1	WASHER	1				
5	1	WASHER	1				
6	1	WASHER	1				
7	1	WASHER	1				
8	1	WASHER	1				
9	1	WASHER	1				
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91	1	WASHER	1				
92	1	WASHER	1				
93	1	WASHER	1				
94	1	WASHER	1				
95	1	WASHER	1				
96	1	WASHER	1				
97	1	WASHER	1				
98	1	WASHER	1				
99	1	WASHER	1				
100	1	WASHER	1				

~~A. 80. FR INDUSTRIAL FASTENER DIV. JENKINTOWN, PA.~~
B. TO BE SUPPLIED BY ENGINEER.
C. 80. SWAGelok 300-6-(042)-316

NOTE:

- 1- WELD MADE AFTER INSERTION OF METALLURGICAL SPECIMEN
- 2- WELD MADE AFTER ASSEMBLING OF BULB ITEM'S 6, 7, 8 ETC.
- 3- ~~WELD MADE AFTER ASSEMBLY OF ITEM 3 ON ITEM 6 AND THROUGH ITEM 2~~
- 4- ~~WELD MADE AFTER ITEM 4 HAS BEEN POSITIONED AND DRILLED FOR ITEM 3 TO BE REMOVED AND REMOVE CURS BEFORE WELDING~~
- 5- WELD PER WPS 2927G-3-1.
- 6- BRAZE PER WPS 2927I-3 USING NICKROBRAZE #50 BRAZE ALLOY
- 7- BRAZING PROCEDURE TO BE SUBMITTED FOR WAPD APPROVAL. (PREBRAZE #10 90% GDN GOLD, 10% NICKEL)
- 8- BRAZED ASSEMBLY MUST BE HYDROSTATIC TESTED AT 3150 PSIG 351 AT ROOM TEMPERATURE. TESTING PROCEDURE MUST BE SUBMITTED TO WAPD FOR APPROVAL.
9. ALL BEND RADII .060 MIN.

[illegible]

Power and Fuel Depletion Calculations for Non-
Instrumented Capsules in NASA's Plumbrook
Reactor Facility

R. E. Schreiber

INTRODUCTION

Part of a cooperative irradiation program between Westinghouse Atomic Power Division and NASA* involves the irradiation of a number of fuel bearing, non-instrumented capsules; both UO_2 and UC fuel pins are included. A large number of fuel pins, in groups of 4, are being irradiated at power levels between 20 and 60 kw/ft to burnups between 10,000 and 80,000 MWD/MTU. During the course of the irradiation, several options are available for holding the power in each capsule (group of 4 pins) fairly constant:

1. As power declines, the absorber sleeve around the capsule can be exchanged for one which depresses the flux less.
2. The reactor position of the capsule can be changed to boost the flux.
3. A burnable poison sleeve - boronated stainless steel - can be used to help hold the power constant. It must be frequently replaced because the B-10 burns out faster than the fuel.

All of these methods of capsule power regulation are being used now or will have been employed by the time the program is completed. To schedule the absorber and reactor position changes and to predict when the desired burnup has been reached, it was necessary to develop a model for power and U-235 atomic density changes with exposure time. The model requires data on the unperturbed flux in the various reactor positions, as well as data on the perturbing effects of fuel and absorber sleeves. A number of gold foil activation runs were made with dummy capsules containing UO_2 fuel pellets in the low power Mockup Reactor (MUR) at the Plumbrook Reactor Facility (PBRF). This paper describes the capsule design, the MUR experiments, and the application of the model to determine a typical scheduling sequence.

* NASA Contract Number NAS 3-3392PB

DESIGN OF THE CAPSULE AND ASSOCIATED COMPONENTS

Figure 1 shows the general assembly which is placed in the primary beryllium reflector of the PBR. The reflector piece is also shown. The assembly envelope is essentially a 2-inch diameter cylinder, 40 inches long. Items 1-4 are the capsules (four different types). Each of the two capsules contains four fuel pins. Item 5 is a latch that holds the capsules in the tubular holder which is riveted to the upper fitting, a plunger assembly. The upper fitting, Item 7, also serves to lock the holder into the reflector piece and acts as a grab for handling the assembly. The spring pin assembly, Item 6, holds the capsules in place axially, but there is no control of radial orientation, i.e., the position of each fuel pin in the capsule is not known.

The capsule design is given in Figure 2. Four fuel pins, contained in NaK (78% K) or Na, canned in Type 304 stainless steel, are held equally spaced on a 1-inch diameter circle. Four aluminum flow diverters are screwed to the capsule body. Not shown are recesses in which tensile wires of various materials are trapped between each diverter and the body. There are 13 fuel-bearing capsules in the program and three dummies with the same flow characteristics. The capsule bodies are either aluminum or stainless steel.

A typical canned fuel pin and associated data are shown in Figure 3. Three tensile wires are attached to the fuel pin by means of spiders which fit over the end plugs. The wires see the temperature of the liquid metal in the annulus.

Figure 7 shows the fuel pin itself. The enriched fuel column, UC or UO_2 (numbered pellets) occupies the center 4 inches of the pin. The UO_2 and UC enrichments for each capsule were selected to give the same linear power. Most of the capsules contain three UO_2 pins and one UC pin.

The characterization of the fuel materials and their irradiation behavior will be reported elsewhere.

INITIAL AND BURNUP RELATION OF POWER AND FLUX IN THE FUEL PINS

The thermal (2200 m/s) cross sections for absorption and fission in U^{235} were adjusted for temperature (180°F) and non-1/v behavior. Values selected were:

$$\sigma_f^{25} = 509 \times 10^{-24} \text{ cm}^2$$

$$\sigma_a^{25} = 607 \times 10^{-24} \text{ cm}^2$$

The number density of U^{235} was obtained as a function of enrichment fraction, E:

$$N_{UO_2}^{25} = \frac{0.6 \times 10^{24} \text{ atoms/mole} \times 10.74 \frac{\text{gUO}_2}{\text{cc}} \times 0.8815 \frac{\text{gU}}{\text{gUO}_2} \times E}{235 \text{ g/mole}} \quad (1)$$

$$= 0.0242 \times 10^{24} E, \text{ atoms } U^{235}/\text{cc}$$

$$N_{UC}^{25} = \frac{0.6 \times 10^{24} \times 13.3 \times 0.951 \times E}{235} \quad (2)$$

$$= 0.0323 \times 10^{24} E, \text{ atoms } U^{235}/\text{cc}$$

where $E = \text{g } U^{235} / \text{g total U}$

The fission density, $\Sigma_f^{25} \phi_p$, is related to the linear power, P, in these fuel pins by the expression:

$$\Sigma_f^{25} \phi_p = \frac{3.1 \times 10^{10} \frac{\text{fissions}}{\text{watt-sec}} \times P}{\frac{\pi}{4} (0.2995 \text{ in})^2 \times 6.45 \frac{\text{cm}}{\text{in}^2}} \cdot \frac{\text{fissions}}{\text{cc-sec}} \quad (3)$$

where P = linear power rating of each pin, watts/cm

ϕ_p = mean perturbed flux in the fuel, $\text{n/cm}^2\text{-sec}$

The macroscopic fission cross section,

$$\Sigma_f^{25} = N^{25} \sigma_f^{25} \quad (4)$$

FIGURE 2 TRANSIENT TEMPERATURE AT JUNCTURE OF PIN AND COLLAR OF GAMMA THERMOMETER.

FIGURE 3 TRANSIENT TEMPERATURE AT FREE END OF PIN.

$$\text{thus } (\Sigma_f^{25})_{UO_2} = 12.33E, \text{ cm}^{-1} \quad (4a)$$

$$(\Sigma_f^{25})_{UC} = 16.41E, \text{ cm}^{-1} \quad (4b)$$

The perturbed flux corresponding to linear power, P, and enrichment E, is determined from:

$$\phi_p = \frac{\Sigma_f^{25} \phi_p}{\Sigma_f^{25}} \quad (5)$$

$$\text{So that, } (\phi_p)_{UO_2} = 5.54 \times 10^9 \left(\frac{P}{E} \right)_{UO_2}, \text{ n/cm}^2\text{-sec} \quad (5a)$$

$$(\phi_p)_{UC} = 4.17 \times 10^9 \left(\frac{P}{E} \right)_{UC}, \text{ n/cm}^2\text{-sec} \quad (5b)$$

Conversely, the power for a given perturbed flux and enrichment,

$$P_{UO_2} = 1.805 \times 10^{-10} (\phi_p E)_{UO_2}, \text{ watts/cm} \quad (5c)$$

$$P_{UC} = 2.398 \times 10^{-10} (\phi_p E)_{UC}, \text{ watts/cm} \quad (5d)$$

The variation of power with time is determined from the proportion:

$$\frac{P(t)}{P_o} = \frac{N(t) \phi_p(t)}{N_o^{25} (\phi_p)_o} \quad (6)$$

where P_o , N_o^{25} , and $(\phi_p)_o$ are the initial power, atom density and perturbed flux and $P(t)$, $N(t)$, $\phi_p(t)$ are these quantities at time, t.

MATHEMATICAL MODEL OF TIME DEPENDENCE OF U^{235} ATOMIC DENSITY

The differential equation for U^{235} depletion is:

$$\frac{dN^{25}}{dt} = -N^{25} \sigma_a^{25} \phi_p \quad (7)$$

The initial conditions are:

$$N^{25} = N_0^{25} \text{ @ } t = 0 \quad (8)$$

$$\phi_p = (\phi_p)_0 \text{ @ } t = 0 \quad (9)$$

As depletion occurs, the perturbed flux increases asymptotically to the unperturbed value. This, in turn, causes an even faster burnup rate. Thus, the desired burnup is achieved in a shorter period than would be estimated using a constant (initial) perturbation factor. In addition, the variation of power in the fuel can be more accurately determined if the rising flux is considered. This makes it possible to schedule reactor position changes and changes in absorber sleeves (stainless steel or aluminum holder assembly) to hold the power in the fuel more nearly constant over its exposure.

Data given by Harrington^{/1} suggest that the perturbation factor for single rods containing U^{235} and 20 to 40 grams of stainless steel per inch can be expressed:

$$\frac{\phi_p}{\phi_u} = \frac{0.75F}{1 + CW} \quad (10)$$

where ϕ_p = perturbed flux, n/cm^2 -sec

ϕ_u = unperturbed flux, n/cm^2 -sec

W = linear weight of U^{235} , g/in.

C = a correction factor to account for the flux depressing effect of the adjacent three fuel pins in the 4-pin bundle

F = flux depression caused by the holder and capsule body

The factors C and F in our configuration were determined in a series of foil activation experiments in the MUR. (See Appendices I and II.)

In this experiment, W is given by:

$$W = 0.452 \times 10^{-21} N^{25} K, \text{ g/in.} \quad (11)$$

$$\text{where } K = \frac{N^{25} \sigma_a^{25} + X \sigma_x}{N^{25} \sigma_a^{25}} \quad (12)$$

The K-term makes W the effective linear weight of U^{235} , since it includes a term for flux depression by other atomic species, as well. The equilibrium Xe^{135} concentration is the principal species to be considered. The maximum equilibrium concentration, X, is reached a few hours after start-up:

$$X = \frac{N^{25} \phi_p \sigma_f^{25} Y}{\lambda_x + \sigma_x \phi_p}, \text{ atoms } Xe^{135}/\text{cc} \quad (13)$$

where Y = 0.062, the yield from fission and from decay of the Te^{135} and I^{135}

$\lambda_x = 2.11 \times 10^{-5} \text{ sec}^{-1}$, the decay constant of Xe^{135}

$\sigma_x = 3.5 \times 10^{-18} \text{ cm}^2$, the thermal neutron absorption cross section of Xe^{135}

$$\sigma_f^{25} = 509 \times 10^{-24} \text{ cm}^2$$

$$\sigma_a^{25} = 607 \times 10^{-24} \text{ cm}^2$$

$$N^{25} = \text{atoms } U^{235}/\text{cc}$$

} defined as before

$\phi_p \geq 4 \times 10^{13} \text{ n/cm}^2\text{-sec}$, the lowest value of perturbed flux in the experiment

If $\phi_p = 4 \times 10^{13} \text{ n/cm}^2\text{-sec}$, or larger, the term, $\sigma_x \phi_p \gg \lambda_x$ so that the flux dependence virtually cancels out.

Under these circumstances, N^{25} cancels in the expression for K, leaving:

$$K \approx 1.048 \quad (12a)$$

this gives:

$$W = 0.473 \times 10^{-21} N^{25} \quad (11a)$$

Each capsule spends half its irradiation time in the upper axial position and the rest in the lower position. The flux ratio, ϕ_p/ϕ_u , at the mid-point of the fuel column in the pin, is slightly different for the two axial positions (see Figure A-I-4). The mean of the flux ratios was taken to calculate a correction factor:

$$C = 1.5494 \quad (14)$$

Substituting expressions (10), (11a), (14) into equation (7), we have

$$\frac{dN^{25}}{dt} = \frac{-0.75 \sigma_a^{25} N^{25} F \phi_u}{1 + 0.733 \times 10^{-21} N^{25}} \quad (15)$$

For initial condition (8), this integrates to:

$$\ln \left(\frac{N_0^{25}}{N(t)^{25}} \right) + BN_0^{25} \left(1 - \frac{N(t)^{25}}{N_0^{25}} \right) = At \quad (16)$$

$$\text{where } A = 0.75 \sigma_a^{25} F \phi_u, \text{ sec}^{-1}$$

$$B = 0.733 \times 10^{-21}, \text{ cc/atom}$$

$$\phi_u = \text{average unperturbed flux for the upper and lower capsule positions, }^* \text{ n/cm}^2\text{-sec}$$

Three different pairs of UO_2 and UC enrichments are being used in the experiment. These give $N_0^{25} = 1.988, 2.924, \text{ and } 4.093 \times 10^{21} \text{ atoms/cc.}$

* Also averaged over the reactor cycle, which is short compared to the irradiation times considered in this experiment.

The F-factors are obtained from Figure A-II-1. Not enough tests were run to establish F for all three U^{235} concentrations or for all possible combinations of capsule and holder material, so it is necessary to interpolate or extrapolate as indicated. It is not clear why F should be a function of U^{235} concentration. Where F is a constant, it is the product,

$$F = F_{\text{capsule}} F_{\text{holder}}, \quad (17)$$

but where boronated stainless steel is used in either or both the capsule and holder, the product, At, in equation (16) is replaced with:

$$A \int_0^t f_B(t) dt \quad (18)$$

where $f_B(t)$ = flux depression due to B^{10}

The fact that the differential equations for B^{10} burnup and U^{235} burnup are not strongly coupled makes it possible to integrate the B^{10} equation separately, as illustrated in Appendix II.

Equation (16) and expression (18) are used to find the time at which a desired N^{25} is reached. The graphic integral, $\int_0^t f_B dt$, is solved for time. When it is desired to boost power by changing the holder from stainless or boron-stainless to aluminum, the time scale is changed by the ratio of F-factors after and before the holder change. A change in reactor position changes ϕ_u . This also has the effect of changing the time scale by the ratio of ϕ_u after to before the change. The variation of power with time and holder and position changes is determined by combining equations (6), (10), (11a) and (14) with the instantaneous N^{25} :

$$P(t) = \frac{N^{25} P_o}{N_o^{25} (\phi_p)_o} \left[\frac{0.75 F \phi_u}{1 + 0.733 \times 10^{-21} N^{25}} \right] \quad (19)$$

where $\frac{P_o}{(\phi_p)_o}$ is obtained from equation (5c) or (5d)

and $F = F_{\text{capsule}} F_{\text{holder}} f_B$

($f_B = 1.0$ if no B^{10} is present).

ILLUSTRATION OF POWER AND DEPLETION CALCULATION

Capsules 1, 2 and 3 in this experiment are to be irradiated* in reflector Position IA-1 or LA-11. The unperturbed flux in these positions, equivalent to the average distribution over a reactor cycle, is displayed in Figure 5. The enrichment in this group of capsules is 8.5% (UO_2) which gives:

$$\frac{P_o}{(\phi_p)_o} = 46.7 \times 10^{-14} \frac{\text{kw/ft}}{\text{n/cm}^2\text{-sec}}$$

Gamma heat in these positions can be as high as 2 kw/ft, but is ignored in this analysis.

The product,

$$F = F_{\text{capsule}} F_{\text{holder}} = .786 \times .81 = 0.637 \text{ for capsules 2 and 3}$$

$$F = 0.81 \text{ for capsule 1}$$

$$\phi_u = 2.875 \times 10^{14} \text{ n/cm}^2\text{-sec}$$

$$N_o = 1.988 \times 10^{21} \text{ atoms/cc}$$

$$f_B, \text{ initial} = 0.64 \text{ for capsules 2 and 3}$$

$$f_B(t) = 1.0 \text{ for capsule 1}$$

The variation of f_B and $\int_0^t f_B dt$ with time is given in Figure 6.

The final U^{235} atomic concentration in each capsule:

$$\begin{aligned} N_f (\text{Cap. 1}) &= 1.718 \times 10^{21} \text{ atoms/cc (After actual* time of } 3.24 \times 10^6 \text{ effective full power seconds)} \\ N_f (\text{Cap. 2}) &= 1.15 \times 10^{21} \text{ atoms/cc} \\ N_f (\text{Cap. 3}) &= 0.664 \times 10^{21} \text{ atoms/cc} \end{aligned} \quad \left. \vphantom{\begin{aligned} N_f (\text{Cap. 1}) \\ N_f (\text{Cap. 2}) \\ N_f (\text{Cap. 3}) \end{aligned}} \right\} \text{(Target values)}$$

At $t = 20 \times 10^6$ sec, the holder containing Capsule 3 (and a dummy) is changed from stainless to aluminum. The F-factor for a stainless holder (at the N^{25} corresponding to 20×10^6 sec) is 0.89. The power is boosted by $(0.89)^{-1}$. The time scale is shortened for power and number density by the factor 0.89 after 20×10^6 sec.

* Capsule 1 has completed its irradiation and has been examined. A report is due to be issued soon, NASA-CR-72019.

If we estimate the time to reach $N^{25} = 1 \times 10^{21}$ atoms/cc based on the initial perturbation factor, we overestimate by almost 20%. At a true $t = 17.7 \times 10^6$ seconds, this is an error of over one month at full power.

COMPARISON WITH RESULTS FROM CAPSULE 1

The mean fission density, determined from Cs-137 measurements on the UO_2 rods, was found to be 0.2413×10^{21} f/cc. This corresponds to a final U-235 atomic density of 1.700×10^{21} atoms/cc, which compares well with the calculated value, $N_f = 1.718 \times 10^{21}$ atoms/cc.

Based on a calculated mean initial power of 33.19 kw/ft, the estimated zone of melting was 55% of the fuel diameter. The melted region observed in the photomicrographs averaged slightly greater than 46% of the fuel diameter.

The estimated fission gas release from UO_2 based on 33.19 kw/ft, mean initial power, was 40.2%. Measured gas release from the UO_2 fuel pins averaged 46.47%.

The good agreement of predicted and measured quantities which are related to power suggests that the model used to calculate power is a reasonable one. The good agreement between predicted and measured burnup is partly fortuitous since some uncertainty is introduced by Cesium migration in the largely molten fuel.

SUMMARY

A method has been developed for simply and accurately determining depletion and power in fuel-bearing capsules as functions of time. The method is suitable for hand calculations, or may be programmed for machine computation. Key to the method is an adequate series of flux perturbation measurements to determine the empirical factors in the formulae. A simplified method is given for treating burnable poisons, such as boron-10, used in conjunction with the experiment. The basic equations:

Depletion:

$$\ln \left(\frac{N_0}{N^{25}(t)} \right) + BN_0^{25} \left(1 - \frac{N^{25}(t)}{N_0^{25}} \right) = At$$

where

$$A = 0.75 \sigma_a^{25} F \phi_u, \text{ sec}^{-1}$$

$$t = \text{time, sec}$$

$$B = CW/N_0^{25}, \text{ cc/atom}$$

$$N_0^{25} = \text{initial U-235 atomic number density, atoms/cc}$$

$$W = \text{effective weight of U-235 per unit length of fuel (weighted for neutron absorption of equilibrium Xe in fuel), g/in.}$$

$$C = \text{factor determined in flux perturbation measurements, dimensionless}$$

$$N^{25}(t) = \text{U-235 number density at time, t, atoms/cc}$$

$$\sigma_a^{25} = \text{microscopic absorption cross section for thermal neutrons by U-235, cm}^2$$

$$\phi_u = \text{mean unperturbed thermal neutron flux during irradiation, n/cm}^2\text{-sec}$$

$$F = \text{measured or estimated flux depression due to absorber sleeves surrounding fuel pins, dimensionless}$$

If the absorber sleeve contains rapidly burnable poison, it becomes $A \int_0^t f_B(t) dt$,

where $f_B(t)$ is the time-dependent flux depression factor due to the burnable poison.

Power:

$$\frac{P(t)}{P_o} = \frac{N_o^{25}(t)}{N_o^{25}(\phi_p)_o} \left[\frac{0.75 F \phi_u}{1 + BN^{25}(t)} \right]$$

where

- $P(t)$ = linear power at time, t , w/cm (or, kw/ft)
- P_o = initial linear power, units consistent with $P(t)$
- $(\phi_p)_o$ = initial perturbed flux corresponding to P_o , n/cm^2 -sec

Other terms are as defined for the depletion equation. If burnable poison is present, F becomes $Ff_B(t)$.

APPENDIX I

FLUX PERTURBATION MEASUREMENTS - C-FACTOR CALCULATIONS

Unperturbed thermal flux measurements have been made in test position LA-1 of the MUR. The measurements were made with the beryllium shims 6, 7, and 8 at 28.00 inches and "reg" rods 1 and 2 at 100%. The fueled bank height for all runs was approximately 17.9 inches. The measurements were made with the beryllium plug removed from the test position and replaced with a special Lucite rig which allows vertical traverses to be made on four sides and in the center. A schematic cross section is shown on Figure A-I-1.

The perturbed measurements were also made with the beryllium shims at 28.00 inches, "reg" rods 1 and 2 at 100% and the fueled bank at approximately 17.9 inches. These measurements required two runs since only one mockup capsule was available. The runs were made using an aluminum capsule and an aluminum holder and 12.5% enriched fuel. Since only one capsule was available, the position of the second capsule was occupied by PBR capsule 12 which has the same composition. Dysprosium foils were placed between the fuel pellets to measure the thermal flux. Gold foils in the core and gold wires in the beryllium reflector were used to determine the reactor power.

Figure A-I-1 is the vertical flux distribution taken from the three specified locations. This is the unperturbed thermal flux. Figures A-I-2 and A-I-3 are cross-plots of the three curves from Figure A-I-1 and show the flux distribution across the hole. The average cadmium ratio was 10.

Figure A-I-4 is a synthesized vertical plot taken from the curves of Figures A-I-2 and A-I-3. It was made at a traverse which would be equivalent to the center of the fuel for fuel pin No. 1. In this case the fueled capsule was inserted into LA-1 such that pin No. 1 was at a 45° angle to LB-2 and pins 2, 3 and 4 were inserted clockwise. The axis of the fuel is 0.5 inches from the axis of LA-1.

Tables A-I-1 and A-I-2 give the perturbed thermal flux distribution in the

Appendix I

fueled capsules and Table A-I-3 gives the unperturbed thermal flux distribution. The numbers at the top of the columns for the unperturbed flux represent the Lucite strip number and the letters indicate their position in the hole. The reference point for the vertical distribution is the core horizontal midplane. Axial measurements for the perturbed flux were made in fuel pin No. 1 for both the top and bottom capsules and radial measurements were made in the other three pins between fuel pellets 2 and 3. Positions LA-1 and LA-11 are symmetrical.

The values reported have an error of 25 percent at the 95 percent confidence level. The thermal flux values given in Table A-I-1 for the bottom capsule are nominally the highest values that could be seen over a reactor cycle because of the low bank position during the measurements. At the top capsule, where the flux would increase with increasing rod bank height, the maximum value will be less than the maximum value that the bottom capsule sees.

In Table A-I-1, the unperturbed fluxes (from Figure A-I-4) at the same elevation as the perturbed fluxes are listed, along with the ratio - the perturbation factor. These ratios and perturbed fluxes are also shown in Figure A-I-4.

Ignoring the Wilkins end effect (flux peaking), the perturbation factors for the upper and lower capsule position are fitted to the expression:

$$\frac{\phi_p}{\phi_u} = \frac{0.75}{1 + WC} \quad (A-I-1)$$

where $W = 1.385 \text{ g U-235/in. (0.300" diameter, 95\% T.D., 12.5\% enriched UO}_2\text{)}$

For the upper capsule,

$$\frac{\phi_p}{\phi_u} = 0.2155, \text{ which gives } C = 1.7908$$

Appendix I

For the bottom capsule,

$$\frac{\phi_p}{\phi_u} = 0.8613, \text{ which gives } C = 1.3504$$

No explanation is offered for the difference in flux ratios, but for an average ratio of 0.8384, $C = 1.5494$. This is the value used in subsequent calculations.

Table A-I-1

Perturbed Vertical Flux Distribution in LA-1, MUR

Rod Bank Height 17.9" Fuel Pin #1, closest to Core, 0.5" off center

	Fuel Pellet Numbers (foils between pellets)	Location above Midplane, inches	Thermal flux/watt, ϕ_p ($\times 10^6$)	Unperturbed flux/watt, ϕ_u ($\times 10^6$) (From Figure A-I-4)	$\frac{\phi_p}{\phi_u}$
Top	1-2	+4-1/16	0.854	3.60	0.2372
Capsule	2-3	3-5/8	0.867	3.95	.2195
	3-4	3-3/16	0.881	4.18	.2108
	4-5	2-3/4	0.960	4.45	.2157
	5-6	2-5/16	1.025	4.72	.2172
	6-7	1-7/8	1.068	4.98	.2145
	7-8	1-7/16	1.219	5.28	.2309
Bottom Capsule	1-2	-5-15/16	1.98	7.43	.2665
	2-3	-6-3/8	1.93	7.35	.2626
	3-4	-6-13/16	1.88	7.25	.2593
	4-5	-7-1/4	1.85	7.10	.2606
	5-6	-7-11/16	1.80	6.90	.2609
	6-7	-8-1/8	1.77	6.73	.2630
	7-8	-8-9/16	1.73	6.50	.2662

6.4.16

Table A-I-2

Perturbed Radial Thermal Flux Distribution in LA-1, MUR

Rod Bank Height 17.9"
Foils located between Pellets 2 and 3

<u>Fuel Pin Number</u>	<u>Thermal Flux/watt ($\times 10^6$)</u>	
	<u>Top Capsule</u>	<u>Bottom Capsule</u>
1	0.867	1.93
2	.723	1.67
3	.606	1.29
4	.725	1.54

Table A I-3

Unperturbed Thermal Flux/watt of Reactor Power in LA-1, MUR

Rod Bank Height 17.9"

<u>Axial Level</u>	<u>Positions shown in inset Figure A-I-1</u>				
	<u>1 (SW)</u>	<u>2 (NW)</u>	<u>3 (NE)</u>	<u>4 (SE)</u>	<u>Center</u>
+12	0.83×10^6	0.68×10^6	0.55×10^6	0.73×10^6	0.73×10^6
10	1.36	1.79	0.92	1.12	1.13
8	2.04	1.16	1.40	1.76	1.82
6	2.91	2.32	1.86	2.36	2.38
4	3.76	3.25	2.48	3.18	3.29
2	5.29	4.17	3.31	4.27	4.33
0	6.32	5.22	3.98	5.15	5.48
- 2	7.35	6.04	4.56	6.16	6.32
- 4	7.98	6.38	4.68	6.33	6.30
- 6	7.71	6.11	4.66	6.20	6.45
- 8	7.23	5.90	4.32	5.88	5.98
-10	5.88	4.46	3.52	4.62	4.58
-12	4.03	3.11	2.46	3.28	3.45

APPENDIX II
FLUX PERTURBATION MEASUREMENTS - F-FACTOR CALCULATIONS

Thermal neutron flux distributions were measured in reflector positions LA-1 and LA-3 of the MUR. All measurements were made with the core poisoned with boron strips to give a rod bank height of 22.2". In addition, the two "reg" rods were at 100% and beryllium shims 6, 7, and 8 were at 28.00 inches.

All thermal flux measurements were made using 6 weight percent dysprosium-aluminum foils. The foils were placed between fuel pellets 2 and 3, numbering from the top of the column of 8 pellets. During the initial measurements, foils provided by WAPD were used exclusively, but when these ran out, some PBRF foils were used. This required that a conversion factor between the two sets of foils be obtained because of differences in dysprosium content and foil size. The problem was resolved by normalizing all WAPD foils to PBRF foils since the latter have been calibrated against gold detectors.

The irradiation schedule is shown in Table A-II-1. The results of the tests are in Table A-II-2.

The thermal flux measurements have a probable error of 20%. Primarily, this is due to uncertainty in power level at the 22.2" rod bank position. Another 5% probable error is due to the variation in dysprosium content of the foils and intercalibration with gold. The accumulated error:

$$\epsilon_p = [(0.20)^2 + (0.05)^2]^{1/2} = 0.206$$

The assemblies used in these measurements were O-ring sealed, demountable fuel pins, cans, and capsules. No sodium or NaK was present; the annular space contained air. The natural UO_2 pellets and axial spacers in the pins to be irradiated in the PBR were simulated with solid aluminum rods. The Dy-Al foils, 0.300" max. diameter and 0.0045" thick, were sandwiched in aluminum guard foils to stop fission products from recoiling into the detector foils.

Appendix II

The average flux for the four locations, given in Table A-II-2, for each enrichment, is compared for different capsule and holder materials. The effect, called the F-factor, of changing from an aluminum to stainless holder with an aluminum or a stainless capsule has been calculated and the results blotted in Figure A-II-1.

The apparent dependence of the F-factor on enrichment is not understood. The value for the flux depression due to a holder change without fuel or capsules present was calculated by a flux screening method described by DeJuren^{1/2}. In this method, the perturbation factor is given by:

$$f = e^{-\tau} (1-\tau) + \tau^2 E_1(\tau) \quad (\text{A-II-1})$$

where $\tau = \Sigma_a t$, dimensionless

Σ_a = macroscopic absorption cross section of sleeve material, cm^{-1}

t = sleeve thickness, cm

$$E_1(\tau) = \int_1^{\infty} \frac{e^{-\tau x}}{x} dx, \text{ dimensionless exponential integral}$$

The exponential integrals have been tabulated^{1/3}. For $\tau < 0.01$, $f \approx 1-2\tau$. For aluminum, Σ_a is very small so that $f \approx 1.0$, but for stainless steel, $f = 0.91$ for the holder.

For a boron-stainless steel sleeve, the flux depression due to B-10 is also treated by the DeJuren method. In the experiment, capsules 2 and 3 will be surrounded by a 0.010 inch thick sleeve of 1.5 weight percent natural boron in stainless steel welded into a recess of the same depth in the outside of a stainless holder. Each capsule also has the same thickness of 1.5 B^N-SST attached in the same manner. For this material,

$$\Sigma_a^{B^N} = 4.86 \text{ cm}^{-1}$$

The combined thickness, t , of the two layers of B-SST is 0.0508 cm. This

Appendix II

gives $\tau = \sum_a^{BN} t = 0.247$, which corresponds to $f_{DeJ.} = 0.64$. The perturbation change with B-10 burnup is handled in a manner similar to the U-235 burnup calculation:

$$f_B = \frac{A}{B + CN^{10}} \quad (A-II-2)$$

where

$$A = B = 1.0$$

$$N^{10} = \text{B-10 atomic number density, atoms/cc}$$

$$f_B = \text{flux depression factor due to B-10}$$

$$C = \text{empirical factor}$$

$$\begin{aligned} \text{For } N_O^{10} &= 1.23 \times 10^{21} \text{ atoms/cc, initially, and } f = 0.64, \\ C &= 0.457 \times 10^{21} \text{ cc/atom} \end{aligned}$$

The differential equation for depletion integrates to:

$$\ln \left(\frac{N_O^{10}}{N(t)^{10}} \right) + C N_O \left(1 - \frac{N(t)^{10}}{N_O^{10}} \right) = \sigma_a \phi_u t \quad (A-II-3)$$

where

$$N_O^{10} = 1.23 \times 10^{21} \text{ atoms/cc}$$

$$C N_O = 1.23 \times 0.457 = 0.562$$

$$\sigma_a = 0.3840 \times 10^{-20} \text{ cm}^2, \text{ the microscopic absorption cross-section for B-10}$$

$$t = \text{time, sec}$$

$$\phi_u = \text{unperturbed flux, n/cm}^2\text{-sec}$$

The unperturbed flux in the absorber sleeve position is modified to account for flux depression by the U-235. Arbitrarily, the factor is taken as:

$$f_{N^{25}} = \frac{1 + \frac{0.75 F}{1 + 0.733 \times 10^{21} N_O^{25}}}{2} \quad (A-II-4)$$

Appendix II

where $N_0^{25} = 1.9 \times 10^{21}$ atoms/cc

and $F = 0.637$

This gives,

$$f_N^{25} = 0.60$$

Then,

$$\phi_u = \phi_o f_N^{25} \quad (\text{A-II-5})$$

where

$$\phi_o = 2.875 \times 10^{14} \text{ n/cm}^2\text{-sec}$$

giving,

$$\phi_u = 1.725 \times 10^{14} \text{ n/cm}^2\text{-sec}$$

so that,

$$\sigma_a \phi_u = 0.6624 \times 10^{-6} \text{ sec}^{-1}$$

Equation (A-II-3) is solved for $N^{10}(t)$ which is substituted in equation (A-II-2) to find $f_B(t)$. This, and the integral, $\int_0^t f_B(t) dt$, are plotted

versus t in Figure 6. Note that the time for substantial burnup of the B-10 is short compared to the time for a significant depletion of U-235. This means the differential equations, describing depletion of B-10 and U-235, do not have to be solved simultaneously.

REFERENCES

1. Harrington, D. G., "Design Criteria and Experience in Nuclear Fuel Irradiations," NAA-SR-6039, July 1961.
2. DeJuren, J. A., and Paschall, R. K., "Thermal Neutron Transmission Through Cadmium Covered Foils," NAA-SR-7770, March 1963.
3. Work Projects Administration, "Tables of Five, Cosine and Exponential Integrals," Vol. I, 1940, Note that $E_1(\tau) = E_1(-x)$.

Table A-II-1
Irradiation Schedule, MUR

<u>Run Number</u>	<u>Enrichment</u>	<u>Capsule Material</u>	<u>Holder Material</u>	<u>Reactor Position</u>	<u>F-factor Sought</u>
1	8.5	Al	Al	LA-1	Al to SST hold.with Al Cap.
2	8.5	Al	SST	LA-1	
3	12.5	SST	Al	LA-3	Al to SST hold.with SST Cap.
4	12.5	SST	SST	LA-3	
5	12.5	Al	Al	LA-3	Al to SST hold.with Al Cap.
6	12.5	Al	SST	LA-3	

Runs 3 and 5 are compared to compute the F-factor for switching capsule material while keeping the holder the same: Al. Runs 4 and 6 are compared to compute the F-factor for changing capsule material while maintaining a stainless holder.

6.4.22

Table A-II

Radial Thermal Flux Distribution, MUR

Run Number	Axial Position	Pin Orientation*	Perturbed flux, 10^6 n/cm ² -sec per watt of reactor power	
1	Top	South	1.12	Average = 0.9925
		West	1.04	
		North	0.93	
		East	0.88	
1	Bottom	South	1.39	1.280
		West	1.47	
		North	1.14	
		East	1.12	
2	Top	South	0.78	0.800
		West	0.95	
		North	0.81	
		East	0.66	
2	Bottom	South	1.03	1.0325
		West	1.24	
		North	1.03	
		East	0.83	
3	Top	South	1.35	1.3825
		West	1.46	
		North	1.36	
		East	1.36	
3	Bottom	South	1.75	1.7775
		West	1.91	
		North	1.81	
		East	1.04	
4	Top	South	1.21	1.1725
		West	1.25	
		North	1.15	
		East	1.08	
4	Bottom	South	1.55	
		North	1.40	
		East	1.37	
5	Top	South	1.63	1.915
		West	2.10	
		North	2.02	
		East	1.91	
5	Bottom	South	2.74	2.5725
		West	2.73	
		North	2.46	
		East	2.36	
6	Top	South	1.38	1.4150
		West	1.50	
		North	1.37	
		East	1.41	
6	Bottom	South	2.04	1.995
		West	2.12	
		North	1.93	
		East	1.89	

* Pin orientation was controlled in these MUR runs, but not in the PBR irradiations.

Typical F-factor calculations:

$$\begin{aligned}
 F_{8.5, \text{ Al cap., hold. Al } \rightarrow \text{ SST}} &= \frac{\phi_{\text{avg, top, run 2}}}{\phi_{\text{avg, top, run 1}}} = \frac{0.800}{0.9925} = 0.8060 \\
 F_{8.5, \text{ Al cap., hold. Al } \rightarrow \text{ SST}} &= \frac{\phi_{\text{avg, bottom, run 2}}}{\phi_{\text{avg, bottom, run 1}}} = \frac{1.0325}{1.280} = 0.8066
 \end{aligned}$$

Range of Values

The F-factors are presented in Figure A-II-1.

647J123

FIGURE 1

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647J123
SUB 1 SEC 2 OF 2

6.4.24

WESTINGHOUSE ELECTRIC CORPORATION
ATOMIC POWER DIV. PITTSBURGH, PA., U.S.A.
TITLE: PLUM BROOK REACTOR FACILITY
IN-PILE IRRADIATION EXPERIMENT
GENERAL ASSEMBLY
SCALE: 1:1
DIMENSIONS IN INCHES
DO NOT SCALE
SUB # 23 SEC. 1 OF 2

674C473

AA

BILL OF MATERIAL					NO. REQ.											
NOTE	ITEM	TITLE	DRAWING & GR. OR IT.	MATERIAL SPEC.	EQUIV. SPEC.	OR. 1	OR. 2	OR. 3	OR. 4							
	1	CAPSULE END PLUG	674C472-IT.1			2	2	2	2							
	2	FLOW BLOCKER	674C472-IT.2			4	4	4	4							
	3	CAPSULE TUBE	674C472-IT.3			1	1	1	1							
	4	SCREW	674C472-IT.4			12	12	12	12							
	5	FUEL ROD CAN ASSEM.	881D852-Gr.1			1										
	6			-Gr.2		1										
	7			-Gr.3		1										
	8			-Gr.4		1										
	9			-Gr.5			1									
	10			-Gr.6			1									
	11			-Gr.7			1									
	12			-Gr.8			1									
	13			-Gr.9				1								
	14			-Gr.10				1								
	15			-Gr.11				1								
	16			-Gr.12				1								
	17			-Gr.13					1							
	18			-Gr.14					1							
	19			-Gr.15					1							
	20	FUEL ROD CAN ASSEM.	881D852-Gr.16						1							

FIGURE 2

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W. J. H. 3-15-63

6.4.25

 WESTINGHOUSE ELECTRIC CORPORATION
 INDICATES FULLY APPROVED
 ENGINE BACKGROUND OF PRINTS
 INDICATES PRELIMINARY DRAWING
 INDICATES DESIGN IN PROGRESS
 INDICATES DESIGN NOT FOR CONSTRUCTION
 INDICATES DESIGN NOT FOR CONSTRUCTION

NEXT ASSEMBLY 67473123

REF. ENG.

DFTM T. B. Wiese 1/1/63

CHKR T. B. Wiese 7/1/63

DES. ENG. R. DUNCAN 2/1/63

MFG. ENG.

MTLS. ENG.

APP. P. H. H. 2/1/63

APP.

DFTG. SUPV. W. J. H. 2-13-63

SCALE 2-1

DIMENSIONS IN INCHES

DO NOT SCALE

SUB # 2

 WESTINGHOUSE ELECTRIC CORPORATION
 ATOMIC POWER DIV. PITTSBURGH, PA., U.S.A.
 TITLE:
 PLUM BROOK REACTOR FACILITY
 IN-PILE IRRADIATION EXPER:
 CAPSULE ASSEM.
 674C473

FUEL ROD CAN No. 1-1
 FUEL PIN No. 1-1
 FUEL TYPE UO₂
 ENRICHMENT 8.5%
 ENRICHED PELLETT COL. LENGTH 3.7253"
 WEIGHT 45.2327g
 PIN CLAD WALL TH. .010
 PIN WEIGHT 78.8103g
 PIN ASSEMBLY VOLUME 11ml
 LIQUID METAL 22%Na(78%K)
 FUEL CAN ASSEM. WEIGHT 200.805g
 WITHOUT LIQUID METAL
 FUEL CAN ASSEM. WEIGHT 212.301g
 WITH LIQUID METAL
 LIQUID METAL VOL. ADDED 14.34cc
 LIQUID METAL WEIGHT 11.496g

POSITIONED IN
 CAPSULE No. 1

CAPSULE No. 1-1
 FUEL PIN & CAN No. 1-1
 (CAPSULE POSITION No.)
 TYPE OF FUEL
1 UO₂
2 UO₂
3 UO₂
4 UC

(9.546") OVERALL LENGTH
 (9.0408)

FUEL ROD WIRE No.	MAT'L	CONDITION
1-1-1	348	ANNEALED
1-1-2	348	ANNEALED
1-1-3	348	ANNEALED
CAPSULE No.		
PIN & CAN WIRE No.		

.6242"

.6242"

.6243"

.6243"

IDENT. No.

PRE-IRRADIATION FUEL
 ROD CAN DATA OF PBIF
 IN-PILE EXPERIMENT
 R. SCHREIBER 4-18-63 SK. No. 51

Figure 3

Figure 4

6.4.27

Figure A-I-1 Unperturbed Thermal Flux in LA-1. Bank Height 17.9", MUR.

Figure A-I-2 Unperturbed Thermal Flux in LA-1. Bank Height 17.9", MUR.

Figure A-I-3 Unperturbed Thermal Flux in LA-1. Bank Height 17.9", MUR.

VII. INSTRUMENTATION AND DATA REDUCTION

HEATERS FOR IRRADIATION EXPERIMENTS

Co-ordinator - N. H. Hancock

Abstract

This paper describes U.K.A.E.A. experience with electric heaters used in irradiation experiments and allied laboratory apparatus.

Resistance heaters of mineral-insulated cable form have been manufactured and developed to a high standard, and the further development to extend their working temperature up to 1200°C by the use of refractory and noble metals is detailed. A range of bare metal, ceramic and cermet heaters is being investigated for continuous operation in reactor experiments at about 1600°C. Laboratory tests have established useful data on high-temperature compatibility, grain growth, evaporation and oxidation rates in inert gas atmospheres. Design data and experience are presented for both induction and low-voltage resistance heating, these systems having been applied to a number of reactor engineering experiments. Heat transfer work involving surface heat fluxes in excess of 400 watts/cm² has required the development of a heat source more powerful and reliable than a resistance heater; some U.K.A.E.A. work on a thermionic device for this purpose is described.

Metallurgy Division,
U.K.A.E.A.,
Atomic Energy Research Establishment,
HARWELL.

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 - 2.3 M.I. Cables under 1.5 mm Diameter (Miniature) (S. L. Naylor and E.E.C. Hutchins, A.E.R.E., Harwell)
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7. Electron Bombardment Heating for Reactor Experiments (W. McAndrew, A.E.R.E., Harwell)

1. Introduction

For reactor irradiation experiments one of the important requirements is to maintain a pre-determined temperature in the specimens for the duration of the experiment. In most of these experiments the control of this temperature is by electrical heaters.

This paper describes the U.K.A.E.A. experience of the production, development and use of electric heaters for the control of temperatures in reactor irradiation experiments and in allied experimental work performed in the laboratories. The types of heaters are divisible approximately into two overlapping groups, namely:-

- (1) Those required to produce and/or maintain a high temperature, and
- (2) Those required to produce a high surface heat flux.

Group 1 types are in the majority and are designed for temperature control of fissile and non-fissile irradiation experiments; they may have relatively high surface heat ratings only as a consequence of their high power rating and restricted physical size.

Group 2 types are being developed for laboratory simulation of heat transfer conditions for a variety of fuel element configurations and irradiation capsule conditions.

The Mineral Insulated cable type of heater being a most convenient form for heater manufacture, and with the vast experience in the United Kingdom on the use of nickel-chromium alloys as heating elements, considerable work has been concentrated on developing this combination to a high standard. Many of the following contributions therefore describe in detail the development of manufacturing and testing techniques which have resulted in components of great reliability relative to their operating conditions.

Some work has also been done in the use of refractory materials to extend the temperature and rating of the mineral insulated cable type of heater to a range higher than that achieved by the use of more conventional materials. Bare wire, ceramic and non-metallic heaters offer some advantages for still higher temperature operation, but there are many problems to be solved before a satisfactory operating life can be guaranteed.

With the accent on still higher temperatures and ratings reactor engineering experiments on heat transfer or simulation work have demanded the development of other types or forms of electrical heaters and heat sources. This work has produced many new problems.

2. Mineral-Insulated Cables (M.I. Cables)

2.1 Introduction

Current irradiation rig requirements for heaters capable of operating for periods of six to twelve months in nominally inert atmospheres, and at temperatures of 1000°C upwards, have stimulated interest in the performance limitations of M.I. Cables.

Conventional cables having 80/20 nickel-chromium cores, magnesia insulation and either stainless steel or 'Inconel' sheaths have been studied intensively by the U.K.A.E.A.

Answers to the following questions were sought:

- (i) What is the maximum operating temperature for M.I.C. heaters consistent with long-term (8000 hours) reliable operation in commercial grade helium?
- (ii) What factors determine the temperature limit?
- (iii) Can the limit be raised by employing other materials in M.I.C. construction?

This section presents a summary of the results of the investigations.

Because of the urgent requirements of an Irradiation Programme in the BR.2 Reactor at Mol which required miniature heaters, laboratory experiments were confined to cables having O.D.s of approximately 1 mm (0.045 in.) and 1.5 mm (0.062 in.) with core diameters of approximately 0.38 mm (0.015 in.) and 0.5 mm (0.020 in.) respectively. Tests conducted on both cable and bare wire specimens demonstrated that satisfactory performance could be guaranteed only when the wire temperature was limited to 1050°C maximum. Substantial increase above this temperature was attended by a marked increase in the rate of grain growth embrittlement, invariably leading to premature failure preceded by the formation of highly localized hot spots. Operational life was limited to 300 to 500 hours.

Apart from grain size and growth rate, other factors most likely to influence the performance of miniature cables are:-

- (i) The degree of uniformity in the cross-sectional area of the core wire: where the wire diameter is small, slight local reductions in its value assume relatively great importance and are likely to accelerate the development of hot spots.
- (ii) The degree of concentricity of the core wire within the sheath: the smaller the radial thickness of magnesia insulation the more likely a departure from concentricity is to cause electrical breakdown.

Experience in the development of miniature heaters clearly indicates the importance of consistent close control over fabrication processes in order to secure consistency in the dimensions of the finished product.

- (iii) The density of compaction of the insulant: for a given sheath temperature and radial insulation thickness the maximum watts rating per unit length of cable consistent with limitation of the core temperature to 1050°C depends upon the thermal conductivity of the insulant. In turn, the thermal conductivity of magnesia compacted to more than 90% theoretical density is critically dependent on its density. The thermal conductivity increases by a factor of 4 approximately for a decrease of porosity from 12% to 5%. In this range local variations in compaction density therefore become important and result in relatively large temperature variations along the core wire.
- (iv) The resistivity and dielectric strength of the insulant: at 1000°C and with a radial thickness of 0.2 mm (0.008 in.) of insulant, failures due to electrical breakdown of the magnesia appear to be unlikely, even at 100 volts. Operating experience at 1000°C in a reactor environment is however too limited at the present time to permit a positive conclusion. No laboratory tests on cable specimens

have shown any evidence of failure due to deterioration of the insulant, even after several thousands of hours. It is worth noting, in this context, that cable specimens having a molybdenum core and MgO insulation within an 'Inconel 600' sheath have been operated continuously in the laboratory at 1200°C without any indication of approaching break-down of the insulant after 2,500 hours.

The dielectric strength of compressed MgO at 1000°C has been reported⁽¹⁾ to be approximately 50 V/mil* at 85% theoretical density, increasing to 60 V/mil at 90% theoretical density. The tests were performed in normal atmospheric air.

- (v) Compatibility of insulant with core and sheath: in laboratory tests, cable insulated with MgO of approximately 99.5% purity has given no failures attributable to incompatibility of the insulant with core or sheath materials. At any given temperatures the same order of useful operating life was obtained from 80/20 'Nichrome', irrespective of whether it was tested in bare wire form or in M.I.C. form, the mode of failure being the same in both cases. Also, subsequent metallographic examination revealed no positive evidence to suggest that a reaction between MgO and either 'Nichrome', stainless steel or 'Inconel 600' had occurred.
- (vi) Effect of impurities in helium: it is known that the avoidance of sulphur-bearing and reducing atmospheres is essential for the satisfactory performance of 80/20 'Nichrome'.

In some laboratory tests on bare wires in purified helium containing 300 ppm to 400 ppm H₂ + CO, the life of 80/20 'Nichrome' was drastically curtailed at all temperatures above 1000°C. It is apparent that any impurities in the helium should be of an oxidizing nature if premature failure of 'Nichrome' is to be averted.

In commercial helium at 1000°C to 1050°C oxidation of core and sheath materials does not appear to present a serious problem. In this respect the results of laboratory experiments have shown 'Inconel 600' to be superior to 18/8/1 stainless steel.

- (vii) Cables other than miniature: it has not so far been possible to conduct a systematic enquiry into the effect of the physical size of M.I. cables upon their useful operating life at any given temperature. However the incomplete results of tests on some heaters manufactured from 2.5 mm O.D. cables, having a core diameter of 1.1 mm (0.046 in), throw a little light on this question. They suggest that, given a sufficiently robust construction, a life of over 4000 hours may be possible with a core temperature in the range 1150°C to 1200°C.

2.2 M.I. Cables Over 1.5 mm Diameter

The principal use of heaters of this type, of 3 mm cable diameter, has been in connection with sodium-filled irradiation capsules, the first use being in 1958. Since that date some 300-400 heaters of the general type shown in Fig. 1 have performed satisfactorily at temperatures up to 700°C for periods of up to 13,000 hours.

*1 mil = 0.001 in.

Fig.1. 3kW COILED HEATER

FIG. 2. HEATER JOINT

Early experience showed that the three most likely modes of failure were:-

- (i) sheath defects, leading to sodium penetration into the cable and consequent insulation failure.
- (ii) seal weld defects, causing sodium leakage from the irradiation capsule,
- (iii) overheating at the heater/supply lead joint.

The first two difficulties were overcome by careful attention to manufacturing techniques and by stringent testing of the heaters, both before and during immersion in the sodium. The last problem was completely eliminated simply by making the ends of the heater cable of larger diameter than the working length, a factor of 2 being common. By this means the IR heating at the joint was reduced by a factor of 4 and a fully satisfactory joint could be made.

The majority of these heaters have been made to a U.K.A.E.A. design whose bases were:

- (i) maximum core diameter for maximum life and low voltage operation,
- (ii) maximum cable length to minimise power density and thus minimise the temperature gradient between core and sheath.

Subsequent work on insulation resistance of MgO and on the measured radial temperature gradients in these cables has verified the wisdom of these bases, which had led to the production of a heater cable 3 mm (0.125 in.) O.D. with a 1.45 mm (0.057 in.) diameter core. The sheath was 18/8/1 stainless steel (to facilitate argon arc welding) and, following existing commercial practice, the insulant was MgO. Initially the requirement was for a cable length of 122 cm (48 ins.) rated at 8.3 watts/cm. length but, with the great improvement in insulation resistance obtained by the use of high purity fused MgO, an increase in rating to 40 watts/cm. length has proved practicable. In both cases, a sheath temperature of 800°C was required and attained.

A somewhat less frequent use for cable of this size is in the form of a coil wound around and brazed onto the capsule it heats. In this format the ends of the heater and the power leads may not be larger than the work-in diameter. A joint (Fig. 2) originally developed at Harwell for 1.5 mm diameter cable and proven to be satisfactory over more than 10,000 hours operation may be used, but this requires some enlargement of the diameter. One major problem is the production of a fully annealed power lead having a stainless steel sheath (for use in CO₂ atmospheres) and MgO-insulated copper core. Differences in the annealing temperatures are considerable and have not yet been overcome.

A variant of the simple immersion heater is shown in Fig. 3. In these a metal former is circumferentially helically grooved, the cable being laid in the grooves. In some types (Fig. 3) the groove is subsequently filled by rolling metal from the former into it; in other types (Fig. 4) a sprayed metal is used for the filling. Nickel, copper and stainless steel have been used for the filling, and zirconium, 'Zircaloy 2', and stainless steel as the former. The fact that copper, when used as the filling in conjunction with 'Zircaloy 2' and stainless

Fig.3. ORIGINAL "COIL ON FORMER" HEATER

Fig.4. "COIL ON FORMER" HEATER WITH SPRAYED METAL INFILL

steel, can promote the formation of a ternary eutectic which melts between 600°C and 700°C has resulted in this combination being avoided.

Inevitable imperfections in contact between cable and former limit the ratings to 10 watts/cm. length and a former temperature of 950°C. In pile use has so far been limited to a former temperature of 900°C at about 8 watts/cm, with brief excursions to 900°C at 10 watts/cm. and at these ratings lives approaching 8000 hours have been obtained.

In all heaters of this type, two primary problems are the manufacture of a cable of the precise length necessary to match the length of helical groove, and the type and location of the end connections. On those heaters in which an enlarged cable end is used the first problem is only solved by careful hand fitting, and the second is normally solved either by packing pieces or deposition of relatively large amounts of sprayed metal.

A further construction method has been evolved by the U.K.A.E.A. which solves some of the problems that arise when M.I. cable is used as the heating medium. The heating element is produced by swaging from commercial 3.0 mm (0.125 in.) diameter, metal-sheathed, magnesia-insulated cable. Diameter is reduced by multiple swaging to a minimum of 1.5 mm over the region required to form the high temperature zone. A suitable length of cable at each end of the stock is maintained at the original diameter to provide connecting leads. Low-resistance connecting leads are formed at each of the exposed ends of the nickel-chromium conductors by brazing on copper wires and enclosing the exposed conductors in metal tubing insulated from the wire by preformed, crushable magnesia sleeves. The assembly is then compacted by swaging to form integral copper cored mineral insulated leads having an outside diameter of 3.75 mm. It is necessary to braze the overlapping joint between the sheath tubes in order to produce a gas-tight system.

The heating elements produced by the above method are applied in either helical or 'grid-iron' configuration to a tubular support or former. They are metal sprayed to ensure good mechanical and thermal bonding to the former and then machined to produce a uniform outline for incorporation in irradiation rigs (see Figs. 5 and 6). This method ensures that the highly rated cable section and the critical transition zones from swaged to unswaged cable are embedded in a good heat sink. Additionally, in the preferred 'grid-iron' configuration, a broad tolerance on the length of the swaged section can be absorbed and the risks arising from tight bends in the cable, in the region of the swaging shoulder are avoided, since the transition occurs in the end legs of the grid iron permitting the cable to emerge from the former in a straight line.

Several rigs under construction employ the techniques outlined, though as yet no in-pile experience has been obtained. A number of samples are under test in simulated rig conditions, and are giving satisfactory performance at specimen temperatures up to 840°C and power inputs up to 750 watts. A typical assembly failed under destructive test in open air when dissipating 2 kW at a heater former temperature of 1000°C, corresponding to a rating of 16 watts/cm. over the swaged cable length.

2.3 Cables under 1.5 mm Diameter (Miniature)

Development of small heaters using nickel-chromium-cored, mineral-insulated cable of approximately 1 mm diameter has been carried out by the U.K.A.E.A. over the past four years with a view to obtaining the maximum possible heat output consistent with a life of about 4000 hours. This work originated in support of rig designs for use in the BR.2 Reactor but it is

Figure 5 Latest Type of 'Coil on Former' Heater

Figure 6 "Grid-Iron" Heater Configuration

now proving of great use in other applications.

The main problems encountered when operating small diameter cable close to its limiting heat output are:

- (i) the temperature limitation on the core material due to grain growth, intergranular attack by trace impurities etc.,
- (ii) the uniformity required in core cross-sectional area to avoid hot spots forming and consequent premature failure,
- (iii) the corrosion of the core and sheath due to reaction with impurities in the insulant or the surroundings at high temperatures.

The most important single parameter is the core temperature, and once the cable materials have been selected the problem resolves into determining the maximum allowable core temperature when in the cable form and relating this to the heat output, i.e. the heater rating and the heater matrix temperature. It is obviously highly desirable to put the cable into intimate contact with the heater matrix and in larger sizes this has been accomplished by rolling the cable into grooves, swaging on sleeves, etc. However, with miniature cable the small radial thickness of the sheath and insulation prohibits these methods, which involve considerable deformation. Other methods of ensuring good heat transfer from cable to matrix are metal-spraying into grooves or casting the cable in a metallic matrix. These methods however sometimes make the manufacture of the heater-specimen pack assembly much more complicated and expensive and also make recovery of the specimens after irradiation much more difficult. For these reasons the simpler method of fitting the cable into grooves with an outer retaining sleeve was investigated.

A typical heater assembly is illustrated in Fig. 7. The 'hairpin' winding was dictated by the space required for specimens in the heater matrix and for ease of recovering the active specimens remotely. The butt-swaged arrangement using a continuous nickel-chromium core with the larger diameter cable at the two 'tails' was adopted to avoid introducing a joint in the hot region with its associated problems of core inhomogeneity and poor insulant packing. The original cable diameter chosen was 1.0 mm in the hot zone with 1.50 mm diameter 'tails', but difficulties in swaging to 1.0 mm resulted in the final adoption of 1.14 mm and 1.65 mm as minimum diameters. The required heat output of this heater was 15 watt per cm. length of cable at temperatures of 350°C, 700°C, 900°C, and 1000°C.

Experimental and development work was carried out in order to

- (i) determine the maximum operating core temperature of the commercially available 80% Ni 20% Cr alloys for a life of 4000 hours when they are used in the M.I. cable form,
- (ii) determine the thermal conductivity of the MgO insulant in the M.I. cable operating at maximum core temperature,
- (iii) develop the technique for butt-swaging cable below 1.25 mm diameter,
- (iv) develop the technique for forming the hairpin winding without damaging the cable,

FIG. 7.
B.R.2. METALLOGRAPHY IRRADIATION RIG

FIG.8. MAX. RECOMMENDED POWER RATING FOR 4000 HOURS LIFE
.045" O.D. .015" CORE DIA. (NICHROME V)

- (v) test heater assemblies in conditions closely duplicating those in rigs, using the same type of electrical supply, to determine burn-out ratings and heater life at reduced rating,
- (vi) determine the heater 'tail' temperatures in helium and neon atmospheres.

2.3.1 Tests on Bare Wires to determine maximum operating core temperature.

A series of tests was conducted in commercial helium on bare 80/20 'Nichrome' wires of 0.355 mm (0.014 in.) diameter with the object of relating their useful life to operating temperature over the range 1000°C to 1250°C.

It was found that in order to ensure an operating life of 4000 hours or more, it was essential to limit the wire temperature to a maximum of 1050°C. Between 1050°C and 1100°C, satisfactory performance could not be guaranteed, the chance of premature failure increasing rapidly with increasing temperature. Over the temperature range 1100°C to 1250°C, the operating life was invariably limited to 300 to 500 hours.

In every case, failure was preceded by the appearance of highly localized hot spots, and was primarily associated with a high rate of grain growth, with subsequent severe embrittlement of the wire.

The evidence suggested the existence of a critical temperature between 1040°C and 1075°C, operation above which would lead to greatly accelerated grain growth. At the former temperature, specimens showed no serious embrittlement after 5000 hours operation. On raising the temperature to 1075°C, however, very severe embrittlement developed within a further 400 hours of operation.

A further series of tests on 80/20 'Nichrome' wires of similar diameter incorporated into M.I. cables illustrated that their performance limitations were not much influenced by intimate contact with the MgO insulation. The correlation between useful operating life and core operating temperature closely followed that obtained when wires were tested in bare form.

It has not so far been possible to conduct a systematic investigation into the influence of wire diameter on useful operating life at any given temperature. Other things being equal, an increase in wire diameter should be followed by an increase in endurance because the effect of grain growth would be less catastrophic. This contention is supported by the results of experiments with heaters made from 2.5 mm O.D. cable having a core diameter of 1.17 mm (0.046 in.). In some cases, operational lives measured in several thousands of hours have been obtained, with core temperatures approaching 1200°C.

2.3.2 Experimental Determination of MgO Thermal Conductivity.

The chief difficulty encountered in determining the thermal conductivity of the insulant in M.I. cables, is associated with the accurate measurement of the core wire temperature. The technique adopted consisted simply in drilling a 0.5 mm diameter hole through the sheath and insulant at the mid-point of a 15 cm. length of miniature cable test specimen. The apparent temperature of the minute area of the core surface thus exposed could then be monitored with a precision optical pyrometer of the disappearing filament type. The apparent temperature of the sheath surface was measured with the same instrument. Owing to end losses and convection currents, the directly

heated, vertically mounted test specimens exhibited an axial temperature gradient with maximum values occurring in a zone extending approximately 12 mm above and below the axial centre. This zone exhibited good temperature uniformity, and all sheath surface measurements were taken at a point 6 mm above the core sighting hole.

Precision ammeters and voltmeters were employed for monitoring the 'Variac' controlled power input to the specimen. The total input was divided by the length, 15 cm., to obtain the power dissipation/cm length. Although a temperature gradient was present along the test length, this was considered to be a valid procedure in view of the extremely low temperature co-efficient of resistance of 80/20 nickel-chromium.

In order to minimise the effects of mains voltage fluctuations and to facilitate fine control of specimen temperatures, a voltage stabilizer unit and a suitable step-down transformer were incorporated in the electrical supply circuit. Also, the specimen was enclosed in a silica glass tube in order to exclude random air currents.

Using the empirically derived data and the known dimensions of the cable, the thermal conductivity (k) of the compacted magnesia was calculated from the relationship:-

$$k = \frac{Q \log D/d}{2\pi(T-t)}$$

where : Q = watts rating/cm length of specimen
D = inner diameter of sheath
d = diameter of core wire
T = temperature of core °C
t = temperature of sheath °C

Typical values of k derived using this method ranged from 0.022 to 0.023 watt/cm°C.

The difference between the unexposed core temperature and the core temperature as measured through the sighting hole was probably the major source of uncertainty in this method. The latter value could be expected to be slightly larger than the former, because of the less favourable heat transfer conditions existing due to partial exposure of the wire surface. Its use for calculation purposes however could be justified on the ground that it would give a worst figure for the thermal conductivity of the MgO.

Another possible source of inaccuracy could be the difference in emissivity between exposed core and sheath surfaces. In practice no temperature measurements were made until the surfaces were fully oxidized. Their emissivities were then assumed to be practically identical.

At a later date the technique was modified with the aim of eliminating its major uncertainties⁽²⁾.

2.3.3 Simulation Testing

Simulation tests were performed at temperatures of 700°C, 900°C and 1000°C. The first tests were to determine the short-term burn-out power and the mode of failure, e.g. within the matrix, in the 'tails', at the hairpin bends, etc. The results are shown in Table 1, and it can be seen

immediately that operation of these heaters at 15 W/cm for an intended life at 900°C and 1000°C is not possible.

Table 1 - Heater Burn-Out Rating

<u>Heater Matrix Temp.</u>	<u>Burn-out Rating</u>	<u>Mode of Failure</u>
°C	watt/cm	
700	38.9	} Melt-out of core and sheath within heater matrix progressing up the tails
900	21.0	
1000	23.4	
1000	23.6	

Assuming that failure occurred by melting of the core at 1400°C and that the temperature difference between matrix and core is proportional to heater power, one can deduce maximum allowable heater ratings for each temperature, based on a core temperature of 1050°C. These are 3 W/cm at 1000°C, 8.5 W/cm at 900°C and 20 W/cm at 700°C.

Life tests were then run at ratings considered useful for BR.2 rig heaters with results shown in Table 2.

Table 2 - Life Tests

<u>Heater Matrix Temp.</u>	<u>Cable power</u>	<u>Estimated Core temperature</u>	<u>Life to Failure</u>
°C	w/cm	°C	hours
700*	18	1010	1330
700*	21	1070	470
900	9	1060	1440
900	11.5	1100	585
900	12	1110	900
1000	12.7	1220	5

*'Tail' diameter increased to 1.80 mm (0.071 in.) for these heaters. These results in fact confirm that these heaters can only be operated at relatively low power at 900°C and 1000°C. It would appear that at 700°C a power rating of 15 watts/cm would be the maximum allowable. The mode of failure at 900°C and 1000°C was melting of the core and sheath at the tails and, in one case, within the heater matrix.

At 700°C in both tests the cable melted out at the swaged section where it emerged from the matrix, even though great care was taken to avoid any exposed length of the smaller diameter portion of cable. This emphasises the attention that must be given the fitting of the cable to avoid hot regions forming.

Recommended maximum heater ratings for various temperatures are shown in Fig. 8.

Short lengths of cable of 1.62 mm (0.064 in.) diameter to 1.80 mm (0.071 in.) diameter were operated in glass cells in atmospheres of helium and neon in turn and the sheath temperature measured with an optical pyrometer. The results are shown in Figs. 9A and 9B. The effect of increasing tail diameter is beneficial in two ways, the temperature of the cable is reduced for the same power per cm length and also the power dissipated in the tail is reduced for a fixed heater power. Assuming that 900°C is acceptable for the sheath of the tail cable it appears that 1.80 mm diameter would allow operation of the heating section (with 0.38 mm diameter core) up to approximately 20 w/cm even in 100% neon atmosphere.

2.3.4 Effect of improving Heat Transfer from Cable to Matrix

If one considers the ideal case of the cable sheath in such intimate contact with the matrix that the two are at the same temperature, and taking k for MgO as 0.020 W/cm°C, the temperature difference between core and matrix for these cables is calculated to be 7°C per watt/cm. Assuming once more the allowable core temperature to be 1050°C the allowable heater power ratings become:-

at 700°C, 50 watt/cm.
at 900°C, 20 watt/cm.
at 1000°C, 7 watt/cm.

Thus it would appear that even under ideal conditions the heater power is severely limited at 1000°C. In order to attain the maximum power output at 700°C much longer 'tails', of approximately 3 mm diameter would be required.

2.4 Highly-Rated Cable Heaters

A large programme of laboratory experiments is being performed by the U.K.A.E.A. to obtain more thermal data on sodium-filled irradiation capsules. In these experiments the specimen is represented by a simulation heater which is required to operate in sodium for short periods of time at a high surface rating (200-400 watts/cm²). The success that has been achieved so far in obtaining important data on highly rated capsules has been due mainly to the development of a simulation heater of adequate performance. A diagram of the heater assembly in a capsule is shown at Fig. 10, and the significant heater data obtained during development and subsequent operation in a capsule is given in the following text.

The simulation heater is made by close-coiling a M.I. heater cable on a stainless steel former to reproduce the outside dimensions of a given specimen. The cable for the coil is butt-swaged to between 1.0 mm and 1.5 mm diameter from a larger-diameter cable; this increases both the packing density of the insulation and the core-wire resistance. After coiling it is essential that the butts are positioned close to the coil so that both are always below the cold sodium level in the capsule. This guarantees adequate cooling when at power and ensures that only the heater leads operate in the inert gas above the sodium level.

Since the heater operates in sodium, the cable is inspected for sheath integrity at each stage of manufacture and assembly. Before final assembly the coil is soak-tested in 17% HNO₃ solution for crack-detection; the acid also removes fine surface oxides. Experience has shown that the insulation

FIG. 9A. CABLE TEST (WATTS/CM L-T) TESTS CONDUCTED IN NEON

FIG. 9B. CABLE TEST (WATTS/CM L-T) TESTS CONDUCTED IN HELIUM

SIMULATION FUEL CAPSULE
FIG. 10.

FIG. 11. GENERAL CONFIGURATION M.I. HEATER CABLES.

resistance must not be less than 50 Megohms (500V Megger test) after the heater leads are brazed to the top cap of the capsule on final assembly.

Ratings

The significant heater ratings that have been achieved so far are given in Table 3.

Table 3

Heater Identity	Dia. of cable mm	Dia. of cable (leads) mm	Dimm. of coils mm	Resistance ohms	Max. Power (before failure) K Watts	Watts/cm lg. of coil cable	Peak Sodium temp. °C	Simulated Specimen surface rating Watts/cm ²
769 Simulation	1.0	2.0	9.52 dia. x 46.7 long	15	3	24	600	210
706	1.57	5.1	6.98 dia. x 66.7 long	2	5	78	700	340

From the experience gained in running the 3 KW test a more robust heater was developed for the 5 KW test. This heater was designed to have much larger diameter cable ends in order to obtain a more reliable connection at the higher rating. At the 5 KW rating the leads were still running cool, and failure eventually occurred by coil 'burn-out' in the sodium. The reason for this failure is still being investigated, since it was not expected that sodium film boiling would occur at a simulated specimen surface rating of 340 watts/cm². A further test at this rating is now being prepared to verify the results of the 5 KW test.

2.5 High Temperature Cable Heaters

Coaxial heaters having a nickel-chromium core, stainless steel sheath and a ceramic powder insulant have been extensively used in irradiation rigs, but increased chemical activity of the materials at about 1050°C shortens the element life, particularly when small core diameters are used. However, this type of heater is a convenient method of heating specimen matrices, and development has been started to produce a heater capable of operating with a core wire temperature of 1200°C. The required properties of the cable materials are:-

- For the core:
- (a) a melting point well above 1200°C
 - (b) a suitable resistance temperature characteristic
 - (c) ductility, to permit use of known manufacturing methods;
- For the sheath
- (d) a melting point in excess of 1200°C
 - (e) ductility, as for (c)
 - (f) adequate long-term corrosion resistance in its external environment;
- For the insulant:
- (g) adequate electrical strength under specified temperature and voltage,
 - (h) long-term compatibility with core and sheath
 - (j) high thermal conductivity, to minimise the core operating temperature.

In order to determine suitable materials, testing was commenced over a range of refractory metals for the core, and platinum-10% rhodium alloy was finally chosen. The properties of 'Inconel 600' suited the requirements for the sheath (a platinum-rhodium sheath would be better but much more expensive). Various ceramic insulants were considered and the final choice was of two types of high-purity MgO , 99.97% or 99.6%.

Compatibility testing is in progress on samples of these materials, to determine their suitability for 5000 hours' operation at $1200^{\circ}\text{C} \pm 50^{\circ}\text{C}$. Work has been carried out in the following stages:-

- (a) treatment of magnesia insulant in the as-received condition; determination of impurity levels
- (b) manufacturing techniques for producing lengths of coaxial heater cable
- (c) compatibility testing at temperature
- (d) determination of power versus temperature characteristics
- (e) production of the 1200°C heater.

An X-ray analysis of the commercial 99.97% and 99.6% purity magnesias gave actual purity figures of 99.79% and 99.57% respectively. This, together with observations on nickel-chromium-cored heater cables, indicates that impurities may not be dispersed homogeneously throughout the magnesia; the formation of hot spots may therefore be due to local high impurity concentrations. Reheating the magnesia to 1000°C before use was found beneficial in delaying reabsorption of water vapour from the atmosphere.

Small samples of cable were made up for rolling and swaging. Insulant densities of 3 gm/cm^3 were obtained from a starting density of 2.2 gm/cm^3 in the initial work, and development of techniques continues with a view to obtaining higher densities.

Forty-eight specimens (24 with each insulant) have been swaged from 8.0 mm (0.315 in.) starting diameter to 6.1 mm (0.24 in.) and subjected to compatibility tests. Samples containing each insulant will be operated at 1150°C, 1200°C, and 1250°C for periods of 200, 1000 and 5000 hours. The samples will be examined for chemical reaction between core, sheath and insulant. After 200 hours there is no evidence of any attack, and further information will be available at the time of this symposium.

Some attack on the thermocouples of the furnace was observed during the early tests; this was attributed to SiO₂ vapour from the refractory oxide tubes in the furnace. An 'Inconel' liner was fitted to prevent this contamination in later tests.

The preliminary work on this development indicates that after 200 hours at 1250°C there is no marked incompatibility of the materials.

2.6 The Manufacture and Inspection of M.I. Heater Cables

When operated under extreme conditions of temperature and power rating (up to 1000°C and 20 watts/cm length) commercially available mineral insulated heater cable has a high failure rate. Investigations have shown that if the manufacture and inspections procedures could be closely controlled this failure rate could be reduced. To obtain heater cable to the very high standard required for use in irradiation experiments, it was agreed to produce cable within the U.K.A.E.A. where the manufacturing processes could be closely supervised.

In order to simplify development and manufacture, it was decided to standardise heater design on four cable sizes of the geometry shown in Fig. 11; their dimensions are given in Table 4.

Table 4

Cable Diameter		Sheath Thickness		Conductor Diam.		Insulant Thickness	
inches	mm	inches	mm	inches	mm	inches	mm
0.045	1.14	0.006	0.162	0.015	0.381	0.0090	0.228
0.0625	1.59	0.0085	0.216	0.026	0.660	0.0097	0.246
0.080	2.03	0.0135	0.343	0.034	0.863	0.0095	0.241
0.100	2.54	0.017	0.432	0.050	1.270	0.0080	0.203

Chemical analysis and the behaviour of test heaters in laboratory work showed that the materials' compositions varied widely both from batch to batch and within individual pieces of stock. Evaluation of a range of resistance materials, insulants and sheath metals resulted in the following choices:-

- (i) Resistance element of British Driver-Harris 'Nichrome V', for its small grain size.
- (ii) Insulation of G.E.C. Grade 12718 magnesia, which contains 99.6% MgO, 0.15% SiO₂, 0.15% CaO, 0.03% Fe₂O₃ and 20 ppm B.
- (iii) Sheath of 'Inconel 600'; corrosion resistance is better than that of 18/8/1 stainless steel.

2.6.1 Cable Manufacture

M.I. cables are normally produced by swaging, drawing, or a combination of these processes. In general, drawn cables have superior geometry but lower density, and thus a lower thermal conductivity, of insulant. The latter factor is of prime importance in irradiation rigs, since reliability of the heater depends on the resistance wire operating temperature being as low as possible. Butt-swaging was therefore adopted as the manufacturing technique; in addition it lends itself to the production of cables with large-diameter ends, which allows end connections of low electrical resistance and therefore increased reliability. A 2:1 ratio of diameters is usual in U.K.A.E.A. designs.

For a high-quality finished heater the following points must be ensured in the start cable and in the swaging operations:-

- (i) cleanliness of components
- (ii) dryness of insulant
- (iii) straightness and concentricity of conductor and sheath
- (iv) uniform density of insulant to give a uniform final cross-section of core
- (v) sufficiently high initial density of insulant to prevent the conductor moving during the first swaging operations,
- (vi) thorough annealing of the cable between swaging passes (Normally 10 minutes at 1050°C in argon)
- (vii) sufficiently low feed rate (about 60 cm/minute) and diametral reduction per pass (about 10%) to prevent distortion of the cable and cracking of the sheath.

The starting cable dimensions are calculated from the known required finished cable dimensions, with an allowance for the change in volume of the insulant.

Experimental results, presented graphically in Fig. 12, show that maximum compaction of the insulant is achieved after 6:1 diametral reduction, the change in density being 9%.

A difficulty arises in the manufacture of cables to close tolerances on length, namely that any error in the starting length is magnified in the final product due to the square-law relation between ratio of starting and final lengths and diameters, viz:

$$\frac{\pi}{4} d_1^2 l_1 = \frac{\pi}{4} d_2^2 l_2$$

or

$$\frac{l_2}{l_1} = \left(\frac{d_1}{d_2}\right)^2$$

In practice measured errors do not agree with calculated errors, and preliminary investigations indicate that there may be alternate compaction and regranulation of the insulant during the swaging operations.

FIG. 12. INSULATION THICKNESS AND CABLE DIAMETER FOR M. I. CABLES.

FIG. 13. TYPE I TUBULAR HEATER.

2.6.2 Cable Inspection

Commercial inspection procedures were regarded as inadequate for achievement of the required high quality of heater for irradiation experiments, therefore more stringent tests and standards were devised⁽³⁾.

Cables are inspected at intermediate and final stages of manufacture; the applied tests and standards of acceptance are outlined below.

Direct measurements of the conductor dimensions cannot be made without destroying the cable, therefore a piece of cable adjacent to the workpiece is examined as follows:-

- (i) Direct measurement of diameter; the periphery must fall between two circles whose diameters are the nominal diameter plus and minus 10%.
- (ii) Direct measurement of cross-sectional area; a variations of more than 10% from nominal is unacceptable.
- (iii) Measurement of electrical resistance; this must be within 2½% of the average value for the batch.

In addition the workpiece itself is subjected to radiography in two planes to show variations in core diameter; any substantial variation causes the cable to be rejected.

The cable is also thermal-cycled ten times between 200°C and 800°C in an inert atmosphere; observation of hot spots causes rejection.

From the inspection point of view the insulation thickness in itself is not important, but should be of minimum value consistent with electrical strength, in order to minimise radial thermal resistance.

Acceptance levels of insulation resistance and dielectric strength have been derived from tests on vacuum-dried cables. Production cables are vacuum-dried and are accepted providing that:-

- (i) insulation resistance is greater than 10^{11} ohms (cables over 1 mm diameter and less than 360 cm. long).
- (ii) the insulant withstands an applied D.C. voltage equivalent to 12,000 volts/mm (300 volts/mil). It has been thought that the insulation properties would be permanently impaired by this test, but no evidence has been found to support this view.

The sheath of the cable is checked for cracks by immersion (except for its open ends) in water containing a wetting agent; penetration is detected by a severe drop in insulation resistance of the magnesia.

Ductility of the sheath is checked by forming six turns of a sample length of cable round a mandrel $2\frac{1}{2}$ times the cable diameter, after which the above crack-detection test is applied.

The inspection procedures described above resulted in a high rejection rate but produced a considerable improvement in rig reliability and performance. Analysis of 19 rigs irradiated in the BR.2 reactor showed that the U.K.A.E.A. manufacturing and inspection methods resulted in a 30% reduction of heater failures and an increase of 50% average irradiation time.

BR.2 irradiation assemblies are similar to that shown in Fig. 7. Nominal temperatures are from 100°C to 1200°C, and heater power may be up to 1 kW with a 1.14 mm. (0.045 in.) diameter heater cable.

2.7 Alternative Materials for M.I. Cable Construction

This section deals with the development of M.I.C. heaters capable of operation in the 1200°C temperature range in irradiation rigs, in nominally inert atmospheres for periods of six to twelve months.

Laboratory tests on conventional cables having 80/20 nickel-chromium cores, magnesia insulation and sheaths of stainless steel or 'Inconel' have demonstrated that these are unsuitable for use under the specified conditions of service. Hot spots develop after 200-300 hours of continuous operation.

Experiments with bare wires produced similar results, even when conducted in an atmosphere of highly purified helium. It was therefore concluded that the replacement of the nickel-chromium element by an alternative material was an essential first step in the development of a M.I. cable for use in the high temperature range.

2.7.1 Selection of Materials for Preliminary Testing

An initial survey of metallic elements and alloys with sufficiently high melting points (1500°C upward) indicated that Re, Mo, Zr, Pt, Ti, W/Re, Pt/Rh and Pt/Ru alloys merited preliminary investigation as heater element materials. Other materials were rejected on one or more of the following grounds.

- (i) Insufficient ductility at normal temperatures to permit moderate amounts of mechanical working (W, Os, Rh, Cr).
- (ii) Extreme sensitivity of mechanical properties to minute concentrations of certain impurities in the process gas (Ta, Nb, Va are readily embrittled).
- (iii) Very high thermal neutron capture cross-section (Ir, B, Hf, Rh).

The selected materials were tested initially in bare wire form in order to determine their behaviour at 1300°C in contact with gaseous impurities in the helium atmosphere. A standard wire diameter of 0.5 mm (0.020 in.) was adopted for test purposes. Preliminary tests demonstrated that:

- (1) Zr and Ti are unsuitable for the specified application due to their very rapid rate of grain growth and consequent glass-hard embrittlement.
- (2) Mo will fail, even in an atmosphere of purified helium, apparently by catastrophic oxidation. The O₂ content of the helium was certainly below 50 ppm, but the free gas volume in the test cell used was about 1 litre. Hence the total mass of O₂ available for reaction was probably significantly large (the weight of the specimen was about 0.3 gm).
- (3) Because of the high volatility of its oxide, rhenium must be operated in purified helium atmosphere. Its performance under this condition was highly satisfactory. No symptoms of approaching failure or significant deterioration were evident after 9 months exposure, and its ductility after test tended to be superior to that of the wire in the as-received condition.

- (4) W-26Re was found to have a life of 2000 to 3000 hours when operated in a commercial helium atmosphere. Failure was associated with the formation of a surface oxide film about 0.12 mm (.005") thick and with excessive grain growth of the unoxidized metal core, leading to brittle fracture.

In a purified helium atmosphere however no failure symptoms became apparent even after test periods well in excess of 6 months. Some grain enlargement occurred but the loss of ductility is insufficient to warrant rejection of this material since it can withstand severe vibration and thermal cycling.

- (5) Pt-10Rh and Pt-8Ru alloy wires will operate in commercial helium for periods of 1 year. Some increase in resistivity occurs, together with pronounced recrystallization, but ductility is retained and loss of material through evaporation is negligible.
- (6) In the light of the above results, Zr and Ti were eliminated from the list of alternative heater element materials. Mo was not rejected because of the possibility that, when incorporated in a M.I. cable, the insulant might provide adequate protection against oxidation, due to the preferential filling of the insulant pores by helium and hydrogen.

Mo, together with Re, W-26Re and Pt-10Rh, was therefore selected for further testing as below.

Wire/insulant compatibility tests were performed in which closely fitting MgO sleeves of 99.5% minimum purity were threaded on to 0.50 mm (0.020 in.) diameter wires of each selected material. A wire operating temperature of $1300^{\circ}\text{C} \pm 25^{\circ}\text{C}$ was maintained with an atmosphere of purified helium in all cases except that of the Pt-10Rh specimens, for which commercial helium was used.

For ease of fabrication and achievement of high compaction densities, MgO is the preferred insulant for M.I. cables. Thus it was the first choice for this series of experiments, the results of which are summarized below.

Molybdenum

As in the case of bare wire specimens, failure occurred within 100 hours of start-up with symptoms of catastrophic oxidation. There was however no evidence of a Mo/MgO reaction.

Rhenium

No clear evidence of interaction between Re and MgO was observed after a test period of over 7 months duration.

W-26Re

Tests in contact with MgO have now been in operation for nearly 4 months (as at 9.12.65). No adverse effects have yet become apparent.

Pt-10Rh

Hot zones appeared after approximately 300 hours continuous operation. A similar result was obtained when Al_2O_3 sleeves (of 99.65% minimum purity) were substituted for the MgO sleeves. In both cases pronounced embrittlement was observed in the failure regions and metallographic examination

showed that grain boundary attack had occurred. The presence of silicon and iron was also detected by electron microprobe analysis.

Since both insulants contained appreciable quantities of SiO_2 and Fe_2O_3 the premature failure of the Pt-10Rh wires may reasonably be attributed to attack by the reduction products of these oxides. In order to guard against this, it is normal to operate Pt alloy elements in an oxidizing atmosphere.

It is apparent that if Pt alloy-cored, M.I. cables are to perform reliably under the specified conditions of service, the use of specially purified insulants is obligatory.

The performance of Re and W-26Re in the above tests indicated the desirability of investigating their behaviour when embodied in a metallic sheathed M.I. cable, and these tests are now in hand.

Of the range of materials investigated, Mo, Re and W-26Re at present show most promise for use as resistance elements. Since these materials do not perform satisfactorily in commercial grade helium, heaters in which they are embodied would need to be operated as sealed units.

Platinum alloys are far superior to Mo, Re and W-26Re for service in commercial helium atmospheres, but require the use of electrical insulants of ultra-high purity.

A limited amount of work has been performed with crushable 'Alundum 21' sleeves with the object of reducing the iron and silicon content. Cable specimens incorporating treated material have recently been placed on test and are still performing satisfactorily after 500 hours operation (as at 14.12.65). Analysis figures for the treated material are awaited.

Given this condition an alloy of Pt with Rh or Ru would be a good first choice as an alternative to 'Nichrome', on the following grounds:

- (i) Unlike Mo, Re or W-26Re, such alloys can sustain large amounts of cold working without great detriment to their mechanical condition. This simplifies the problems of cable fabrication.
- (ii) Pt alloys have excellent oxidation resistance.
- (iii) Their coefficients of linear thermal expansion match that of 'Inconel 600' more closely than those of Re, Mo or W-26Re.
- (iv) Unlike Mo or W-26Re they do not embrittle on recrystallization.
- (v) On nuclear grounds Pt-Ru alloys are preferable to Re or W-26Re, although inferior to Mo.

3. Mineral-Insulated Tubular Heaters

This section covers the design and development of M.I. (mineral-insulated) tubular heaters suitable for continuous long-term operation (12 months) at temperatures up to 1000°C in an inert atmosphere.

Design Considerations

Irradiation experiment heaters must satisfy the following conditions:-

- (a) The radial thickness must be small to minimize radial thermal resistance and neutron flux depression and permit maximum specimen size in irradiation facilities of restricted diameter.
- (b) The temperature difference between the specimen and heater element must be as small as possible.
- (c) The heater element resistance must give an acceptable voltage/current ratio. It is generally desirable to limit the voltage to 110 volts. The current must also be limited to 50 amperes because of the limited space available for power leads.
- (d) The heater has to be of robust design and be capable of continuous operation at full temperature for up to 9000 hours.
- (e) Electrical terminations require to be cool in operation and located at one end of the heater for ease of assembly of the irradiation experiment.
- (f) A further desirable, but not essential, feature is that the body of the heater should form a primary containment for the specimen.
- (g) The heater has to be suitable for manufacture in a variety of sizes and materials as demanded by service conditions.

Consideration of the above points resulted in the Type I heater design, (Fig. 13). The heater consists of a tubular element which is slotted to produce an extended electrical path of the required resistance. This is contained in and insulated from inner and outer shells, the inner shell forming the specimen containment. The total radial thickness of the heating section is 2.03 mm (0.080 in.). At one end of the heater the element is locally thickened to form the terminations. This design satisfies the conditions previously stated. However, to obtain the minimum radial thermal resistance a modification to the heater geometry was made, resulting in the Type II heater (Fig. 14). In this design approximately 30% of the heater element and its surrounding insulant have been removed and replaced with material from which the heater body is formed. Thus the radial thermal conductivity has been considerably improved (by a factor of 5 on a magnesia-insulated heater), but to offset this the maximum operating temperature of the heater is reduced owing to the loss in heating element area. The amount of this reduction is dependent on the power output of the heater. The two types of heater are complementary and in general the Type I heater is suitable for irradiations where maximum temperature and high electrical heat output are required. The Type II heater is employed where specimens having a high specific heat output are being irradiated.

Experience gained from the use of commercial mineral-insulated cable indicated that the required operating temperature of 1000°C could be obtained by using 'Inconel 600' for the inner and outer shells of the heater, 80/20 nickel-chromium resistance material for the element, and compacted, high-purity, fused magnesia grain for the insulant. A typical analysis of an insulant which has been found suitable for this service is that quoted in paragraph 2.6, (ii).

The heater development and proving programme was in two sections:-

FIG. 14. FULLY ENCLOSED TYPE II CONCENTRIC TUBE HEATER.

FIG. 15.

(1) Assessment of the Thermal Characteristics

Longitudinal Temperature Gradients

The graph in Fig. 15 shows typical temperature gradients for this type of heater. It can be seen that the usable hot zone of the heater occupies approximately 50% of the total heater length.

Radial Thermal Conductivity

This was determined by the insertion of an auxiliary heater inside the tubular heater. The radial temperature difference was measured by thermocouples situated on the inner and outer surfaces. From this, and the heater power input, the composite thermal conductivity was calculated from the conductivity equation. The values obtained were:

$$\text{Type I Heater} = 0.030 \text{ watts/cm}^{\circ}\text{C}$$

$$\text{Type II Heater} = 0.150 \text{ watts/cm}^{\circ}\text{C}$$

The Temperature Difference between Element and Internal Body

This was determined by drilling a small hole 0.5 mm diameter through the outer sheath to expose a small area of the heater element. Power was applied to the heater element and by means of a precision optical pyrometer the temperatures of the element and the internal heater surface were recorded.

Type I Heater

As this design approximates closely to a tubular element, the temperature differences introduced are very small and can usually be ignored.

Type II Heater

The element area on this design has been considerably reduced and the following empirical relationship has been found to exist:

$$T_H = T_E - \frac{0.60W}{E_E L}$$

Where T_H = internal temperature of the heater body, $^{\circ}\text{C}$

T_E = temperature of the heater element, $^{\circ}\text{C}$

W = power input, Watts

E_E = diameter of heater element, cm.

L = heater length, cm.

(2) Laboratory Life Tests

Test Equipment

The heater testing was carried out in cells of the type illustrated in Fig. 16. The heater was powered from a variable-voltage A.C. supply entering the cell through hermetically-sealed terminals. Three chromel-alumel

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FIG. 16. HEATER TEST CELL

FIG. 17. TYPE III CONCENTRIC TUBE HEATER.

thermocouples were positioned within the heater to indicate the temperature at points along the heater length, the temperature being monitored by means of switched moving-coil temperature indicators.

Before the test run the heaters were vacuum-dried (vacuum better than 10^{-3} torr) at 200°C for 24 hours. Dry helium was admitted at atmospheric pressure, and an insulation test was immediately carried out using a 500 volt 'Megger'. The acceptable insulation resistance was defined as 100 Megohms. The conductor resistance was measured on a Wheatstone Bridge.

The heaters were then set up in the test cells surrounded by an aluminium sleeve, which transferred the heat to the cooling jacket. To give the required power and temperature characteristics a predetermined gas annulus between the heater surface and the aluminium liner was incorporated. Under vacuum, sufficient power was supplied to the heater to bring it to the required operating temperature; it was maintained at this temperature until outgassing ceased. Dry helium was then admitted to a pressure of 0.14 Atm. and the cell sealed off. The power to the heater was then increased to return the heater to its operating temperature. To simulate reactor operating conditions more accurately the heaters were switched off for 5 minutes every 7 days. Temperature readings were recorded daily for the first 5 days and thereafter at intervals of 1 month.

3.1 In-pile Experience

At the time of writing, operating operational experience of tubular heaters in reactor experiments is limited to four cases, all of the same operating conditions. Each heater dissipated 2 kW at $550^{\circ}\text{--}600^{\circ}\text{C}$ for 22 days in PLUTO reactor, and showed no signs of failure; the thermal neutron flux in these reactor positions was approximately 4×10^{13} neutrons/cm².sec.

3.2 Non-standard Heater Developments - Type III Heater

The heater illustrated in Fig. 17 was designed to accommodate three cylindrical specimens in an irradiation facility of very restricted diameter, as in the BR.2 reactor. It can be seen from the illustration that the heater element is split into three sections spaced at 120° with lands between, enabling the specimens to be positioned in these lands close to the carrier surface. The power capabilities of the heater tested have proved to be 100% greater than a design using commercial mineral insulated cable and it has completed 4000 hours of operation at 1000°C and 1.45 kW power output.

3.3 Totally-Enclosed Concentric Tube Heater

The requirement to operate some irradiation heaters in environments where contamination of the magnesia insulant and the possibility of shorting the heater element to the sheath exists, necessitated the production of a fully-enclosed heater. At the time of writing no long term testing of this heater has taken place but there would seem to be no reason why its performance should not be at least comparable to that of the Type II heater.

3.4 Future Possible Developments

Although these heaters have been designed primarily for irradiation experiment rig work the basic design is capable of modification in detail to render it suitable for the production of high heat fluxes. One heater with larger terminals produced a heat flux of 180 watts/cm² to flowing water at 20°C without failure.

Refractory and noble metals may be suitable for operation at temperatures greater than those possible with 80/20 nickel-chromium. The specific resistance of magnesia and indeed most insulants, above 1200°C seems to be the limiting factor.

4. Bare-Metallic Heaters

High-temperature heaters are arbitrarily defined here as heaters capable of maintaining specimen temperatures higher than 1000°C in the absence of nuclear heating. Development has been attempted on two types of heater for operation at temperatures above 1200°C. A tungsten wire element has been successfully operated for 5000 hours at 1600°C in out-of-pile tests, and development has been abandoned on a tungsten foil element for similar temperatures. The wire-type heater has been incorporated in an experimental rig for irradiating small fissile ceramic specimens; this rig has operated in a materials testing reactor since 13.1.66.

4.1 Tungsten Wire Heater for 1600°C Operation

A tungsten heater to operate in-pile at 1600°C for a period of at least six months is required. The out-of-pile development work to achieve this and the present state of in-pile experience are described.

The major problems in designing a high temperature tungsten heater are firstly to maintain effective insulation and secondly to prevent chemical attack of the tungsten winding. In the heater tests to be described, alumina has been used as the insulating material in the form of sintered rods and tubes. The problem of oxidation of the winding has been overcome by provision of a reducing atmosphere of 98% helium + 2% hydrogen. The thermodynamics of the tungsten-tungsten oxide system indicate that an impurity content of up to 200 ppm water + oxygen may be tolerated at 1600°C in the presence of hydrogen before oxidation of the tungsten occurs. This method of protection is preferred to the use of rigorous clean-up procedures to remove all oxygen and water, since in a complex system such as an in-pile experimental rig it is virtually impossible to remove all traces of oxygen, and particularly absorbed water, from the system. The use of powdered or high-specific-area materials in the construction of the furnace has been avoided because of their readiness to absorb water when exposed to the atmosphere and then to release it when hot.

The furnace (Fig. 18) is essentially an alumina crucible 65 mm x 55 mm x 300 mm long, with three heater assemblies fitted internally. A heater assembly consists of 1.5 mm tungsten wire bent in a hairpin configuration and fitted into the annular space formed between two alumina insulator tubes. The space between adjacent limbs of the heater element is filled with alumina rods, two of which (diametrically opposed across the annulus) are twin-bored to accommodate thermocouples. In the test furnace tungsten-rhenium and one platinum-rhodium thermocouple were fitted. The space between the heater assembly and the inside of the crucible is completely filled with alumina rods. One large tube (8 mm x 5 mm) is inserted among the insulator rods to act as a 'breather' tube to connect the top of the assembly to the bottom with minimum restriction of gas flow. The assembly is supported on two molybdenum discs which fit to the bottom of the 'U' of the crucible, and is further supported on a short length of alumina tube.

The inner former tubes are 550 mm long and are attached at the upper end to individual bellows to allow for expansion.

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Fig. 18. Tungsten Wire Heater for 1600°C In-Pile Operation

Fig. 19. Tubular Tungsten Heater

A radiation heat shield system is provided by 10 molybdenum plates 0.12 mm (0.005 in.) thick, bent into a circular form and dimpled to a depth of 1.25 mm (0.050 in.) to maintain concentricity. Care is exercised during assembly to ensure that dimples are not coincident on adjacent plates. Molybdenum plates (0.12 mm) at the top and bottom of the crucible are dimpled in a similar manner to reduce radiation end losses.

The complete furnace assembly of element insulators and radiation shields is housed in a welded stainless steel pressure vessel, a stainless steel top flange being sealed to the vessel by O-rings. The top flange is water cooled to preserve the O-rings. The pressure vessel is housed in a second containment which is water cooled; this in turn operates in an alumina containment which is helium filled to simulate the reactor experimental atmosphere. Thermocouples are fitted at numerous positions in the assembly and are led out through rubber compression seals. Provision was made for a platinum-rhodium thermocouple to traverse the axis of the heater in order to measure the longitudinal temperature distribution.

The heater element is attached by palladium-nickel braze (M.P. 1250°C) to stainless steel lead-outs which are then bolted to copper leads passing through compression seals in the top plate. The Pt-20 Rh/Pt-40 Rh thermocouples are joined to copper wires before leaving the pressure vessel, and a separate chromel-alumel thermocouple measures the cold junction temperature at this point.

The heater operates in a flowing atmosphere of helium + 2% hydrogen supplied ready mixed at a rate of 50 ml/min. Before entering the furnace the gas passes through a "Deoxo" unit and a manganous oxide bed at 150°C to remove oxygen. A molecular sieve at room temperature is used to remove moisture. Finally an Elliott moisture meter and a Hersch cell oxygen meter are used to measure the moisture and oxygen content of the emerging gas. The system operates at a pressure of 0.20 Atm.

4.1.1 Manufacturing Problems

Dimpling of the molybdenum heat shields required development of a suitable tool. A simple hemispherical tool cracked the molybdenum. The successful tool was designed in such a manner that with an aluminium backing sheet the molybdenum was given adequate support.

Tungsten-rhenium thermocouples broke at the hot junction during assembly and a technique of argon-arc welding of the junction was developed.

Induction brazing of the expansion bellows required re-design of the joint area to obtain satisfactory brazes.

Early work on the brazing of heater and heater leads was successful with dummy joints, but the blanket gas ignited during brazing of the heater assembly due to insufficient containment of the joint assembly. Reduction of hydrogen content in the blanket gas reduced wetting properties and necessitated the use of flux. Successful joints were obtained without flux by using NMP1 braze with a liquidus some 130°C lower than that of PN1.

4.1.2 Laboratory Tests

The furnace assembly was vacuum degassed for a period of five days at a maximum temperature of 1000°C. During this time the outgassing rate

measured at 0.05 torr fell from 3 torr/hour to 0.16 torr/hour. A further period of 2 days at 1200-1600°C reduced the outgassing rate to 0.06 torr/hour.

The impurity levels of the helium-hydrogen mixture were reduced to 12 vpm water vapour moisture and 4 vpm oxygen before the gas was introduced to the furnace chamber. After 24 hours purging the initial oxygen impurity of 67 vpm at the outlet of the cold furnace fell to 35 vpm. The heater was then switched gradually to full power to obtain 1600°C.

After 1 week at this temperature all platinum-rhodium thermocouples had failed and control of the heaters was maintained by the tungsten-rhenium thermocouples. The longitudinal temperature distribution along the furnace was 1600° ± 50° over a distance 12 mm above the centre line to 90 mm below the centre line. The total power input to the three heaters to maintain this temperature was 2500 watts. The effect of failure of individual heaters was checked by switching off each heater in turn. This resulted in a reduction of temperature for the 'off' heater of approximately 400°C, the remaining two heaters registering a temperature drop of approximately 200°C.

After 500 hours the furnace was cycled 20 times from 200 to 1600°C at a rate of 700°C/hour without failure. Thereafter the furnace operated continuously for 5000 hours without failure and was finally stripped for examination of the components.

4.1.3 Examination of Components

The ceramic heater formers and crucible were broken, but had been held correctly positioned due to the rigid construction employed. A 9.5 mm diameter specimen-can could still be passed through the formers without difficulty. All ceramic items showed pronounced crystalline structures in the region where temperatures had been higher than 1500°C; there was evidence of some bowing and discoloration. The tungsten wire elements were in good condition and the maximum reduction in diameter was 0.16 mm from the original 1.62 mm; the wire was bright in the hottest regions. The tungsten wire closest to (but never in contact with) the platinum-rhodium thermocouples had visible growths which were point-analysed as tungsten 53%, platinum 34%, rhodium 10%. Due to the reducing atmosphere of the furnace the platinum-rhodium thermocouples were severely corroded; the tungsten-rhenium thermocouples were embrittled and broke during handling but had operated successfully. Metallurgical examination of selected components showed recrystallisation of the tungsten element in the hot zones, with an intermetallic compound present at the surface in some areas. The grain structure across the wire diameter varied from a recrystallised equiaxed structure at the centre to a fibrous structure at the surface.

4.1.4 Conclusions

This prototype heater had operated successfully for 5000 hours at 1600°C and the success is attributed to three main factors:-

- (a) The 2% hydrogen prevented oxidation of the tungsten.
- (b) The tungsten wire was of uniform cross-section and free from deep defects which would lead to hot-spot formation.

- (c) By using ceramic rods to fill the spaces between the main ceramic components the structure of the furnace was maintained sufficiently rigid to prevent these components falling apart after they had cracked.

The furnace was considered suitable for adaption to an in-pile irradiation rig. The following paragraphs report on the earlier high temperature furnaces used in a research reactor and the behaviour of the prototype rig using this furnace.

4.1.5 In-pile Operation

The out-of-pile development work just described was carried out in support of an in-pile experiment, SD 727, designed to irradiate small specimens of ceramic fuel material at temperatures of up to 1600°C. The centre furnace section is essentially the same as that described in the out-of-pile tests, (Fig. 18), having three furnace assemblies of concentric alumina tubes with tungsten wire heaters between. The assemblies set in a large alumina crucible which is surrounded by molybdenum radiation shields. The whole assembly is fitted into an outer stainless steel can with a welded stainless steel lid. This can replaces the stainless steel vessel with top flange described in para. 4.1. The electrical leads are brought out through ceramic seals welded into the can lid and are extended through the plug to the rig head. The gas supply is fed through the rig plug and enters the can through a pipe welded into the lid. The return gas flow is via the three furnace tube assemblies which are extended to the rig head. These tubes also serve for the introduction of specimens into the furnaces. The gas flow is supplied from a console near the reactor top plate and is returned for monitoring before rejection to extract. The furnace temperature is measured by W-5 Re/W-26Re thermocouples, molybdenum sheathed. Two thermocouples are placed in each furnace assembly in diametrically opposed positions between the vertical limbs of the heater.

This rig was loaded into the reactor in January 1966 and at the time of writing has not yielded any operational data on furnace performance. The furnace assembly under nuclear heat alone in a flux of 4×10^{15} neutrons/cm² sec. reached a temperature of 650°C. Some in-pile experience of tungsten heaters had been gained in other experiments. Four furnaces were operated in-pile at temperatures up to 1600°C with moderate success and are described in detail elsewhere(3,4). They were of vertical hairpin configuration but the tungsten wire was embedded in alumina cement.

Three furnaces operated in a static helium environment and ran without difficulty at 1200°C; one failed after 5 days at 1400°C, one 35 hours after reaching 1600°C and the third ran for 43 days at a temperature of between 1480°C and 1600°C. The fourth furnace was operated in a static helium + 2% hydrogen atmosphere and ran without difficulty at 1400°C, but failed after 5 days at 1600°C.

The reason for the premature failure of these heaters is not known. It is possible that the alumina cement may have contained impurities which attacked the tungsten (probably by forming a liquid phase). Otherwise there may be an effect associated with radiation that causes an insulation breakdown, or the use of a static atmosphere may have been detrimental. Until irradiation tests have been completed on the design that has been successfully tested out-of-pile, further conclusions can not be drawn.

4.2 Tungsten Foil Heater for 1600°C Operation

The heating element was a 0.25 mm thick tungsten foil, slotted and rolled to form a 25.4 mm diameter hollow cylinder with 6 arms of 4.8 mm width. The tubular heater thus formed and shown in Fig. 19, was inserted into a system of molybdenum radiation shields and contained in a stainless steel capsule. The heater was suspended from the top of the capsule, to allow free expansion without bowing, and a ceramic insert in the base of the heater cylinder prevented excessive lateral movement. The capsule was contained in a cooling water jacket, and a radial hole sealed by a quartz window permitted observation of the heater surface. The furnace atmosphere was high-purity helium, bled through the system at 25 ml/min at a pressure of 0.24 Atm. Conventional sealing methods were employed throughout the system.

The high purity helium was obtained by passing commercial helium gas through a manganous oxide trap at 150° and a molecular sieve cooled to liquid nitrogen temperature. This system effectively produced helium with impurity levels less than 1 ppm each of O_2 , N_2 , CH_4 , CO , and CO_2 . H_2 and H_2O levels were estimated at less than 3 ppm.

4.2.1 Manufacturing Techniques

The heater was produced to the rough shape by clamping the foil between suitable templates and shearing the unwanted metal with a small chisel. The final form was obtained by hand grinding, and hot bending was employed to produce the tubular shape.

4.2.2 Operational Behaviour

Difficulty was encountered in maintaining the purity of the helium in the furnace and in measuring the impurity levels; adsorbed gases on pipework presented a major problem. To measure impurities a spark discharge technique was employed. In this method the operator discharges a 14,000 volt spark into a glass tube in the gas supply line and estimates impurity levels from the colour of the discharge. This proved satisfactory since gas chromatograph analysis taken at the same time indicated less than 3 ppm impurities. The spark discharge method is simple, reliable and inexpensive.

Vacuum degassing over a long period and "flaming" of the pipework with a gas torch was necessary to remove adsorbed gases. The adsorbed gases inside the furnace were removed by gentle heating before switching the heater to its full power of 2 kW.

The first heater to be tested failed after 650 hours at 1600-1700°C. Failure occurred when the power was restored to the heater after a resistance measurement had been made.

A second identical heater was tested, and failed after 528 hours at 1600-1700°C.

Simultaneously with the foregoing tests, tungsten strips (0.25 mm x 4.8 mm cross section) were tested in various atmospheres to estimate suitability of the material.

4.2.3 Examination of Furnace, Heater and Test Strips

After the first heater failed the furnace showed little sign of oxidation. Black and yellow deposits on the ceramic components were identified as tungsten. The molybdenum heat shields were bright and the inner shield was coated with tungsten crystals. Five of the six limbs of the heater element had "waisted" considerably at the centre sections and the cross-section at the point of failure had reduced from 0.25 mm x 4.8 mm to 0.18 mm x 3.8 mm. Microscopic examination of the sections showed recrystallised equiaxed grains. Fissures and craters were present and there was some evidence of delamination of the tungsten strip.

After the second heater had failed a similar examination was made and the furnace was found to be considerably discoloured; the heat shields were brown in places and red needle-like crystals of $W_{18}O_9$ had been deposited between the inner and second heat shield. Four of the six heater limbs were reduced to about 50% area in the central regions, and examination revealed a medium-sized equiaxed grain structure increasing in size towards the site of failure, where single crystals were observed. There were occasional whole grains missing from the material and both longitudinal fissures and intercrystalline cracks were present.

Examination of test strips in all cases showed recrystallisation and coarse equiaxed grains at all temperatures up to 2000°C. All strips tested in helium failed after short lifetimes.

Strips tested in 95% He + 5% H_2 operated for over 2700 hours at 2000°C and for over 4800 hours at 1600°C, in each case without failure. One strip at 1970°C in 10^{-5} torr vacuum ran for 5300 hours without failure; a second strip in 10^{-3} torr vacuum lasted 411 hours at 1970°C before failure occurred.

4.2.4 Conclusions

The above tests show that a thin foil is not suitable for operation at temperatures above 1600°C. Whereas introduction of hydrogen into the helium atmosphere gas inhibits the characteristic thinning of the tungsten element, severe recrystallisation still takes place in the material. The resultant large grain structure produces embrittlement with early failure across reduced sections. Because all tests were conducted under vibration-free conditions it is considered that the strength of the foil is insufficient, even though thinning may be prevented, to warrant this type of heater being developed for use in in-pile rigs. The delamination of the foil noticed during metallographic examination indicates that a tungsten wire element is preferable.

5. Ceramic and Non-Metallic Heaters

5.1 Tungsten-Alumina Cermet

Development of a cermet heater has recently commenced in connection with high-temperature experiments in BR.2 and D.F.R.

The design requirements are a small tubular heater approximately 15 mm outside diameter capable of maintaining specimen temperatures of

Fig. 20. Tungsten Alumina Cermet Heater

FIG. 21. SILICON CARBIDE HEATER

1200°C. Calculations predict a heat flux of 55 watts/cm² from the curved surface of the cylinder, equivalent to 250 watts per cm. length of heater.

Previous experience with coaxial sheathed heaters has produced the approximate ruling for this size of specimen matrix that the core temperature of the heater is higher than the specimen temperature by approximately twice the core-to-sheath temperature difference. Based on this, a system is proposed employing a high-melting-point electrical conductor in a matrix designed to reduce temperature gradients. It consists of a central niobium specimen matrix onto which alumina is sprayed for insulation; a cermet of tungsten and alumina is sprayed onto the alumina and a further alumina deposit forms the outer insulation layer. This is illustrated in Fig. 20. The thickness of the successive layers is machined at each stage to 0.25 mm. By careful adjustment of the ratio of tungsten to alumina in the cermet, the costly machining which would be necessary to produce the required resistance is eliminated. The cermet is slotted helically to permit electrical connection at one end only. The advantages of using a deposition system compared with a wound heater are obvious. By using a cermet of similar nature to the insulant good bonding is expected, eliminating delamination problems. This excellent contact together with the thin layer of insulant between heater and niobium matrix should considerably reduce temperature differences. Preliminary work has proved the feasibility of the deposition techniques and has determined that current density is uniform throughout the cermet. Now under study is the relationship between mixture ratio and heater resistance, and a metal deposition technique for connecting the electrical lead outs to the heater.

The final heater is intended to be a hermetically sealed unit in which a niobium outer sheath is welded to the niobium former.

5.2 Silicon Carbide Heater

A resistance heater has been developed to simulate miniature fuel elements having surface heat outputs of up to 500 watts/cm² in liquid sodium. At this heat flux the fuel element cladding temperature should be about 650°C unless film boiling of the sodium occurs. The overall size of the fuel pin assembly is 5.1 mm diameter x 69 mm long, and the total heat output is more than 6 KW.

The heater (see Fig. 21) consists of a 4.32 mm diameter silicon carbide rod (the resistance element) insulated by 0.1 mm of alumina. The silicon carbide rod was prepared by careful grinding with a diamond wheel, and the alumina insulant, deposited by plasma-spraying, was also ground to size with a diamond wheel. The assembly was completed by shrinking the stainless steel cladding onto the alumina.

The original heater had electrical connections consisting of nickel rods pressed against the ends of the silicon carbide inside the cladding tube, the contact surfaces being ground flat and parallel. This achieved a heat flux of 195 watts/cm² (2.5 kW total), above which the heater short-circuited to the cladding tube at the connection points. In the present heater the silicon carbide rod is extended out of the cladding tube so that nickel connectors can be clamped to it. This has given no trouble and has operated up to a heat flux of 375 watts/cm² (4.5 kW total), but it appears essential for all components to be produced to a tolerance of 0.005 mm.

5.3 Zirconia Heater

The programme of neutron diffraction studies at A.E.R.E., Harwell requires the heating of specimens of materials to high temperatures in either an oxidising or a neutral atmosphere and with a minimum of absorption or scattering of the neutron beam by the furnace elements.

A furnace has been designed and developed which will meet these requirements and which will uniformly heat material samples of up to one inch diameter and two inches long to a minimum temperature of 2000°C.

The furnace employs a two stage resistive heating element consisting of two concentric tubes.

The inner element is a stabilised zirconia tube which becomes electrically conducting as the temperature increases above about 1000°C. The passage of an electric current through the tube then raises the temperature of the furnace to the desired level.

The outer tube may be either of silicon carbide or of alumina with a continuous surface film of platinum. This tube is required to be electrically conducting at room temperature and the passage of an electric current through the tube raises its temperature to about 1000°C. Heat is transferred to the inner stabilised zirconia tube which is raised in temperature to the point at which it becomes electrically conducting; the second stage of temperature rise is carried out either using the inner tube alone or together with backing heat supplied by the outer tube.

The inner zirconia tube element can be stabilised to give a single phase with either calcia or yttria. The tube is composed of a series of rings which are graded in cross sectional area so that maximum wall thickness occurs at the end of the tube. The separate rings are joined to form the tube by sintering with a slurry of the tube material. The outer tube is of solid construction and acts as a heat reflecting barrier in addition to being the primary heater.

Radiation shields consisting of alumina and quartz tubes coated with a thin platinum film surround the heating tubes. Water cooling is provided to maintain the heating tube ends at a temperature low enough to enable satisfactory electrical end connections to be made.

The arrangement adopted has the essential advantage of a virtually uniform temperature distribution along the axis in the working section of the inner tube. At 2000°C the mean temperature gradient from the centre to the ends of a four inch zone would typically be about 15°C.

At operating temperature the resistance of the inner tube falls to a few ohms, which is ideal for thyristor control from the mains voltage without a transformer. Furnace temperature level is controlled by a Land miniature radiation pyrometer, the output being fed to the controller in opposition to the demand signal; this results in a reduction in the power supplied to the inner tube as the furnace approaches the set temperature.

An advantage of the system is that it allows continuous automatic control after the initial setting of the desired operating temperature. Means of varying the power to the outer tube primary heating element during the start up period has of course to be provided.

The inner tube has a negative temperature coefficient of resistance. This assists the attainment of uniform temperature distribution in the working section but necessitates the use of a current limiting device in case the pyrometer output should fail.

Although this furnace has been developed for use in neutron diffraction studies it could be developed for use in a wide range of applications.

6. Induction Heating Applied to Reactor Experiments

No in-reactor irradiation experiments using this form of heating have been performed within the U.K.A.E.A., but experience has been gained on this form of heating for out of reactor loop and test rig experiments.

Figure 22 shows a graphite tube furnace used in a mass transfer loop experiment. The furnace 51 mm outside diameter, 38.1 mm inside diameter, is heated at 280 Kc/s to a temperature of 1800°C using a power of 60 kW over a 122 cm. length; the gas environment is helium at 21.4 Atm. Induction heating was used in order to eliminate the use of materials other than stainless steel, graphite, or pure ceramics in the experimental zone. Unfortunately this furnace failed with helium due to the voltage distribution between radiation shields being such that with 7.2 kV across the work coil breakdown conditions were achieved, this being the result of mixed dielectrics conditions between work coil and graphite tube. A separate experiment to investigate the breakdown properties of helium established that the full rating of 60 kW could be obtained when nitrogen or carbon dioxide was used as the furnace atmosphere.

Figure 23 illustrates the layout of an experimental heater for fitting into an existing convection-cooled, pressurised-water test rig. The rig is suitable for pressures up to 137 Atm. and temperatures up to 330°C. The experimental section consists of a cylindrical ferromagnetic insert located within a stainless steel pipe and heated by currents induced by an external coil energised from a 50 c/s supply.

Heat developed in the insert is transferred directly into the water without causing a temperature gradient across the wall of the pipe. In this manner the pipe stresses, which are considerable at pressures of 137-275 Atm., are not further increased by a radial temperature gradient as would be the case if heated externally.

Analysis

Currents are induced in a body by the passage of a varying magnetic flux; if the material is of optimum conductivity and permeability considerable quantities of heat can be produced. Certain conditions however must be fulfilled and these are not easily established. As this form of heating can be performed without mechanical or electrical contact it can be invaluable when pressure vessel walls have to be penetrated to perform heating within. It is this aspect which is of importance to the Authority rather than its ability to heat quickly the surface layers of a solid.

Intrinsic Heat Equation

MacLachlan⁽⁷⁾ has analysed the induction heating of a cylindrical solid wholly within a solenoid whose diameter is small compared to its

FIG. 22. H.T.G.C. FURNACE TEST SECTION

FIG. 23. INDUCTION HEATING OF HIGH PRESSURE WATER TEST LOOP.

length; in establishing the heat equations it is assumed that the permeability of the heated body remains constant and the energising frequency is far removed from the natural frequency of the system. It is shown that:-

$$P = \frac{4\pi n^2 \mu \omega I_{rms}^2 f(W)}{l^2} \quad \text{loss/unit volume} \quad \dots (1)$$

Where n = number of turns in the energising solenoid coil.

μ = permeability of the heated solid.

$\omega = 2\pi \times \text{frequency.}$

I_{rms} = rms current in the solenoid.

l = solenoid length.

$f(W)$ = loss function.

$$= \frac{2 \{M_1(ma) \cos(\theta_1 - \theta_0 - \pi/4)\}}{ma M_0(ma)} \quad \dots (2)$$

$M_1(ma)$ = absolute value of the 1st order Bessel function $J_1(ma^{3/2})$.

$M_0(ma)$ = absolute value of the zero order Bessel function $J_0(ma^{3/2})$.

θ_1, θ_0 = angles subtended by $M_1(ma)$ and $M_0(ma)$ from the zero on the Argand diagram.

a = radius of the heated solid.

$m^2 = \frac{4\pi\omega\mu}{\rho}$

ρ = resistivity.

All values are absolute in computation.

Function $f(W)$ is drawn in Figure 24A for values of ma between 1 and 10. The interpretation of equation (1) is difficult as a large number of interdependent variables affect the power generated, but attempts are made to approach the maximum shown in Fig. 24A. This is obtained when

$$\frac{4\pi\omega}{\rho} = \sqrt{(2.5/a)} \quad \dots (3)$$

$$\text{From this, } \frac{\mu}{\rho} = \frac{I}{4\pi\omega} \cdot \sqrt{(2.5/a)} \quad \dots (4)$$

ρ in the above equation varies linearly with the temperature whilst the intrinsic permeability of the material varies in a complex manner according to the magnetising force, temperature, and the demagnetising factors of the specimen.

Flux Distribution

The heating involves both an in-phase component dictated by equation (1) and an out-of-phase value, where $f(\theta)$ replaces $f(W)$ so that

$$f(\theta) = f(W) \tan(\theta_1 - \theta_0 - \pi/4) \quad \dots (5)$$

This is in fact the magnetising component and is a measure of the flux distribution across the specimen; Fig. 24A shows this function plotted over the same interval as $f(W)$. If an attempt is made to calculate the equivalent inductance of the specimen then

$$\mu_{\text{eff}} = f(\theta)\mu \quad \dots (6)$$

would be the value to be used in the inductance formula.

Current Distribution

The induced current is not distributed linearly across the specimen but is forced to flow around the path of least impedance, namely in a skin whose thickness is a function of ma ; the distribution is given by:-

$$I_r = I_a \frac{f(mr)}{f(ma)} \quad \dots (7)$$

Where I_r = maximum value of current at radius r ,

and I_a = maximum value of current at radius a , i.e. at the surface;
 $f(mx)$ is plotted in Fig. 24B.

Heating and Frequency

It is evident that the current distribution and heating conditions are strongly dependent on the complex variable. If possible the heating should be performed on non-magnetic materials where μ can be disregarded and optimum conditions more easily defined. Provided these conditions are maintained equation (1) shows that the power generated increases with frequency; the current however is confined to a skin defined by equation (7).

7.1.47

FIG. 24A.

FIG. 24B.

High frequency is thus used for techniques which require local heating upon the surface sheath of a body such as for case-hardening or pipes carrying fluid. Where bulk heating is demanded the frequency must be lowered until the current penetration is sufficient for the application under discussion. Analytically there is no real justification for using a lower frequency than is demanded from the above reasoning, as it will be found that either I_{rms} in equation (1) must increase rapidly or a magnetic material must be used to obtain optimum heating conditions.

Limitations to Analysis

The above analysis has been presented only to indicate some of the design problems that are thrown up by this form of heating. Only passing reference has been made to the complexities that are introduced when magnetic materials are used, and for further information on this aspect of the problem standard works on magnetism should be consulted. Generally speaking power frequencies are not used, but two experiments are being conducted to determine what loading figures can be obtained particularly in its use for heating a magnetic material through the walls of a stainless steel vessel. The methods of matching the load to the generator have not been discussed, nor the type of generator and the necessity of operating at unit power factor.

Experiments are continuing in an attempt to examine the possibility of establishing a sound theoretical explanation for many of the apparent contradictions that have so far been obtained in the experimental results.

7. Electron Bombardment Heating

Introduction

For many of the advanced irradiation experiments being proposed, there is a lack of heat transfer data; it has therefore been necessary to develop a heater other than the electrical resistance type. This heater is required to:-

- (i) produce a surface heat flux of 450-600 watts/cm² on a cylindrical surface about 6.5 mm (0.25 in.) diameter,
- (ii) produce surface heat fluxes of up to 3000 watts/cm², for basic work on forced convection heat transfer and bubble nucleation⁽⁸⁾.

If requirement (i) above can be implemented, heat transfer conditions in sodium-filled irradiation capsules containing fissile specimens will be reproducible in the laboratory. At present the only way of producing such heat fluxes is to mount an in-pile experiment containing fissile specimens of appropriate enrichment.

A device has been designed and constructed which is basically a high-power vacuum diode of coaxial, cylindrical configuration. The diode has not yet been operated owing to difficulties in cathode manufacture, but first results should be obtained by late April, 1966.

Design Requirements

- (i) A heat transfer surface diameter of about 6.5mm as in (i) above.
- (ii) Minimum heat flux of 450 watts/cm².

(iii) Potential for development to 3000 watts/cm².

(iv) Use of mains water, demineralised water, or various liquid metals (particularly sodium) as coolants.

Design Conception

A low-voltage, high-temperature-cathode design was chosen in order to avoid:

- (a) the high cost of an EHT power supply,
- (b) the general laboratory risk, and circuit protection difficulties of an EHT supply, and
- (c) the problem of insulator flash-over.

Description of Diode (See Fig. 25)

The central cathode/filament is a slotted rod of tantalum or tungsten designed to operate in vacuo at about 2600°K. It is located in the anode by machined thorium oxide insulators.

The cathode-anode gap is 0.75 mm (.030") and in the present prototype it has proved very difficult in practice to maintain this with any uniformity. A Mk.2 design is at present being detailed which avoids the problems of (i) concentric location and (ii) high-temperature, high-voltage insulation.

The anode is a stainless-steel tube 6.5 mm O.D. x 140 mm long designed to screw into the head of the diode; this allows anodes of various external surface finishes to be used interchangeably. The outer surface is the heat transfer surface to the coolant.

Electrical Power Supplies

Filament

The filament A.C. power supply of 1800 VA maximum is by a fixed transformer of 300 amp. rating with 6V and 12V tapplings insulated to 1000V D.C. This is driven by a 2.1 KVA Variac supplied from single phase 240V mains.

H.T. Supply

A continuously-variable, 3-phase, 415 volt regulator drives a 3-phase fixed transformer with 6-phase secondary output. This is connected to a 6-phase series bridge rectifier employing solid-state diodes. The D.C. output is 0-1000V, 80 KVA maximum, with ripple quoted as 1% without other smoothing.

Theory (See Fig. 26)

The performance calculations take no account of field emission from the cathode - that is to say, free electron emission is calculated from the Richardson-Dushman equation (given below), and field emission current is regarded as a bonus, or positive margin of performance.

7.1.50

ELECTRON EMISSION HEATER
FIG. 25.

FIG. 26. ELECTRON BOMBARDMENT HEATING

Free emission current is given by

$$I \text{ amps/cm}^2 = AT^2 \cdot \exp(-\phi/kT).$$

where A = empirical constant, or Richardsons's 'A'

T = cathode surface temperature, $^{\circ}\text{K}$

ϕ = cathode surface work function, ev

k = Boltzmann's Constant, $86.2 \times 10^{-6} \text{ ev}/^{\circ}\text{K}$.

also Total Power = Total Emission Current x Anode Voltage.

Much design work was based on the above equation for different cathode materials. However for the present tantalum cathode, values of I were taken from experimental work on direct-conversion diode materials⁽⁹⁾. Quoted values of 'A' vary widely from one reference to another⁽¹⁰⁾; also, the effective value of ϕ depends on the purity and metallurgical state of the cathode material⁽¹²⁾. The exponential dependence of I on ϕ (and T) makes it very difficult to predict the performance of this device within about 50%. However this same exponential relationship should make it possible (certainly in theory) to produce both 600 and 3000 watts/cm². from the same diode by only a relatively small change of T (about 8%).

It will be noted that the energy emitted from the cathode is reduced in density in the ratio cathode diameter/anode diameter; this effect could be reversed for some applications by having the cathode exterior to the anode. However the heat transfer surface would then be virtually inaccessible.

Before power tests are made on the diode, filament power will be calibrated against its nominal temperature as measured by a tungsten-rhenium thermocouple. The thermocouple will be removed for high-voltage operation, otherwise it and its instrument would be at 1000 volts potential relative to earth, constituting an extra safety hazard.

Proposed Development

In the event of present tantalum cathodes not giving the required performance it will be necessary to develop oxide-coated or orientated-crystal⁽¹¹⁾ cathodes of appropriate geometry. Pyrolytic deposition^(12,13) is being used at present to produce tungsten cathodes. Another line of development is that of highly-rated emitter materials such as W-La-Zr⁽¹⁴⁾.

American workers proposed in 1964 to develop this type of heater for sodium boiling studies⁽¹⁵⁾. However one project stopped because of financial difficulties⁽¹⁶⁾, and results from another project were very disappointing, the net heat flux developed being about 10 watts/cm².⁽¹⁷⁾

Design of an electron bombardment heater is now in progress at Dounreay for use in sodium boiling work⁽¹⁸⁾.

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THE USE OF NOBLE AND REFRACTORY METAL THERMOCOUPLES
IN EXPERIMENTAL IRRADIATION EQUIPMENT

by

N. H. Hancock

ABSTRACT

For the measurement of temperatures above 1000°C in reactors, noble or refractory metal alloy thermocouples are most commonly used. Some of the advantages and disadvantages of the platinum and tungsten alloys are stated and the irradiation experience at A.E.R.E. Harwell is summarized.

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1. Introduction

When faced with the problem of measuring temperatures of above 1000°C for extended periods in a high neutron flux environment, the only readily available commercial mineral insulated sheathed thermocouples which appear to be suitable, are the tungsten-rhenium, platinum-rhodium, platinum-molybdenum and platinum-palladium types. Each of these have disadvantages such as calibration drift due to transmutation, doubtful reliability due to fragility of conductors or high temperature embrittlement, or poor corrosion resistance in all but very pure atmospheres. In addition the tungsten-rhenium thermocouples in the lengths required for most experimental assemblies are extremely expensive. Experiments at Harwell have concentrated on the use of the platinum-rhodium and tungsten-rhenium types as from early surveys these offered the most readily available high temperature experience.

2. Platinum-rhodium alloy thermocouples

The platinum-rhodium thermocouples for temperature measurement up to 1500°C have the very attractive properties of:

- (a) having excellent oxidation resistance,
- (b) having excellent workability,
- (c) having an accurately known and reproducible calibration,
- (d) are readily available in sheathed form,
- (e) are not excessively expensive for this class of thermocouple.

The well-known disadvantages are the high neutron absorption cross-section of rhodium and the incompatibility of platinum with reducing conditions. However, because of the advantages mentioned above this type of thermocouple has been used with considerable success in some A.E.R.E. Harwell irradiation rigs in conditions for which, at first sight, they would appear to be completely unsuitable. The thermocouples used were $1/16$ " overall diameter, with platinum vs platinum - 13% rhodium cores insulated with MgO and sheathed in platinum - 13% rhodium for 6" to 9" at the hot junction end with a further 10 ft. sheathed in stainless steel. The hot junction was isolated from the sheath and the sheath seal welded at the end. The hot junction end was in contact with graphite over a length of several inches and operated at temperatures up to 1400°C in an atmosphere of commercially pure neon and a thermal neutron flux of approximately $10^{14} \text{ n/cm}^2\text{-sec.}$

The failure of the platinum-rhodium sheath under the reducing conditions was avoided by eliminating as far as was possible all deleterious impurities, in particular silicon and sulphur which can lead to serious alloying with the platinum. This was accomplished by using a

very pure experimental grade of graphite and the thermocouple sheath during assembly of the rig. Post-irradiation examination of thermocouples has verified that no reaction has taken place in the thermocouple sheath.

Although these thermocouples were required to operate for only about 20 days at temperature in reactor, satisfactory operation from 40 to 180 days have been achieved at temperatures of 1300°C before failure by open-circuit. Some of these thermocouples operating at 1000°C to 1200°C have been consistently reliable. The open circuit failures are thought to be due to the grain growth embrittlement of the pure platinum conductor combined with the stressing applied to the cores through differential expansion between the stainless steel part of the sheath and the cores. To overcome this in recent rigs the platinum-rhodium sheath has been extended to well outside the reactor core to a cooler region of the rig and the pure platinum conductor has been replaced by Pt - 1% Rh alloy. Experience to date is promising.

3. Tungsten-rhenium alloy thermocouples

For operation above 1500°C , no thermocouple combination is presently available other than the tungsten-rhenium alloys. Recent laboratory work at A.E.R.E. Harwell has shown that thermocouples of W5%Re vs W26%Re, W3%Re, vs W26%Re or W vs W26%Re are all stable at temperatures of over 2000°C in inert atmosphere or in vacuo. Calibration standards have been established for the W vs W26%Re thermocouple by the National Bureau of Standards on the basis of comparison of a large number of manufacturers products. The two main disadvantages of this thermocouple is the general lack of oxidation resistance (which is common to all the tungsten alloys) and the very poor ductility of pure tungsten. For this reason, W3%Re or W5%Re, alloys which have rather more ductility than pure tungsten, have replaced pure tungsten in the positive element. Unfortunately with these alloys the sensitivity is reduced as compared to W vs W26%Re and linearity is less good. Both disadvantages are acceptable however, for experimental work.

The requirement for measuring fuel centre temperatures in irradiation of experiments between 1600°C to 2000°C has led to the use of W5%Re vs W26%Re in an increasing number of irradiation experiments at A.E.R.E. Harwell. In all cases, the thermocouples were in mineral insulated cable form, with isolated hot junctions and with BeO insulation over that portion of the cable (normally 2 cm to 5 cms) which was in the hot zone. Al_2O_3 or MgO have been used as insulants outside this zone. The material of the sheath has varied accordingly to its environment, tantalum being used alone when in contact with carbonaceous materials at temperatures up to 1800°C . Above this and at 2000°C , a sprayed layer of tungsten is added as an additional compatibility barrier. When in contact with oxide fuels,

molybdenum has been selected as the sheath material, since tantalum is suspected of incompatibility with UO_2 under irradiation at temperatures above $1400^{\circ}C$. The normal overall length of the thermocouple cables (some 4 metres) has been considered to preclude the use of the 'hot end' materials over the whole length on grounds of cost and it has thus been common practice with A.E.R.E. irradiation experiments to braze the short (0.2 metres) tantalum sheathed section to a stainless steel sheathed extension about 3.8 metres long. It would of course be desirable, to avoid the brittle junction between the refractory metal and stainless steel and sheaths, and to make the refractory metal sheath continuous for the whole length of the cable, but the cost of a continuous tantalum sheathed cable is some 20 times that of stainless-steel sheathed. More rarely, and again on grounds of cost, the W5%Re/W26%Re elements have been joined to extension wires of less costly material but this practice was rapidly discontinued when serious difficulties were found in obtaining a satisfactory joint.

In the majority of irradiation rigs, the thermocouple sheath is in contact with the rig atmosphere which is nominally pure helium. The absolute purity of this cannot, however, be guaranteed and thus oxidation of the sheath is a real risk. Laboratory experiments have shown that molybdenum has a somewhat greater oxidation resistance than tantalum and the use of the former is thus preferable on these grounds. However, in respect of ductility, tantalum is very much the better and provides a sheath which is much less sensitive to installation damage. The relative importance of these two criteria is not easy to assess, particularly as, in one irradiation experiment a tantalum sheathed thermocouple cable operated satisfactorily at $2000^{\circ}C$ for 21 days. In this particular case the tantalum was sprayed with tungsten, which would have affected the oxidation rate. If this were the case (post-irradiation examination has yet to be performed) then a combination of a basic tantalum sheath (for ductility) with a molybdenum or tungsten sprayed coating (for oxidation resistance and compatibility) might provide the ideal solution.

The majority of W-Re alloy thermocouples used in our irradiation experiments have operated at $1600^{\circ}C$. The precautions taken to exclude rig atmosphere from the interior of the thermocouple cable have proved generally satisfactory and operational times of up to 8 months have been achieved. Most failures have occurred before irradiation during installation and assembly when the limited ductility of the W5%Re leg is a serious disadvantage.

4. Alternative materials

No attempt has been made at A.E.R.E. Harwell to establish alternative thermocouple materials to overcome on the one hand the susceptibility to

transmutation damage with the platinum alloys and, on the other hand, lack of ductility of tungsten alloys. A small-scale investigation of Tantalum vs Molybdenum and Niobium vs Molybdenum is being carried out and is reported elsewhere⁽¹⁾. The latter combination would appear to be the most promising on grounds of ductility and irradiation stability, but much work has yet to be done on thermoelectric characteristics.

5. Conclusions

The thermocouple error due to transmutation is very difficult to predict in the general case since it involves the neutron flux distribution along the thermocouple, the temperature distribution and the rate of change of thermoelectric e.m.f. with composition change. Thus for any given couple the error may vary from experiment to experiment even for the same peak neutron flux and temperature. From irradiation experiments, not designed to measure thermocouple errors, in which the major portion of the temperature gradient occurred close to the hot junction well within the high flux region, the error due to transmutation in Pt vs Pt - 13% Rh and W5%Re W26%Re couples would appear to be not more than 5% per 10^{20} nv+ (thermal). It is expected that in the near future a good measure of the transmutation error will be obtained from a new design of high temperature graphite irradiation rig which will allow replacement of complete thermocouple assemblies in the in-pile section. By comparison with fixed thermocouples in the rig the transmutation error of both the fixed and the replaced thermocouples should be easily deduced.

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IRRADIATION EFFECTS ON THERMOCOUPLES^(a)

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IRRADIATION EFFECTS ON THERMOCOUPLES

INTRODUCTION

An extensive program on the irradiation behavior of graphite is being conducted by Battelle-Northwest. To obtain the desired data several types of experimental capsules⁽¹⁾ are irradiated in various high flux testing reactors. In most of these capsules thermocouples are used to monitor the sample temperatures. Several of these thermocouples have been checked after irradiation to determine the effect of the neutron exposure on the calibration.

DISCUSSION

The thermocouples used and on which post-irradiation calibration data have been obtained are made from the bare thermocouple wire strung into two-hole vitreous alumina, Al_2O_3 , one-inch long insulators. The bead is made by fusing the two wires together at the tip. A connection is made by means of a hermetic plug and receptacle to polyvinyl-insulated thermocouple extension wire for lead access outside the reactor vessel.

Irradiation temperatures vary from 400 to 1200 C and neutron exposures extend to 4×10^{21} nvt, $E > 0.18$ MeV. On termination of the experiment, the capsule is disassembled in a hot cell. The thermocouple bead and about 2 feet of the thermocouple are removed intact for calibration. In general three thermocouples are calibrated from each capsule.

The calibration setup consists of a copper or boron nitride cylinder, 1" diameter by 3" long with 4 holes drilled half-way through the cylinder lengthwise.

Three irradiated thermocouples and one new thermocouple are slipped into the holes. The cylinder is then put inside a laboratory-type tube furnace. The thermocouple and furnace are connected outside the hot cell to a recorder and controller respectively. The calibration runs are made by maintaining the furnace at three discrete temperature levels for 1-hour periods.

Extensive calibration data have been obtained from Geminol N - Geminol P thermocouples since these have found to give satisfactory performance in contact with graphite under temperature conditions up to 800 C. Geminol N-P wire is manufactured by Driver-Harris Company and has the following compositions:

<u>Geminol N</u>	<u>Geminol P</u>
96.95% Nickel	77.9% Nickel
3 % Silicon	20 % Chromium
0.05% Carbon	1 % Niobium
	1 % Silicon
	0.1% Carbon

The calibration data obtained on 45 Geminol N-P thermocouples are presented in Figure 1. Each data point indicates one thermocouple calibrated at that temperature and the numbers indicated several gave the same calibration result. Irradiation temperatures varied from 400 to 800 C and neutron exposures varied from 0.5 to 2×10^{21} nvt, $E > 0.18$ MeV. Over this exposure range the thermocouple calibration shifted downward about 4 percent and it was not possible to define separate calibration curves for the different exposures. The equation of the "least square" line is:

$$\text{True temperature} = 1.044 \text{ indicated temperature} + 7.5^\circ\text{C}$$

Consequently the calibration error at a true temperature of 1000 C amounts to about 50 C. There is apparently a threshold exposure at which the shift in calibration occurs. The exact exposure is not known, but it is probably less than 1×10^{20} nvt as reported for noble-metal thermocouples.⁽²⁾

Calibration data have been obtained on one platinum versus platinum - 10% rhodium thermocouple. These data are presented in Figure 2. A smooth line

is shown since the points varied only ± 5 C from this least squares line. The irradiation temperature at the thermocouple bead was 1100 C and the neutron exposure 3.8×10^{21} nvt. The calibration error is much larger than for the Geminol thermocouples, being about 16% rather than about 5%.

The changes in calibration are most likely due to a transmutation of the elements in the wire as reported for tungsten-rhenium thermocouples;⁽³⁾ however, detailed examination of the wire has not been made. Chemical attack or effects of temperature cycling may also contribute to the calibration change.

These changes in calibration point out that rather large errors can exist in temperatures read out by these thermocouples during service in irradiation capsules. Additional work at both higher irradiation temperatures and exposures are underway.

ACKNOWLEDGEMENTS

Special thanks go to P. W. Mathay of the General Electric Company, Vallecitos Atomic Laboratory for conducting most of the calibration runs on the Geminol thermocouples.

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FIGURE 1
CALIBRATION OF IRRADIATED GEMINOL N-P THERMOCOUPLES

FIGURE 2
CALIBRATION OF IRRADIATED PLATINUM VERSUS PLATINUM-10% RHODIUM THERMOCOUPLES

INSTRUMENTATION AND DATA REDUCTION TECHNIQUES FOR IRRADIATION EXPERIMENTS

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INTRODUCTION

A number of fuel element irradiation experiments are being conducted or planned by Atomics International, in conjunction with the SNAP Program. The experiments are designed to evaluate fuel element performance over a wide range of compositional variables, temperatures, burnups, and burnup rates. They are tandem assemblies, consisting of as many as six separately controlled capsules, and containing up to 17 clad fuel specimens. Display and automatic recording of fuel centerline temperatures, cladding temperatures, neutron flux, and gas pressure within the fuel element are required to provide the desired information. The data must be reduced and analyzed, to provide a two-phase program consisting of:

- 1) "Real-time" experiment surveillance for operational requirements
- 2) Periodic machine computations for correlation with pre- and post-irradiation measurements of the fuel.

In addition to data acquisition and display, the instrumentation system for these experiments must provide a means of controlling the gas mixture which is used for temperature adjustment. To meet the complex requirements, custom solid-state data acquisition and temperature control systems have been designed, installed, and operated at the 100K Area, Hanford, Washington.

TEMPERATURE CONTROL

Experiment temperature control is accomplished by varying the thermal conductivity of the heat path between the fuel specimen and the reactor coolant. This is done by adjusting the ratio of helium and argon gases in an annulus located between the fuel and the coolant, as shown in Figure 1. A simple gas mixture control system, (Figure 2), used in earlier experiments, did not

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provide adequate safety requirements. A satisfactory control system must supply protection from gas supply failure, gas line blockage, gas system component failures, and operator error during manual adjustment, as well as from reactor or experiment over-power operation. Figure 3 is a schematic of the gas flow control system which was developed after consideration of these safety requirements, and Figure 4 shows one of these gas flow modules.

The key component of this system is a pressure differential monitor which detects variations in the pressure of either control gas, as well as variations in gas flow. A change in gas pressure drop, due to these variations, is developed across the variable orifices (NV-1, NV-2), located upstream of the differential gas pressure indicator. Any change in pressure drop across these orifices will result in a change in differential pressure between the two gas supply lines at the indicator, which includes limit switches to shut off the control mixture and to simultaneously initiate a safety purge. The safety purge introduces pure helium into the capsule from a separate, comparatively high-pressure helium supply for a short timed interval. This promptly brings the capsule temperature to a safe level. At the end of this period, a flow of helium from the regular low-pressure helium supply is established automatically, while the argon supply remains shut off.

The operator adjusts the gas mixture by the use of two needle valves (NV-3, NV-4), which must be operated in small steps to prevent differential pressure switch activation of the purge supply. Changes in the gas composition from such adjustments are delayed in the mix tank. This delay provides an overall system response of 8 to 10 min, except for the safety purge, which initiates temperature reductions in a few seconds. Figure 5 is a plot of typical time-temperature response of the experiment to various operational changes in the gas flow system, based on empirical information.

DATA ACQUISITION

Data acquisition is obtained by the use of custom designed solid state data loggers with punched paper tape output. High/low temperature alarms, as well as the command logic for the gas control modules, are provided. Figure 6 is a block diagram of the system. Figure 7 shows one of these systems, with the data logger in the left rack and eight gas flow modules on the right. The

strip chart recorder is used for temperature trending by the operator, and serves as backup to the paper tape. Figures 8, 9, and 10 show details of the equipment. The analog to digital conversion method is a modified reference voltage ramp type, which includes linearizing for the emf to Chromel-Alumel relationship. The digital output, both visual and tape, is 0 to 1998°F; and it is presented as 1/2 of the value (3 digits, 000 to 999), to provide a higher packing density on the paper tape. A unique three-dimensional plugboard provides versatile programming of high/low limit values and mode control of each thermocouple channel, with provisions to "skip" or "force-to-000" on each channel.

DATA PROCESSING

The punched paper tape is initially processed to obtain a tab listing, to allow manual scanning of the data for improper or erratic operation. Figure 11 illustrates the steps involved in obtaining experiment surveillance information at a weekly rate. A program is being written to obtain final output information in various forms, such as: time-at-temperature histograms, CRT plots, tab listings, etc., which will provide information concerning fuel burnup, temperature profiles, and swelling.

THERMAL CONDUCTIVITY

<u>GAS</u>	<u>@ 1300°F (Btu/hr - ft°F)</u>
HELIUM	0.196
ARGON	0.025

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Figure 1. Temperature Control Concept

7568-15121

Figure 2. Basic Binary Gas Mixture Control System

7568-15116

Figure 3. Improved Binary Gas Mixture Control System

7568-15126

Figure 4. Gas Flow Module

7568-15125

Figure 5. Typical Time-Temperature Response of Irradiation Experiment to Changes in Binary Gas Mixture Control System

7568-15123

Figure 6. Block Diagram of Data Acquisition System

7568-15129

Figure 7. Data Acquisition System Console

Figure 8. Master Panel

7568-15130

Figure 9. Logic Drawer

7568-15127

Figure 10. Analog to Digital Conversion Board

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Figure 11. Data Processing Flow Diagram

THERMOCOUPLE DEVELOPMENT FOR USE IN
MEASURING OXIDE FUEL TEMPERATURES
IN-PILE ABOVE 4000°F

by

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Work performed for
Southwest Atomic Energy Associates for Transmittal to
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ABSTRACT

Investigations are under way to determine the performance of W3Re/W25Re and W5Re/W26Re thermocouples up to 4700°F for up to 1800 hours in (Pu, U)O₂ fuel. Bare wire thermocouples were calibrated up to 5400°F and thoriated, W26Re sheathed thermocouples were calibrated to 5100°F. The compatibility of W26Re and Ta with mixed oxide was studied up to 4700°F.

INTRODUCTION

It is necessary to measure fuel temperatures under both steady-state and transient operating conditions in SEFOR (Southwest Experimental Fast Oxide Reactor) to aid in the correlation and evaluation of reactor parameters, primarily the Doppler effect. SEFOR is a 20 MW(t) fast spectrum reactor fueled with PuO₂-UO₂ and cooled with sodium. It will be used to obtain physics and engineering data at fuel compositions, temperatures, and crystalline states characteristic of power reactor operating conditions. The fuel, and hence, the temperature sensors, will be subjected to a maximum reactor exposure of 6×10^{20} nvt (>1 MeV) in a PuO₂-UO₂ environment with temperatures up to 4700°F for an in-core life of approximately 3 years at 7% load factor.

Based upon a review⁽¹⁾ of candidate sensing devices, it was concluded that thermocouples would best meet the requirements for SEFOR fuel temperature measurements. Because of the relatively broad base of available tungsten-rhenium alloy technology, the reference thermocouple design was specified to utilize an 1/8-inch O. D. tungsten-26% rhenium sheath and tungsten-rhenium alloy thermoelements. Thoriated insulation was chosen over beryllia because of its higher melting point (m. p. (ThO₂) = 5970°F vs. m. p. (BeO) = 4460°F) and its reportedly better high temperature compatibility with tungsten-rhenium.

TUNGSTEN-RHENIUM ALLOY WIRE CALIBRATION

The absence of reliable calibration data for both tungsten-3% rhenium/tungsten-25% rhenium (W3Re/W25Re) and tungsten-5% rhenium/tungsten-26% rhenium (W5Re/W26Re) thermocouple alloys above 4200°F warranted bare wire calibration studies up to the 5000°F range for use in SEFOR. Figures 1 and 2 show the emf vs. temperature relationships for these two thermocouple combinations between 3000°-5400°F.⁽²⁾ These data show agreement with the manufacturer's published calibrations^(3, 4, 5) up to their maximum values (4200°-4350°F). Stability tests conducted for 14 hours at 4700°F indicated no detectable drift of either thermocouple combination.

THORIA INSULATED THERMOCOUPLE PERFORMANCE

Thoria insulated thermocouples incorporating 1/8-inch diameter W26Re sheaths, vitrified thoria insulation and 0.010 inch diameter W3Re/W25Re thermocouple wire were fabricated and calibrated to 5100°F with both insulator purity and hot zone immersion depths as experimental parameters⁽⁶⁾. Figure 3 shows the percent deviation from true temperature indicated by these thermocouples as a function of temperature. The increasingly negative deviation above 4200°F is due to a reduction in the high temperature electrical insulating properties of thoria. However, it should be noted that the thermocouple emfs continually increase with increasing temperature, are reproducible upon thermal cycling, and are insensitive to thoria purity between 99 and 99.9% and hot zone immersion depths between 1 and 3 inches. Thus, a reliable calibration curve, such as the one shown in Figure 4, may be generated for thoria insulated thermocouples up to at least 5100°F.

SHEATH/FUEL COMPATIBILITY

Compatibility tests between mixed oxide fuel typical of that to be used in SEFOR (0.17 Pu-0.83 U)O₂, tungsten-26 rhenium, and tantalum were conducted to aid in determining expected in-core thermocouple life.⁽⁶⁾ Metallographic examination of a W26Re capsule containing mixed oxide and heated to 2700°F for 522 hours indicated no attack. Comparative tests on a Ta and a W26Re capsule, each containing mixed oxide and heated to 4700°F for 11 hours, indicated gross attack of the Ta and no attack but some slight cracking of the W26Re. An additional W26Re/mixed oxide capsule heated to 4700°F for 38 hours indicated localized grain boundary attack in the W26Re.

Since the vacuum furnace testing conditions at 4700°F are not the same as the inert gas atmosphere in an operating fuel rod, these results can only be taken as indicative that sheath/fuel compatibility problems may exist after a few hours of full power operation in the highest temperature regions of the fuel. Furthermore, W26Re sheathed thermocouples used to measure mixed oxide fuel temperatures in-pile at 4200°F indicate no compatibility problem after 300 hours.⁽⁷⁾

CONCLUSIONS

1. Both W3Re/W25Re and W5Re/W26Re thermocouple alloy combinations are capable of measuring temperatures to 5400°F.
2. Vitrified thoria insulated, 1/8-inch O. D. W26Re sheathed, 0.010-inch diameter W3Re/W25Re thermocouples are capable of reproducibility measuring temperatures to 5100°F.
 - (a) Thermocouple accuracy is independent of commercially produced thoria insulator purities between 99 and 99.9%.
 - (b) Thermocouple accuracy is independent of thermocouple immersion lengths between 1 and 3 inches into isothermal hot zones.
3. W26Re sheathed thermocouples are compatible with (Pu₂U)O₂ fuel at 2700°F for times in excess of 500 hours and at 4200°F for at least 300 hours. Compatibility may be limited to a few hours at 4700°F.

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FIGURE 1 CALIBRATION DATA FOR W3R/W25R THERMOCOUPLE IN VACUUM

FIGURE 2 CALIBRATION DATA FOR W5R/W26R THERMOCOUPLES IN VACUUM

NONDESTRUCTIVE TESTING OF
THIN SODIUM ANNULI TO DETECT VOIDS

by

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INTRODUCTION

In the fabrication and inspection of capsules to be irradiated in the EBR-II, we were required to develop a nondestructive testing method to find voids in the sodium annulus between the fuel rod clad OD and capsule wall ID. A void-free annulus is desired for adequate heat transfer across the gap from the fuel to the sodium coolant. X-rays of two capsules are shown in Fig. 1.

Several methods of testing were investigated: X-ray, ultrasonic, and eddy current testing. The eddy current testing showed the most promise and indeed worked out to be a successful method.

The Magnetic Analysis Corporation of Long Island City, New York, developer and manufacturer of eddy current testing equipment, worked with UNC in the development of a test method.

TEST PROCEDURE

The eddy current test procedure developed utilizes a constant-rate longitudinal traverse of the capsule with an eddy current probe, energized by a sine wave at the test frequency, while the capsule is rotated at a constant speed. As the capsule rotates and the probe advances, inspection occurs along a helical path of 0.01-in. pitch. The probe is excited by eddy currents of a fixed frequency. Voids in the sodium annulus cause changes in the detection system which are visually

displayed on an oscilloscope screen and are recorded simultaneously on a strip chart. A capability of detecting voids as small as 3/64-in. diameter was developed.

Two test frequencies are used to verify the existence of annulus voids: (1) a relatively low frequency (60 kc) which can penetrate through the capsule wall to a depth into the sodium annulus, and (2) a high frequency (415 kc) which has less penetrating power but is more sensitive to discontinuities in the capsule shell. By analyzing the results obtained from test runs at the two frequencies over the same capsule area, it is possible to locate sodium voids and to estimate their magnitude.

Testing at both frequencies was performed with the capsule rotating at 330 rpm and the eddy current probe in contact with the capsule shell. Under certain unusual conditions, which will be described subsequently, rotational speed was increased to 1800 rpm and an adjustment of the probe was made so that it no longer was in contact with the capsule. By means of this technique, designated as "air-riding" by Magnetic Analysis Corporation, the influence of surface dents on the probe is eliminated, assuring that defect indications are due to actual material or fabrication faults.

Due to physical limitations of the small bench lathe which was used in the testing process to rotate the capsule, it was necessary to inspect the capsule in sections, relating the strip charts to the inspected sections by means of match marks, denoted as "start," on the charts. In translating defect indications from the strip charts of the 330-rpm tests to the capsule surfaces, a conversion factor of 1.8 was used. This is the ratio of tape length to capsule length traversed. To assure reliability, each capsule section test was repeated to obtain a "check" chart.

The development of the nondestructive testing program can be divided into the following steps:

1. Inspection of simulated capsules and correlation with eddy current traces
2. Fabrication and inspection of a defect standard against which the final assembled rods would be compared
3. Inspection of actual fuel capsules.

Four simulated (dummy) capsules were fabricated: two in strict accord with the procedure established for the irradiation capsules, and two with intentionally "planted" voids in the sodium annulus.

All four capsules were eddy current tested at the 60-kc frequency. On the basis of defect indications recorded on the strip charts, two capsules were selected for destructive examination. The capsule shell was peeled away to permit photographing underlying voids, which then were related to the defect indications on the strip charts. From the conclusions derived from this procedure, one of the remaining dummy capsules (containing "planted" voids) was selected as a "Defect Standard." Because the process is basically a comparative one, this standard was used in all subsequent eddy current testing of the irradiation capsules to calibrate the equipment.

The traces from one of the two dummy capsules containing intentional defects along with the corresponding eddy current signal are shown in Figs. 2, 3, 4, and 5. Figs. 6 and 7 show traces of the Defect Standard.

As shown in Fig. 5, the smallest void found was 3/64-in. diameter and was clearly indicated on the test trace.

SURFACE DEFECTS

The eddy current charts of Capsules 78 (Figs. 8 and 9), and 79 (Figs. 10 and 11), indicated possible voids in the sodium annulus. It was noted that the peaks were sharp in contrast to the rounded peaks seen in Figs. 6 and 7. At each point of the

capsules that produced a defect signal, visual inspection disclosed a defect in the surface of the capsule tube (see Fig. 12). Further testing proved that the signal was generated due to the surface defect and not because of a void in the sodium annulus. The additional testing consisted of:

1. Attaching a Strobotac into the eddy current probe circuit so that it would be activated by the defect signal. With the probe traverse performed manually and the capsule rotating at 300 rpm, the Strobotac was only activated when the probe shoe rode over the visible surface defect. Moving the probe slightly to either side of the defect deactivated the Strobotac.
2. Testing the capsules at a frequency of 415 kc and with the eddy current probe not in contact with the capsule (air riding). This mode of testing proved capable of locating voids in the sodium annulus and was not affected by surface defects.

Capsules 78 and 79 were tested using this method, and no defect indications appeared. This proved that the original defect traces were generated by surface imperfections and not by voids.

The validity of this testing procedure can be seen from Fig. 13 (a, b, c, and d). Fig. 13 (a and b) shows traces of voids in the Defect Standard capsule. Fig. 13 (a) was a test made at the standard conditions of 60-kc, 330-rpm capsule rotation and a probe traverse of $3\frac{1}{4}$ in./minute.

Fig. 13 (b) was a test at 415 kc, air-riding. This test is made at a faster rotation and probe traverse speed, hence the chart length is shortened. Corresponding defects in Fig. 13 (a and b) are apparent.

Fig. 13 (c and d) shows tests of a suspect area of Capsule 79. The test conditions of Fig. 13 (a and c) are identical, as are those of Fig. 13 (b and d). It can be seen that the defect signal shown in Fig. 13 (c) is not

present in Fig. 13 (d) demonstrating that the defect signal was generated by a surface imperfection.

SUMMARY

Correlation between the eddy current strip charts and the destructively tested simulated capsules was excellent. For every indication on the eddy current charts we were able to find either a void in the sodium or a surface imperfection on the capsule OD.

By modifying the operating conditions, surface imperfections can be distinguished from voids in the sodium annulus.

The inspection method is reproducible as proved by identical charts of a standard run before and after every capsule inspection.

The following procedure is recommended for eddy current inspection of EBR-II fuel capsules.

1. Eddy current test the capsule tubing at 60 kc and 415 kc with the probe in contact with the tube surface to ascertain that no defects exist prior to assembly of the capsule.
2. Eddy current test the completed capsule at 60 kc with the probe in contact to inspect for sodium annulus defects.
3. Rerun Test 2 at 415 kc. If defect indications persist, defects are bubbles in the sodium annulus or faults due to damage suffered by the capsule tubing during the capsule fabrication process. In resolving defect indications, an air-riding run at 415 kc should be performed to eliminate indications due to minor surface imperfections which may have occurred during capsule fabrication.

4. If no defect indications are observed at either the 60-kc contact run or 415-kc contact run, the capsule assembly can be considered acceptable.

ACKNOWLEDGEMENTS

The authors wish to thank Mr. Joseph Callan and Mr. Edward Speirer of Magnetic Analysis Corporation for their assistance and suggestions in developing the eddy current inspection technique, and Messrs. Donald Rosh, Stanley Hurwitz* and Millard Snyder† for their assistance in development of the capsule assembly procedure and preparation of the standard capsules.

*Presently with IBM.

†Presently with General Electric, Vallecitos.

Fig. 1 — Radiographs of Capsules 78 and 79

Fig. 2 — Correlation of Voids in Sodium Annulus of Dummy Capsule with Eddy Current Trace — Defects: $5/32 \times 5/32$ in., $1/16 \times 1/16$ in.

Fig. 3 — Correlation of Voids in Sodium Annulus of Dummy Capsule with Eddy Current Trace — Defects: $1/16 \times 1/16$ in.

Fig. 4 — Correlation of Voids in Sodium Annulus of Dummy Capsule with Eddy Current Trace — Defects: $1/16 \times 1/16$ in. (left), $1/8 \times 1/16$ in. (right)

Fig. 5 — Correlation of Voids in Sodium Annulus of Dummy Capsule with Eddy Current Trace — Defects: $3/64 \times 3/64$ in.

Section A

Section B

Fig. 6 — Eddy Current Traces of Defect Standard Capsule (lower half)

Section A

Section B

Fig. 7 — Eddy Current Traces of Defect Standard Capsule (upper half)

Fig. 8 — Eddy Current Traces of Capsule 78 (Lower Half)

Section A

Section B

Fig. 9 — Eddy Current Traces of Capsule 78 (upper half)

Fig. 10 — Eddy Current Traces of Capsule 79 (lower half)

Section A

Section B

a. Capsule 78 6×

b. Capsule 78 6×

c. Capsule 78 6×

d. Capsule 79 6×

Fig. 12 — Capsules 78 and 79 — Surface Defects

a. 60-kc Test of Standard Capsule
Showing Sodium Void Indications
(Probe Riding on Surface of Capsule)

b. 415-kc Test of Defect Standard
Capsule (Probe Air-Riding)

c. 60-kc Test of Capsule 79 Showing
Suspect Defect Indication (Probe
Riding on Surface of Capsule)

d. 415-kc Test of Capsule 79
(Probe Air-Riding)

Fig. 13 — Eddy Current Traces — Verification of Surface Defect Signals

VIII. GENERAL TEST REACTOR ACTIVITIES

**BURST IRRADIATION FACILITIES
AT THE
NATIONAL REACTOR TESTING STATION
AND A REVIEW OF THE
SUBASSEMBLY TESTING PROGRAM**

For over a decade, Phillips Petroleum Company, under auspices of the Atomic Energy Commission, has conducted a program of reactor safety tests at the National Reactor Testing Station in Idaho. These tests have been of an integral-core type wherein the entire reactor core is subjected to conditions which simulate theoretical accidents. Primarily, the tests have been of the power transient type, initiated by stepwise insertions of reactivity, to simulate the well-known reactivity accident such as occurred with the SL-1 reactor in 1961.

Tests of this type have been conducted on several different reactor types with reactivity insertions up to 3\$, and more, in order to determine both the extent to which each reactor type is capable of withstanding such accidents and the process in which the reactors effect self-shutdown. It was found, generally, as the amount of reactivity inserted for each test was increased, the potential for injury to the reactor also increased. For each of the reactor types studied thus far, threshold reactivities insertions were found below which no damage to the core occurs and above which damage arises and progressively worsens.

It became clear from these programs that although full-system destructive tests are necessary to confirm the expected behavior of integral reactor systems under accident conditions, the study of destructive processes could be more efficiently undertaken first in small-scale capsule or sub-assembly tests. Capsule tests, it was reasoned, should be performed in which small quantities of the fuel in question would be subjected to conditions duplicating those in a reactor undergoing an accident. The advantage of the capsule approach is obvious. Instrumentation can be more refined and specific to the phenomena to be observed. Destruction of data channels can often be avoided; and, compared to full-system tests, cost and time are greatly reduced with the added advantage that greater statistics can be gathered when needed.

1. THE SUBASSEMBLY TEST PROGRAM

As a consequence of these conclusions, an in-pile subassembly testing program has been initiated at the NRTS. Specific objectives of the Subassembly Program are, to a large degree, directed toward obtaining data applicable to the analysis of those severe accidents in power reactor systems which may result in the ultimate release of fission products to the environment. Emphasis will be placed on experiments which study (a) the causality of fuel element failure, (b) interactions and reactions between the fuel element and its environment that are important in determining the severity of an accident, and (c) the nature of release, transport, and deposition of fission products following failure of the initial containment in the fuel element.

Experiments investigating the causality of fuel element failure will allow a determination of the effects and relative importance of various conditions (eg, heating rate, fission gas content, and cladding physical properties) on the

failure of fuel cladding by such various modes as stress rupture from internal pressure or melting from insufficient heat transfer. Interaction and reactions between the fuel element and its environment will be studied to obtain such data as the rate and extent of metal-water reactions which may contribute additional energy to an accident and the conversion ratio of thermal-to-mechanical energy which may occur as the result of rapid steam formation due to large increases in heat transfer rates and/or heat transfer area during an accident. The relative amount of individual fission products released and their chemical state and particle size, as a function of given accident conditions, will be among the important items in the area of fission product studies.

It is planned, as a part of the Subassembly Program, to simulate accident conditions which range from the rapid heating rates typical of excursions which may cause fuel failure by stress ruptures to the slow heating rates typical of a loss-of-flow or loss-of-coolant accident which may result in fuel failure from clad melting. These require that the samples be subjected to neutron burst irradiations with initial periods as short as one to two milliseconds or reductions of heat removal capabilities with the sample initially being operated at conditions (eg, power density, temperature, pressure, and flow) ranging from those typical of initial loading to end-of-life power operation. Specifically, the following will be studied:

- (1) Fuel element failure thresholds and failure mechanisms
- (2) Heat transfer processes and physical characteristics bearing on failure mechanisms
- (3) Chemical reactions which can yield additional energy
- (4) The pressure wave shock phenomena and water hammer resulting from thermal-to-mechanical energy conversion
- (5) The type and amount of fuel dispersed and fission products released by failure.

In the Subassembly Test Program, the fuel element types to be studied will, generally, be those which have proved satisfactory or which show considerable promise for future development based on previous operating experience and/or on experiments performed under steady-state conditions. Two major types which meet these requirements, at present, are (a) rod-type fuel elements of UO_2 pellets or powder which are clad with zirconium or stainless steel and (b) highly enriched, metallic, plate-type elements which are clad with aluminum, zirconium, or stainless steel.

It is expected that in addition to these "proven" fuels, testing of advanced or prototype fuels will be desirable. In this way, it is hoped that the development of safety information about specific fuels can occur earlier in the course of each fuel concept and diminish a historical lag between the two, which has compounded problems of safety analysis.

2. BURST IRRADIATION FACILITIES REQUIREMENTS

The irradiation services required for subassembly capsule tests of the type just described are decidedly different from the conventional. For example, rapidly changing flux histories similar to those which might exist in a reactor undergoing a reactivity accident are required rather than the usual, well-controlled steady-state flux -- this means a high nv burst with the rise time (or period) of the burst in the region of a few milliseconds, possibly even into the sub-millisecond region. The burst nvt must be sufficient not only to melt the selected fuels but, in some cases, to cause vaporization within a few milliseconds. While, in some cases, it may be possible to induce these high-energy releases in fuel samples by simply increasing the U-235 enrichment of the samples, this artifice is not always successful since it may create undesirable flux and temperature distributions.

A further demand upon burst irradiation services for safety studies is that of size. It is desirable, and in some cases considered necessary, that full-size subassemblies of fuel, as built for reactor service, be subjected to burst irradiation experiments. The reasons for large experiments are twofold. First, when coupons or miniature samples of fuel are used, scaling "laws" are tenuous. Relating measured effects, such as pressure and energy conversion around small samples, to similar effects in a full-scale reactor system is difficult at best and liable to suffer in credibility. Second, fuel responses during rapid heatup are often size- and geometry-dependent and must be tested under conditions as closely duplicating the full-scale reactor as possible. For example, transient temperature distributions, meltdown patterns, interaction of melting and distorting fuel, metal-water ratios, and other parameters which may be important to the gross damage of a reactor must be maintained as near to realistic dimensions as possible.

Safety itself also becomes a design problem with a burst irradiation facility. Explosive mechanisms are often the object of study, and a burst facility must be protected against damaging forces which might arise in the experiments. Explosionproof shielding must be incorporated between the experiments and the reactor core. Within the current safety requirements for reactors and within the present status of ignorance about the destructive processes and forces to be encountered, design safety of a burst facility can be extremely complex.

Of greater safety import, though, is the transient response of the reactor itself. It appears obvious from the requirements set forth that the "driver" core supplying neutrons for these experiments will be pushed to its safe limit to provide the highest nvt possible. This means that energy releases and fuel temperatures will be as high as possible short of damaging the core and also that safe, positive shutdown must be assured following each transient.

Two reactors have been conceived and designed to satisfy both the requirements of the proposed Subassembly Testing Program and the necessary safety requirements. The principal reactor is the Power Burst Facility which was designed specifically for this work. To serve in the interim while the Power Burst Facility is being constructed, a simpler and less capable reactor, the Capsule Driver Core has been built. These reactors are described in following sections.

3. THE POWER BURST FACILITY

Following an extensive analysis by Phillips Petroleum Company, it was determined that an oxide-fueled reactor could best satisfy the service and safety requirements discussed. This analysis led to the design of the Power Burst Facility, or PBF[1, 2], which has now been designed completely (except for certain details of the fuel rods), and construction is underway. Criticality is expected in 1968, and irradiation service should be available early in 1969.

The Power Burst Facility is an oxide-fueled, epithermal, water-moderated reactor designed expressly for operation in the transient mode at reactor periods down to 1 msec. The parent core serves as a driver for fueled experiments located centrally within the core. Power peaking within the experimental fuel samples is obtained primarily by utilizing the thermal flux trap principle and, whenever possible, by sample enrichment. A heavy-walled tube, termed "the in-pile tube", provides the primary containment for the experiment and isolates the parent core from the effects of pressure generated within the experiment. This tube may be either a sealed capsule or the in-pile section of a test loop. The tube, in combination with a permanent stainless steel flux suppressor, minimizes neutronic coupling from the experiment to the parent core. The core and flux suppressor are shown in plan view in Figure 1.

The reactor is capable of limited steady-state operation at power levels up to 20 MW. Steady-state operation for short periods (less than 1 hour) may be used to provide power and temperature distributions within the experiment typical of normal operating conditions specified for the experimental fuel sample. Steady-state operation may also be used in simulating flow channel blockage accidents. Self-limiting power excursions with initial periods as short as 1 msec may be initiated from zero power. Excursions with periods as short as about 3 msec may also be initiated from a 20-MW power level to accomplish destructive testing of experimental fuels under simulated system operational conditions.

3.1 Facility Description

As with other Spert reactors, the PBF reactor will be remotely operated from the control center, 0.5 mile from the reactor building.

The reactor building, shown in Figure 2, houses the reactor tank, canal, auxiliary equipment and machinery, mechanical setup and working areas, and personnel convenience areas. It consists of the reactor bay flanked by low wings on either side above grade and two basement levels below grade. The main floor is the working and control area for nonnuclear plant operations. Wings on either side of the reactor bay are divided into instrument and process control rooms, working areas, and personnel convenience areas. The first basement contains most process equipment, eg, the pumps, heat exchangers, resin columns, etc. The equipment associated with experimental loops is housed in shielded loop cubicles to provide radiation shielding following nuclear operation. The second basement contains the major piping to the reactor primary coolant system and the warm and hot waste room.

The reactor tank, shown in Figure 3, is of open-top construction and fabricated of Type 304 stainless steel. The cylindrical tank section is 28 ft 9 in. deep and has an inside diameter of 15 ft. A 6-ft-wide by 10-ft-deep removable

Figure 1. Plan View of the Power Burst Facility Core.

Fig. 3 Elevation of the PBF Vessel.

gate at the top of the tank is provided to allow underwater transfer of radioactive materials between the tank and an adjacent canal.

3.2 Core Description

In the design of a burst irradiation facility, it is essential that the fuel selected possess a high heat capacity and that the core possess a prompt mechanism for reactor shutdown to terminate power bursts without dependence upon mechanical scram systems. Urania (UO_2) possesses both the desired heat or energy capacity (enthalpy) and the prompt shutdown mechanism in the form of Doppler broadening of the resonances in U-238 and was chosen as the basic fuel for PBF. However, pure urania with an enrichment of the order required for a reasonable system would have had a U-238 content far in excess of that required to provide the Doppler feedback desired for the design transient (ie, $\tau = 1$ msec). To achieve the appropriate U-238 content and, hence, the desired reactivity feedback, part of the urania was replaced by calcia-stabilized zirconia ($\text{ZrO}_2\text{-CaO}$), and the proper U-235 concentration was preserved by additional enrichment. The stabilized zirconia content required to give the proper Doppler feedback is about 60 percent by volume. The enrichment of the uranium is about 11 atom percent. The stabilized zirconia has a volumetric heat capacity equivalent to that of urania so that the high enthalpy of the core was preserved.

Since the PBF design criteria require reactor periods of the order of 1 msec and peak fuel temperatures of the order of 2350°C , it is necessary to provide protection for the fuel cladding against thermal shock by reducing both the fuel-clad interface temperature and the temperature drop across the clad. An annular layer of stabilized zirconia, nominally 55 mils thick, has been incorporated in the design of the PBF fuel rods, as shown in Figure 4, to provide the desired thermal protection of the cladding. The rods are fabricated with a nominal 1-mil interference fit between the zirconia sleeve and the cladding. A 12-mil

FIG. 4 CROSS SECTION OF A PROTOTYPE PBF FUEL ROD.

radial gas gap between the fuel and the zirconia sleeve allows for expansion of the ternary pellets during the burst. The fuel pellets are nominally 0.56 inch in diameter and 1 inch long. The clad is a Type 304L stainless steel tube with a wall thickness of 28 mils and an outside diameter of 0.75 inch. The calculated temperature distribution within the fuel rod for a 1-msec transient is shown in Figure 5.

FIG. 5 TEMPERATURE DISTRIBUTION IN PBF FUEL ROD FOLLOWING A 1-MSEC TRANSIENT.

The basic fuel rod is 36 inches in active length and contains 36 fuel pellets plus an equal number of insulator sleeves. Helium gas, at 1 psig, is introduced in the rod during fabrication to provide an inert atmosphere within the rod. Zirconia insulator discs are provided at both ends of the fuel rod to restrict the rate of heat transfer, and a spring is provided at the top end to accommodate axial expansion of the fuel.

The PBF core is built around the experiment hole and flux suppressor, as shown in Figure 1. All experiments will be placed within the experiment hole. The maximum outside diameter of the in-pile tube is about 8.5 inches. The thickness of the tube wall is a function of the nature of the experiment being tested. A typical water-cooled experiment may require wall thicknesses up to about 1 inch, leaving a tube inside diameter of approximately 6 inches. Experiments up to 4.5 inches in diameter and 36 inches long are contemplated. Detailed design of the in-pile tube will vary for particular experiments, but the primary design requirement is that the tube shall effectively protect the parent core from pressure pulses generated in the experiment during destructive testing of fueled specimens.

A summary of pertinent characteristics of the PBF core is given in Table I. From the data presented, it will be evident that the PBF reactor is under-moderated as a consequence of the close packing of the fuel rods. The core is

TABLE I

CORE CHARACTERISTICS FOR THE PBF REACTOR

Reactor

Reactor Type	open pool
Operating Power	
Steady-state	20 MW (th)
Transient	≈ 350 GW (th)
Transient Energy Release	
1-msec power excursion	2000 MW-sec
Shaped power bursts	< 3000 MW-sec
Core Diameter (overall)	52 in.
Active Core Height	36 in.
Number of Fuel Rods	≈ 2600
Fuel Rod Characteristics	
Rod OD	0.75 in.
Pellet diameter	0.560 in.
Zirconia sleeve thickness	55 mils
Clad thickness	28 mils
Clad material	Type 304L stainless steel
Nominal helium gap	≈ 11 atom percent in feed material
Fuel enrichment, U-235	≈ 40 percent urania by volume
Fuel composition	≈ 60 percent calcia-stabilized zirconia by volume
Fuel Temperature	
Steady-state	1200°C peak 700°C average
Transient	2350°C peak ≈ 1600°C average
Core Enthalpies	
Fuel at design conditions	2000 MW-sec
Zirconia sleeve at melting point	≈ 1400 MW-sec
Stainless clad at melting point	≈ 500 MW-sec
Poison Rods	
Control rods	
number	8
type	cylindrical
poison	natural boron carbide in stainless steel

TABLE I (Contd.)

CORE CHARACTERISTICS FOR THE PBF REACTOR

poison OD	2.75 in.
poison thickness	0.125 in.
poison length	36 in.
central region	contains polyethylene sleeve
rod drive	electrical withdraw or insert; compressed-air scram
rods per drive	1
cooling supply	air
follower	none
Transient rods	
number	4
type	cylindrical
poison	natural boron carbide in aluminum clad
poison OD	2.75 in.
poison thickness	0.375 in.
poison length	58 in.
central region	contains air
rod drive	compressed-air fast ejection; hydraulic positioning
rods per drive	1
cooling supply	air
follower	aluminum
Primary Coolant	
Pressure	atmospheric
Volumetric flow	15,000 gal/min (maximum)
Flow rate in core	6.8 ft/sec (average)
Inlet Temperature	
transient operation	86°F
steady-state operation	97°F
Outlet temperature	
transient	96°F
steady-state (average)	107°F
Cold pressure drop	
cold	3 to 5 psi
hot	25 psi(maximum)
Heat Transfer	
Active surface area	1512 ft ²
Heat flux -- steady-state	
average	50,000 Btu/hr-ft ²
maximum	100,000 Btu/hr-ft ²

TABLE I (Contd.)

CORE CHARACTERISTICS FOR THE PBF REACTOR

Power Density	
Steady-state (20 MW)	
core	16 kW/liter
fuel	53 kW/liter
Transient (peak $\approx$ 350 GW)	
core	280 MW/liter
fuel	930 MW/liter
Core peak-to-average power distribution	
	2.0
axial	$\approx$ 1.4
radial	$\approx$ 1.4
Core Lifetime	
	1000 transients at 1 msec
Facility Lifetime	
	10 years ($\approx$ 5000 cycles)
Reflector	
	stainless steel plus water
Core Composition by Volume	
Ternary-oxide fuel	0.31
Zirconia insulator	0.15
Water	0.30
Structural material (steel and Al)	0.24
Eigenvalues and Reactivity	
Coefficients (water-filled experiment hole)	
$k_{eff}(\text{cold})$	
all rods in	0.925
control rods out (TR in)	0.994
transient rods out (CR in)	0.977
all rods out	1.040
Control rod worth	
transient rods in	10.7\$
transient rods out	8.8\$
Transient rod worth	
control rods in	8.3\$
control rods out	6.4\$
Average moderator temperature coefficient of reactivity	
	-2.6¢/°C

TABLE I (Contd.)

CORE CHARACTERISTICS FOR THE PBF REACTOR

Average moderator void coefficient of reactivity	-5.0 ϕ /percent void
Average Doppler reactivity feedback (in fuel)	$\approx -0.3 \phi/^{\circ}\text{C}$
<u>Reactor Tank</u>	
Inside Diameter	15 ft
Depth	28.75 ft
Cylindrical Steel Thickness	0.5 in. (minimum)
Material	Type 304 stainless steel
Design Pressure	
Above canal gate	hydrostatic head
Below canal gate	hydrostatic head plus 25 psi
Design Temperature	200°F
Capacity when full	36,800 gal

properly classed as epithermal (cadmium fission ratio ≈ 3.5), and the neutron density is high. The high (2000 MW-sec) enthalpy of the fuel at the design condition, coupled with the high enthalpy of the zirconia sleeves and stainless clad (in excess of 1400 and 500 MW-sec to their respective melting points), provide a core well suited to the generation of large quantities of neutrons during short-period power bursts.

3.3 Irradiation Performance of the PBF

A measure of the performance capability of the core for a particular experiment may be obtained by comparing the power (or energy) densities in the experiment and the parent core. A specific ratio taken at the core midplane compares the planar average density in the experiment to the peak density in the parent core. This ratio, which is defined as the figure-of-merit of the core, is a measure of the gain of the system and indicates the relative ease or difficulty associated with destructive testing of a fuel sample. Since the specific enthalpy (cal/g) of the parent fuel at the peak temperature for the design transient is about 210 cal/g ($\approx 10^4$ joules/cc), a figure-of-merit of two will permit deposition of 420 cal/g in the central region of a test sample. This energy density is sufficient to begin vaporization of a urania fuel specimen. A figure-of-merit of four would provide 840 cal/g which is sufficient to cause complete vaporization of urania. The figure-of-merit is highest for small experiments where flux depression is least and in highly enriched samples where the fission density is greatest.

To illustrate typical performance of the PBF, figures-of-merit and possible enthalpy increases (for the design, 1-msec transient) have been computed for

Yankee-type fuel clusters. Yankee fuel rods consist of pelleted, 3.4 percent enriched UO_2 , clad with 0.021 inch of Type 348 stainless steel. Results of the calculations are shown in Table II.

TABLE II

PBF PERFORMANCE WITH YANKEE FUEL
IN THE DESIGN TRANSIENT (1 msec)

<u>Cluster Size (No. of Rods)</u>	<u>Calculated Figure-of-Merit</u>	<u>Enthalpy Rise of UO_2 (cal/gram)</u>
7	2.6	566
25	2.1	457
64	1.9	414
100	1.8	392

3.4 Experimental Program

In describing the experimental activities to be pursued, it is convenient to establish three categories of experiments: startup tests, loop tests, and capsule tests. This categorization groups together under startup tests the experimental activities associated with the startup and checkout of the reactor, the measurement of its various characteristics and capabilities, and, in general, the overall program of obtaining information from experiments in which the PBF performance is the object of the experiment. This is in contrast to the other two experimental categories; both of which encompass the conduct of tests designed to investigate component behavior under representative environmental conditions simulated in in-pile loops and capsules.

It is implicit in the nature of the overall program that certain portions of the nuclear startup phase must proceed in a particular order. However, in general, no specific schedule commitments are made. It is the intention to pursue the experimental program in such a manner as to maximize the utilization of all experimental and analytical resources. Thus, several simultaneous programs may be pursued concurrently in the facility. For example, a given month of reactor operation might include boiling mechanism studies using aluminum-clad, plate-type fuel, fiducial transients extending a previously demonstrated operating range of the basic facility, studies of meltdown of an assembly of zirconium-clad fuel rods from a specific on-line power reactor, and additional boiling mechanism studies with a single parametric variation from the first series. This type of experimental program scheduling is expected to minimize perturbations to the overall experimental effort which might be expected as a consequence of changes in program, since it will facilitate scheduling and performance of new experiments of a high-priority nature which might be proposed. It is felt that a program attuned to intervals of relatively short experimental operation that are separated by longer intervals in which analysis, planning, and out-of-pile support work can be

pursued will yield benefits far in excess of any loss in productivity which might arise from such a concession to flexibility.

3.41 Startup Tests. Following the initial loading operation and certain "statics" measurements, the first stage of dynamic nuclear operation of the PBF will consist of fiducial transient power excursion tests designed to provide information on the dynamic performance characteristics of the reactor system. These tests will consist of the two basic types of transients: (a) "power excursions" initiated by stepwise insertions of reactivity and generally characterized by a spike-type, power-time response and (b) "shaped power bursts" initiated by stepwise reactivity insertions followed by rampwise reactivity insertions such that the reactivity compensation resulting from the initial portion of the stepwise-induced transient is partially offset by the ramp reactivity addition and such that the reactor power remains at a specific quasi-equilibrium level for a certain time. The shaped power burst is thus intended to approximate a square-wave power time response. Figure 6 shows ideal and probable burst shapes for the PBF.

FIG. 6 IDEAL AND PROBABLE BURST SHAPES.

In the fiducial test programs, it is expected that tests will proceed in the order of increasing severity. In general, this may be interpreted as starting with small reactivity perturbations (ie, relatively long asymptotic reactor periods) and gradually increasing the size of the reactivity insertion such that the reactor period becomes shorter and such that the power, temperature, and pressure

resulting from the excursion increase with a reduction of period. Fiducial transients are planned in the period range extending down to approximately 1 msec, which is the maximum excursion reflected in the facility design criteria.

3.42 Capsule Tests. Capsule experiments are defined as those in which the fuel or test apparatus being studied is placed in a closed container in the experiment hole. The degree of detail in capsule designs varies from simple closed-end pipes which house fuel samples for pre-post type measurements to elaborately instrumented multi-walled devices affording measurement and/or control of such parameters as pressure, temperature, neutron flux, power, and energy.

In the PBF program, capsule tests will be performed to obtain information on a wide range of subjects including: reactivity compensation mechanisms, mechanisms for component failure and/or destruction, metal-water reactions, fission product release studies, and scoping-type data for proposed experimental programs. Studies will include investigations of various types of failure thresholds such as (a) fuel clad bursting, distortion, and discoloration and (b) cracking, fragmentation, and melting of fuel. Other areas of specific interest are determinations of the efficiency for conversion of heat to mechanical energy and studies of heat transfer mechanisms and rates. This latter study area includes investigating the effects of changes in environmental parameters on heat transfer properties.

3.43 Loop Tests. The highest stage of refinement of subassembly-type testing is that in which the environmental parameters of reactor systems under investigation are accurately simulated in the subassembly experiments. In the case of PBF, this degree of sophistication in experimental capability is to be provided by means of in-pile loops.

These loops will be installed in the PBF reactor building in which cubicle space is provided for the out-of-pile components of two separate loop concepts. Pumps, reservoirs, heat exchangers, pressurizers, and other equipment related to operation of a loop will be housed within the cubicle with appropriate loop piping leading to and from the in-pile tube section which will contain the experiment. The in-pile tube section of the loops is located inside of the stainless steel flux suppressor in the experiment hole; and from experiment containment considerations, it is equivalent to the capsule discussed previously.

Several types of loops or accident simulations are possible in the PBF including

- (1) Cold startup reactivity accidents
- (2) Hot standby or operational reactivity accidents utilizing electrical heaters and/or steady-state operation of the parent core to simulate the appropriate temperature distributions with the loop at the operational pressure and flow conditions appropriate to the reference core
- (3) Fuel destructive testing investigating the energy thresholds for various failure modes and including effects of experiment size and geometry on type of failure, transient pressure generation, metal-water reactions, and fission product release

- (4) Investigations of the effects of burnup on fuel failure utilizing elements pre-irradiated elsewhere and subjected to various accident conditions
- (5) Mechanisms studies investigating the dependence of fuel failure upon the initial flow, temperature, and pressure conditions plus core geometry and size.

Among the basic types of loops being considered are (a) liquid-metal-cooled loops such as NaK and sodium and (b) gas-cooled loops such as helium, CO₂, and air. The types of fuels tested will be limited to uranium fuels. The number and nature of tests with each loop concept will be dependent upon the needs of industry. In general, however, the PBF is capable of performing reactivity accident simulation and fuel destructive tests for a wide variety of fuel and coolant configurations. Thermal neutron filters around the experiment hole may be inserted for some liquid-metal cooled (fast reactor) experiments to obtain a hardened spectrum. In some systems, minor enrichment of the fuel specimens above those specified for typical reactor designs may be required to permit vaporization of the fuel specimen.

In liquid-metal-cooled systems, accidents involving fast-plugging of coolant channels can be readily simulated; whereas, meaningful simulation of partial channel-blockage accidents may be marginal or unfeasible. Within limits, the loss-of-coolant accident can be simulated using the parent core to provide the relatively low power densities appropriate to afterheat and thus permitting initial simulation of such accidents with cold-clean assemblies.

Similar applications could be made for gas-cooled loops by making allowance for differences in fuel concept and thus redefining the appropriate parameters applicable to system failures.

Each system introduced into the facility will possess its own unique characteristics and will, therefore, be expected to leave a different effect on the performance of the reactor. Thus, it will be necessary to examine each experiment with regard to its effect upon the safety of system operation and to ensure that the experiment falls within approved operations.

4. THE CAPSULE DRIVER CORE

Many of the studies to be performed in the Subassembly Test Program do not require the full capacity of the PBF and can be completed, or at least initiated, in a reactor of more conventional design. Thus, in order to initiate the Subassembly Program prior to completion of the PBF, the Capsule Driver Core (CDC) has been designed and constructed utilizing available powdered UO₂ fuel, swage-compressed in stainless steel cladding, which was originally used in the NS Savannah critical experiments^[3]. The CDC is not equipped with extensive environment control in the test space as is the PBF and is more limited in capabilities, particularly energy release, than the PBF; but it will soon be available for general use, probably in the second half of this year. Since integrated fluxes are expected to be sufficient to melt and partially vaporize several uranium fuel types, valuable safety studies in the Subassembly Testing Program can be started.

At the present time, the CDC has completed all phases of loading and static nuclear checkout [4, 5, 6, 7, 8, 9]. Transients have been initiated successfully with periods as short as 4 msec. Periods as short as 2 to 3 msec are considered possible with the CDC, but initial irradiation services will probably not extend below about 5 msec until minor modifications have been made to the core this year.

4.1 Facility Description

The CDC is assembled in the Spert IV facility which is located approximately 1/2 mile from the Control Center building and approximately 3/4 mile from the nearest other reactor facility (Spert III). Nuclear operation of the CDC will be carried out by remote control from the control console located in the Control Center building.

The reactor building, shown in Figures 7 and 8, houses the reactor tank, storage tank, the necessary auxiliary equipment and machinery, mechanical setup and working areas, and personnel convenience areas. It consists of a reactor bay flanked by low wings on either side above grade and a basement level below grade. The core is located within the open-top, north tank which is 20 feet in diameter and 25 feet deep. The south tank is used for fuel and irradiated test material storage. The two tanks are connected at the top by a 6- by 6-foot removable gate shown in Figure 9.

The reactor control drives are located on a control bridge which spans the width of the tank and which is movable, on rails, for the full length of both tanks.

4.2 Core Description

Figure 10 is a cutaway view of the CDC, and Figure 11 is a plan view of the core. A photograph of a top view of the CDC is shown in Figure 12. The core is divided by 12 aluminum rod-guide crosses which house the 8 cruciform-shaped control rod blades and the four cruciform-shaped transient rod blades. A cylindrical, stainless steel experiment tube in which test samples will be placed for kinetic testing lies along the vertical center line of the core.

The fuel rods are six feet long with welded-seam, stainless steel tube clad. They contain low-enrichment UO_2 powder, swage-compressed to an effective density of approximately 85 percent of the theoretical density of UO_2 . The rods contain nominally 1600 grams of UO_2 (three percent enriched) and are clad with a nominal 28 mils of Type 304 stainless steel. Reactor control is accomplished by the use of four control rod units, each consisting of two cruciform-shaped, neutron-absorbing blades connected by a yoke assembly and shaft to the armature of a coupling magnet. The four transient rods, which are ganged and used for step insertions of reactivity, are similar to the control rods except that their poison sections are normally below the active core region. The poison sections of the control and transient rods are 12 weight percent natural boron as boron carbide in an aluminum matrix and have follower sections which are constructed of aluminum.

The experiment tube, which extends the length of the core, consists of a six-foot-long, Type 304 stainless steel cylinder with inner and outer diameters of 3.438 and 4.500 inches, respectively. The tube, which has a welded end-plate on the bottom with a two-inch hole to allow for possible flow service to cool the capsule, is attached to the lower grid support.

A summary of CDC characteristics is found in Table III.

Fig. 7 The CDC (SPERT IV) Reactor Building - Exterior Cutaway View.

FIG. 8 FLOOR PLAN OF THE CDC (SPERT IV) REACTOR BUILDING.

8.1.19

FIG. 9 ELEVATION DRAWING OF THE CDC (SPERT IV) REACTOR BUILDING.

FIG. 10 CUTAWAY VIEW OF THE CDC AND CORE SUPPORT STRUCTURE.

FIG. 11 CROSS-SECTION PLAN VIEW OF THE CDC CORE AND EXPERIMENT TUBE.

Fig. 12 Top View of the CDC Installed in SPERT IV Tank.

TABLE III

CHARACTERISTICS OF THE CDC REACTOR

Reactor Type	open pool
Reactor Vessel	
Inside diameter	20 ft
Depth	25 ft
Wall thickness	0.31 in.
Material	Type 304 stainless steel
Design pressure	
above gate	hydrostatic head plus 10-psi sur-charge
below gate	hydrostatic head plus 50-psi sur-charge
Design temperature	130°F
Primary Coolant	
Type	light water
Pressure	atmospheric
Volumetric flow	convection
Temperature	ambient ($\approx 70^\circ\text{F}$)
Fuel	
Type	compressed UO_2 powder in rod form
Rod OD	0.500 in.
Clad thickness	0.028 in.
Clad material	cold-worked, Type 304 stainless steel
Fuel enrichment, U-235	3 weight percent
Fuel density	9.28 g/cm ³
Length of fuel section	67 $\pm$ 1 in.
Control Rods	
Number	8
Type	cruciform blades
Width	4.75 in.
Thickness	0.245 in.
Poison	12 weight percent as a sintered dispersion of B_4C in aluminum
Poison length	61 in.
Rod drive	electric withdraw or insert; compressed air accelerated, gravity scram
Rods per drive	2
Follower	aluminum

TABLE III

CHARACTERISTICS OF THE CDC REACTOR (Contd.)

Transient Rods

Number	4
Type	cruciform blades
Width	2.125 in. by 3.375 in.
Thickness	0.245 in.
Poison	12 weight percent boron as a sintered dispersion of B_4C in aluminum
Poison length	28 in.
Rod drive	compressed air positioning and fast ejection
Rods per drive	4
Follower	aluminum

Core

Moderator/coolant	light water
Reflector	light water
Lattice spacing	0.663 in. square pitch
Nonmoderator-to-moderator ratio	0.807
Core diameter (overall)	28 in.
Active core height	67 ± 1 in.
Operational core loading	
number of fuel rods	1660
kg of UO_2	2.66×10^3
flux ratio (peak/avg)	3.6
Heat transfer active surface area	1221 ft ²
Reduced prompt neutron lifetime	4.18 msec
Reactivity coefficients	
control rod worth	12.2\$
excess reactivity	2.9\$
shutdown margin	9.3\$
isothermal temperature coefficient (at 20°C)	- 0.57¢/°C
uniform void coefficient of core	- 1.7×10^{-3} ¢/cc
transient rod worth	> 3\$

Experiment Tube

Tube material	Type 304 stainless steel
Length	6 ft
OD	4.500 in.
ID	3.438 in.
Wall thickness	0.531 in.

Flux

nv/watt (average over core)	$\approx 3.2 \times 10^5$
-----------------------------	---------------------------

TABLE III

CHARACTERISTICS OF THE CDC REACTOR (Contd.)

nv/watt (peak nv in test space -- with no fuel and no capsule present per watt of reactor power)	$\approx 3.5 \times 10^6$
nvt (peak in test space as above and for 3-msec transient)	$\approx 10^{15}$

Although the CDC has an active core length of six feet, control rod poisoning has reduced the useful length of the core to about 40 inches. This is shown in the vertical flux profile (cobalt) of Figure 13 taken at a fuel rod position (3-3) adjacent to the experiment tube. The depression in flux at a vertical position of 18 inches is due to the presence there of an aluminum support grid. This "intermediate" grid serves to prevent fuel rod movement which arises due to skewed temperature distributions in the fuel and cladding.

Figure 14 shows the radial flux profile (cobalt) taken at a vertical height near the intermediate grid and with no capsule present in the experiment tube. The effect of the experiment tube as a flux suppressor is clearly evident.

More important to capsule experiments is the flux profile in the test space itself. Within a typical capsule, but with no fuel present, Figure 15 shows the vertical flux (cobalt) profile along the center line of the experiment space. There is no obvious depression in this profile arising from the intermediate grid.

4.3 Irradiation Performance of the CDC

Kinetic properties of the CDC have been well predicted by a calculational model incorporating Doppler broadening feedback, and results of these predictions are presented here rather than actual experimental measurements since the latter measurements are still preliminary and incomplete. Generally, we have found that power levels and energy releases for the CDC are slightly lower by a few percent than predicted.

Typical burst shapes of the CDC are shown in Figure 16 for initial asymptotic periods of 2.0, 3.0, 7.5, and 10 msec. The upper decade of these power bursts

FIG. 13 VERTICAL FLUX PROFILE AT POSITION N3, E3 -- ADJACENT TO THE EXPERIMENT TUBE.

FIG. 14 CDC RADIAL FLUX PROFILE.

FIG. 15 AXIAL (VERTICAL) FLUX PROFILES IN THE CDC TEST SPACE
-- COMPARISON OF TWO EXPERIMENTS.

FIG. 16 COMPUTED CDC POWER PLOTS FOR VARIOUS PERIOD TRANSIENTS.

are quite symmetric around the peak; but as the power falls after peak into the region of about 100 MW, the CDC typically begins a long, relatively flat power runout. Conventional operation calls for scram shortly after the main power burst in order to eliminate the power tail.

Figures 17 and 18 show the predicted relationships of peak power and of the energy release in the CDC to the reciprocal of the asymptotic period. In Figure

FIG. 17 COMPUTED CDC PEAK POWER VERSUS RECIPROCAL PERIOD.

FIG. 18 COMPUTED CDC ENERGY RELEASE VERSUS RECIPROCAL PERIOD.

18, the "total energy release" is defined arbitrarily as the energy release up to a time of 0.5 second after peak power.

Performance of the CDC as an irradiation facility is related to the number of neutrons supplied to the test space during a power burst. The major limitations here lie almost entirely upon the heat capacity of the core material and the maximum enthalpy to which the core may be raised without damage.

From previous experience in Spert I with similar fuel, it has been shown that a hot-spot enthalpy of at least 100 cal/g in the CDC oxide material should be possible. Higher enthalpies may be possible, but scorching, burnout, and thermal expansion problems are expected to arise above 120 cal/g.

As a measure of the performance of the CDC, a figure-of-merit, F_m , has been defined which is simply the ratio of the hot-spot energy releases in the test fuel to the core fuel. Thus, if a sample fuel rod achieves a maximum energy release of 200 cal/g of UO_2 while the core hot-spot achieves only 100 cal/g of UO_2 , the figure-of-merit is 2.

Representative of the CDC performance is the figure-of-merit possible when a standard CDC fuel rod (stainless steel clad, 0.5-inch-OD, 3 percent enriched UO_2) is placed in a capsule in the experiment tube. For this configuration, the presently measured figure-of-merit is about 1.9. If the same fuel rod were 4 percent enriched, the figure-of-merit would be 2.7.

Thus, for a 3-msec-period transient which is expected to raise the CDC fuel to a peak of ≈ 110 cal/g, a 4 percent enriched CDC-type fuel rod could be raised to almost 300 cal/g which should nearly complete melting of the UO_2 .

Higher figures-of-merit are possible with smaller diameter fuels and with higher enrichments. Lower figures-of-merit arise when clusters of fuel rods are placed in the capsules.

Table IV illustrates figures-of-merit possible with various fuel samples. From the numbers shown, it is apparent that the CDC should be able to produce massive vaporization of UO_2 in the highly enriched samples.

4.4 Experimental Program

Like the PBF, the CDC also has to undergo a nuclear check-out or startup phase before it is available for general irradiation service. The CDC is presently in this initial startup phase and has completed the fiducial transient checks with initial periods to about 4.0 msec. Concurrent with this transient check of the system, investigations of the core fuel have been followed by placing core fuel samples in the central experiment test space. Placement of the core fuel in the test space permits a prior indication of core performance at an early date since the test sample in this case experiences energy releases nearly twice as large as the hot spot of the core. To date, the maximum core energy density has been approximately 60 cal/g of UO_2 , and the maximum energy density in the sample of core fuel in the test space has achieved approximately 120 cal/g of UO_2 . No systematic deleterious effects have been observed with fuel during any of these tests. Difficulty has been encountered, however, with flaws in end-closures of the core fuel that has allowed some waterlogging to occur. This waterlogging has resulted in a few fuel rod ruptures, and the program is being delayed somewhat on this account until such time as corrective measures can be taken.

As soon as the defective fuel rods in the core are replaced, it is anticipated that the CDC will be fully capable of repeated operation at initial periods of

TABLE IV

FIGURES-OF-MERIT FOR CLUSTERS OF FUEL RODS WHICH ARE 0.25 IN. OD, 0.014 IN. STAINLESS CLAD, 95 PERCENT THEORETICAL DENSITY, AND WITH ENRICHMENT AND CLUSTER SIZE AS SPECIFIED

Cluster (No. of Rods)	Enrichment (percent)	F_m
1	3	4.4
1	5	6.2
1	10	10.0
3	3	3.4
3	5	5.0
3	10	7.6

3.0 msec or somewhat less. This belief is borne out by the excellent performance observed with the fuel in the test space and is expected to be further confirmed in the near future by repeated or "endurance" testing of CDC core fuel in the test space.

The experimental program specifically directed at fuel testing will begin in the next few months. This program is primarily directed at UO_2 fuel types and will initially include a 4 percent enriched fuel of the CDC type and a 4.8 percent enriched fuel of the pellet type. Developmental testing of the PBF fuel will also be included in this initial testing phase.

In order to obtain data on fuel failure mechanisms which are more basic to understanding their performance, the variables of fuel element design and construction must also be examined in the Subassembly Testing Program. Toward this end, several different fuels of specific design, construction, and tolerances have been purchased for CDC testing. These fuels (for which the data of Table IV apply) will be available late in 1966.

Each type of fuel element will be subjected to a "scoping" series of capsule tests which determined the thresholds of failure and also to tests at shorter periods which will cause complete destruction of the element to determine the modes of failure. After the scoping series has been completed on each type of element, a more extensive test series will be initiated to provide details on the observed behavior and to assure that the objectives have been obtained. The results obtained during each portion of the test programs will strongly influence the specific areas to be investigated in followup tests. That is, information obtained in the initial tests will, to a large degree, establish areas (eg, metal-water reactions, fission product release, or others) which appear to be more urgent or fruitful to pursue with detailed experiments.

It should be noted that, as a general rule, all experiments will be placed in capsules for irradiation. These, in turn, will be inserted in the experiment tube in the CDC. The nature of the capsules to be used can vary from experiment to experiment to suit particular needs but must meet certain safeguard requirements as to strength and containment. Figure 19 illustrates a capsule design currently intended for UO_2 -fuel irradiations in the CDC. The capsule consists of standard 3-inch-OD, Type 304 stainless steel pipe with a 0.065-inch wall thickness and with a bolt-on head and end-weldments. It is capable of withstanding certain internal explosions and static pressures without rupture or leakage. Details of capsule design criteria and testing procedures are discussed fully in Section II, "Development of Capsules and Loops for Dynamic Pressure Containment", by T. F. Lagier, W. E. Driskell, and W. R. Day.

Finally, many other types of experiments can be accommodated in the CDC including the general category of unfueled radiation-effects tests on materials, components, and instrumentation; and every effort will be made to accommodate such tests consistent with the primary requirements of the Subassembly Test Program already discussed.

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FIG. 19 TYPICAL CAPSULE DESIGNED FOR FUELED IRRADIATION EXPERIMENTS IN THE CDC. THE BODY OF THE CAPSULE IS 3-INCH-OD BY 0.065-INCH WALL THICKNESS, TYPE 304 STAINLESS STEEL.

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DEVELOPMENT OF CAPSULES AND LOOPS FOR DYNAMIC PRESSURE CONTAINMENT

1. INTRODUCTION

The Subassembly Test Program which will be conducted in the Capsule Driver Core and the Power Burst Facility centers on the planned destruction of encapsulated reactor-fuel test samples. It is therefore necessary to consider the effects which transient pressures generated in such tests will have upon the containment of the experiments, the test instrumentation, and the safety of the test facility. Previous experience at Spert[1,2] has indicated that pressures resulting from transient fuel failures had magnitudes of hundreds to thousands of psi and probable rise times in the region of 20 to 30 μ sec. It is possible, however, that higher magnitudes and shorter rise-time pressure pulses may occur under more severe test conditions. It is also known that the dynamic behaviors of most materials are significantly different from those observed for slowly applied loads[3]. Therefore, since fuel failures may result in pressures that have the characteristics of explosive loads, an out-of-pile testing program was conducted to obtain the information required for the inclusion of this type of loading in the design of subassembly test capsules.

To provide a basis for relating the effects of in-pile failures to the results of an out-of-pile testing program, it was necessary to assume that the limited data which are available regarding in-pile explosions can be generalized in establishing out-of-pile test conditions. In particular, assumptions regarding the conversion efficiency of nuclear energy to mechanical energy and the maximum pressure values were needed. In this investigation, it was assumed that up to 10 percent of the nuclear energy generated during a test could be converted to mechanical energy in the form of a pressure pulse. This conversion efficiency is believed to be a conservative approximation on the basis of the estimated conversions from nuclear to mechanical energy in the SL-1 accident[4] (2.7 percent), the SNAPTRAN 2/10A-3 destructive test[5] (4 percent), and the Spert I Oxide Core test[2] (2 to 3 percent). Also, in this investigation, it was assumed that 15,000 psi was the peak magnitude of the pressure that would be encountered in the in-pile test geometry[1, 4, 6, 7].

The out-of-pile investigation was conducted in two phases. The first phase dealt with the behavior of potential capsule materials in a dynamic pressure environment. This phase of the investigation was directed toward the problems associated with the design of capsules as the containment of the experiments. The second phase dealt with the development of a filter to damp out transient pressures propagating through in-pile loop experiments. This phase was directed toward the problem of protecting the pumps and other loop hardware associated with PBF loop experiments. These two phases of the investigation will be discussed separately.

2. CAPSULE CONTAINMENT EXPERIMENTS

The objectives of the capsule containment test series were directed toward selecting (a) the material to be specified in the capsule designs and (b) the size of capsules required to contain the in-pile experiments. To meet these objectives, open-ended cylinders were subjected to line-charges of Primacord^[a] explosives in a water environment. The length of the test cylinders was specified so that the cylinders would respond as if infinitely long, except near the ends.

2.1 Description of Test Equipment

In this investigation, five different materials were tested: Type 304 stainless steel, Type 304L stainless steel, A106 Grade B carbon steel, 6061-T6 aluminum, and Zircaloy-2. The sizes and mechanical properties of the test cylinders are given in Table I. The mechanical properties of those cylinders

TABLE I
STATIC MECHANICAL PROPERTIES OF TEST CYLINDERS

Material	Cylinder Size	Heat Number	Tensile Yield Stress (psi)	Ultimate Tensile Stress (psi)	Elongation at Failure (percent)
Type 304 stainless steel	2-inch Schedule 80	X25471	43,850	83,680	67
Type 304 stainless steel	2-inch-OD by 0.065-inch wall	---	43,800	83,700	60 to 65
Type 304 stainless steel	3-inch-OD by 0.065-inch wall	---	43,800	83,700	60 to 65
A106 Grade B carbon steel	2-inch Schedule 80	631007	61,600	89,200	30
Type 304L stainless steel	2-inch Schedule 80	90211	38,200	82,800	53
Zircaloy-2	2-inch-OD by 0.225-inch wall	---	44,000	71,000	26
6061-T6 aluminum	2-inch Schedule 80	---	35,000	42,000	10 to 12

[a] Trade name for detonating cord manufactured by the Ensign Bickford Company, which contains an explosive core of pentaerythritol tetranitrate (PETN). The unit, grains per foot (gr/ft), means grains of PETN per foot of Primacord.

identified by "heat numbers" are the actual values determined by the vendors; quoted mechanical properties of other cylinders are nominal values. A number of tests were conducted using the specified cylinder sizes, except for the Zircaloy-2 specimens; only three tests of this material were conducted.

Tests were conducted in a 9-foot-diameter steel tank filled with water to a depth of 6 feet. The test cylinders were positioned vertically at the center of the tank with the tops of the cylinders approximately 1-1/2 feet below the water surface.

The test cylinders were 3 feet long, except for the Zircaloy-2 cylinders which were 18 inches long; and linear charges of Primacord, which extended over the lengths of the cylinders, were located along the vertical axis of the cylinders. The charges were held in position with Styrofoam plugs placed in both ends of the cylinders. The plugs were slotted to allow water to flood the cylinders when positioned in the tank. The explosive charges were detonated using electrical blasting caps taped to the upper end of the Primacord charges.

Measurements attempted during this investigation include (a) pressure measurements inside the cylinders using tourmaline crystal pressure transducers, (b) strain measurements of the test cylinder walls using weldable strain gages, and (c) measurements of the growths of the test cylinders using precision micrometers. Due to the extreme conditions imposed upon the test cylinders by the explosive charges, attempts to measure pressures within the cylinders and dynamic strain of the walls of the cylinders met with little success. Consequently, virtually all of the data from the testing program came from the micrometer measurements of the deformations incurred by the test cylinders.

2.2 Experiment Procedures

Two types of tests were conducted during this investigation. The first type of test consisted of subjecting the same test cylinder to repeated low-yield charges until the cylinder ruptured. Measurements of the permanent diametrical deformation incurred were made at a number of locations on the cylinder surface after each test. Since it has been determined that the deformation response of a material is dependent upon the total history of the deformation loads, rather than the increments in which they are administered^[8], this first class of tests determined the deformation response of a given material and cylinder size to any magnitude of charge below the point of failure. The second type of test consisted of bracketing the single-test burst points of the cylinders. In this type of test, only one charge was fired in each test cylinder. The burst point of the cylinder was bracketed as narrowly as possible with the supply of test cylinders that were available. Deformation measurements of burst cylinders were made by measuring the circumferential growth of the cylindrical surfaces around the point of failure.

In addition to the two types of tests described above, free-field tests were also conducted in which the pressure fields around 28-inch-long charges were measured. These tests were similar to those described above except for the absence of the test cylinders.

2.3 Discussion of Results

The free-field characteristics of the explosive used in this investigation are shown in Figures 1 and 2. Figure 1 shows the response of a tourmaline crystal pressure transducer that was positioned six inches from a 28-inch

FIG. 1 PRESSURE PULSE RECORDED BY A TOURMALINE-CRYSTAL PRESSURE TRANSDUCER POSITIONED AT A RADIAL DISTANCE OF 6 INCHES FROM A 28-INCH-CHARGE OF 350-GR/FT PRIMACORD IN WATER.

FIG. 2 FREE-FIELD PEAK PRESSURE AS A FUNCTION OF RADIAL DISTANCE FROM 28-INCH LONG CHARGES OF PRIMACORD IN WATER.

length of 350-gr/ft Primacord. The resulting pressure had a peak magnitude of 16,000 psi and a duration of about 250 μ sec. It should be noted that the rise time of the recorded pressure, 16 μ sec, is the limit of the signal conditioning and recording system. Figure 2 shows the peak pressures measured at distances of 1, 6, 12, and 18 inches from 28-inch-long explosive charges. These locations were on the perpendicular bisector of the charges. For tests that were conducted with charges exceeding 100 gr/ft, no measurements could be made at the one-inch position. The curves shown in the figure were fitted to the data by assuming a power function relationship between peak pressure and charge. The one-inch curve was extrapolated using the exponent determined for the six-inch position.

A summary of the deformation measurements made during the investigation is shown in Figure 3. In this figure, the percentage permanent diametrical deformation (Δ) is plotted as a function of linear Primacord charge measured in grains per foot. The data

FIG. 3 PERMANENT DIAMETRICAL CYLINDER DEFORMATION AS A FUNCTION OF LINEAR CHARGE DENSITY.

points shown were taken from repeated tests of individual cylinders in which a given test cylinder was subjected to repeated low-yield tests until rupture occurred. The data for the single tests (one charge per cylinder) were in good agreement with the repeated test data, but are not shown in the figure.

Burst regions that were determined for the cylinders tested in the investigation are presented in Table II. In this report, the burst region is defined as the charge range between the highest single charge that does not result in failure of the test cylinder and the lowest single charge that does result in failure. Also shown in Table II is the permanent deformation at the highest no-burst charge for the test cylinders and the mode of cylinder failures. A burst region was not determined for the Zircaloy-2 test specimens. The elongations at failure presented in Table II, when compared with corresponding static elongations in Table I, clearly indicate the need for considering the dynamic properties of materials in the design of capsules rather than the static properties which are usually used. This change in behavior is largely due to the effects of high rates of loading upon the responses of the materials.

Tests of Type 304L stainless steel cylinders were performed to investigate the effect of carbon content upon the dynamic response of Type 304 stainless steel. Data reported on the development of a new high-strength steel^[9] indicated that a decrease in the carbon content of the steel increased the Charpy impact strength of the steel. A detailed comparison of the two test cases is presented in Table III. As can be seen from the table, the Type 304L cylinders were able to contain about 78 percent more explosive energy than Type 304 cylinders of the same size. This result would not have been expected from the static strength properties shown in the table since both the static yield point and ultimate strength are slightly greater for the Type 304 than for the Type 304L materials. As a result of these tests, the capsule development effort was directed toward the use of extra-low-carbon stainless steel in all capsule designs. It should be pointed out, however, that the results presented in Table III are for one type of explosive and one cylinder size. More experimental data will be required before any general conclusions can be drawn regarding the effect of carbon content upon dynamic behavior of steel.

The deformation response curves of the test cylinders shown in Figure 3 were correlated using a dimensional analysis approach. The complete expression developed for the test system included 13 parameters (10 independent dimensionless terms); from physical reasoning, some of these dimensionless terms required for similarity obviously had little influence on the test results. Other dimensionless terms could not be evaluated using the test data that were available. From considerations of the pertinent terms which could be evaluated, the following equation was developed:

$$\Delta = k \left[\frac{w}{(\sigma_y)(r_i)(t)} \right]^a$$

where

Δ = permanent diametrical deformation, percent

w = charge of explosive per unit length, gr/ft

σ_y = static yield stress of cylinder material, psi

TABLE II

BURST CHARACTERISTICS OF TEST CYLINDERS

<u>Test Cylinder</u>	<u>2-Inch Schedule 80 6061-T6 Aluminum</u>	<u>2-Inch Schedule 80 Type 304 Stainless Steel</u>	<u>2-Inch Schedule 80 Type 304L Stainless Steel</u>	<u>2-Inch-OD by 0.065- Inch Wall Type 304 Stainless Steel</u>	<u>3-Inch-OD by 0.065- Inch Wall Type 304 Stainless Steel</u>	<u>2-Inch Schedule 80 A106 Grade B Steel</u>
Burst region (gr/ft)	120 to 150	400 to 450	700 to 800	320 to 400	500 to 525	< 100
Elongation as a result of explosive failure(per- cent)	5.7	12.5	23.9	26.7	27.4	< 3.6
Mode of failure	Shear	Shear	Shear	Shear	Shear	Tensile

TABLE III

COMPARISON OF TYPES 304 AND 304L
STAINLESS STEEL CYLINDER TESTS

<u>Material</u>	<u>Cylinder Size</u>	<u>Material Specifi- cation</u>	<u>Static Yield Point (psi)</u>	<u>Static Ultimate Strength (psi)</u>	<u>Static Elongation (percent)</u>	<u>Carbon Content (percent)</u>	<u>Burst Region (gr/ft)</u>	<u>Elongation as a Result of Explosive Failure (percent)</u>
Type 304 stainless steel	2-inch Sched- ule 80	ASTM-A312	43,850	83,680	67	0.05	400 to 450	12.5
Type 304L stainless steel	2-inch Sched- ule 80	ASTM-A312	38,200	82,800	53	0.02	700 to 800	23.9

r_i = inside radius of the test cylinder, in.

t = thickness of cylinder wall, in.

k, a = constants.

Figure 4 is a plot of the deformation test data using the above equation.

FIG. 4 CORRELATION OF PERMANENT DEFORMATION DATA OBTAINED FROM CYLINDER TESTS.

The constants for the equation were determined for the fitted curve shown in the figure. The equation of this curve is

$$\Delta = 169 \left[\frac{W}{(\sigma_y)(r_i)(t)} \right]^{0.82}$$

This equation provides a reasonable estimation of the permanent deformation incurred by a range of cylinder materials and sizes when subjected to internal Primacord charges and has found application in estimating the deformation response of other sizes of cylinders to similar explosive charges.

The correlation of the data shown in Figure 4 indicates that although the effect of strain rate upon the behavior of materials is significant and complex, the responses of a variety of materials is orderly and can be predicted, reasonably well, using classical techniques. This observed response of materials supports the feasibility of investigating a variety of reactor materials in out-of-pile tests once the character of the pressure force which is to be mocked up has been defined.

It was observed during the cylinder test series that most of the ruptured cylinders exhibited deformation patterns in which one or two points of peak deformation were established over the lengths of the ruptured regions. Since this type of preferential deformation behavior was not anticipated, an analysis of the observed patterns has been initiated.

Figure 5 shows one type of rupture pattern that was established in a stainless steel tube which was subjected to one test above the rupture point. In the figure, the charge was detonated from the top end of the cylinder. The detonation velocity of Primacord is about 21,000 ft/sec, and the acoustic propagation velocity in water is about 4900 ft/sec. Since the detonation velocity is about four times as fast as the progression of the pressure disturbance through water, the point at which the acoustic signals, generated at the ends of the charge pass each other in the cylinder will be displaced from the midpoint of the cylinder. The amount of this displacement is equal to half the product of detonation time and the velocity of sound in water. This point at which the signals pass each other is shown graphically in Figure 6. The point at which pulses first meet from opposite ends is located at a distance, X_m , equal to $5/8$ of the length from the end at which detonation was initiated. The duration of the impulse at any point in the cylinder is the time between the time of arrival of the first signal (the detonation wave front) and the time of arrival of the last signal (the acoustic wave from the farthest element of charge). This impulse duration time is a minimum at X_m . Since the total impulse received by each element of the cylinder is a constant, except near the ends of the cylinder, the point of minimum impulse duration then corresponds to a point of highest average pressure. This point is shown in the photograph in Figure 5 where the peak deformation point is located $5/8$ of the length of the pipe from the top of the cylinder. Although this type of deformation behavior was not typical of that observed in most of the ruptured cylinders, the phenomenon that has been discussed is postulated to have had an influence of the more typical type of deformation pattern that was observed.

An example of the most typical deformation pattern observed is shown in Figure 7. In this figure, the charge was also detonated from the top of the cylinder. The two points of peak deformation are somewhat symmetrical about the point of minimum impulse duration, X_m . An analysis is currently underway to determine if the observed deformation patterns can be explained in terms of standing wave phenomena which may be occurring in the cylinders. At the present time, it appears that the combined effects of the postulated wave phenomena and the minimum impulse duration point may provide a model which explains the deformation patterns observed.

Figure 5 Shear Failure of a 2-inch-OD by 0.065-inch-thick Type 304 Stainless Steel Cylinder Subjected to a Linear Charge of 400 GR/FT.

Figure 6 Duration of Total Impulse as a Function of Position on the Axis of a Line Charge Detonated from One End.

Figure 7 Shear Failure of a 2-inch, Schedule 80, Type 304 Stainless Steel Cylinder Subjected to a Linear Charge of 800 GR/FT.

3. LOOP FILTER EXPERIMENTS

Part of the Subassembly Test Program proposed for the Power Burst Facility consists of transient testing of fuel samples under conditions of pressure and flow. Since this type of testing will be conducted in an in-pile loop, it was necessary to make provisions in the system for restricting the propagation of transient pressure pulses through the loop piping, pumps, and associated hardware. However, pumping power considerations required that the flow resistance to the system be as small as possible. Since many of the problems associated with the design of a filter to perform the above functions could not be handled analytically, the filter design was determined by the results of a series of experiments.

The basic criterion that was established for satisfying the design requirements was that the filter must reduce all transient pressures generated by the experiment to a maximum value of 500 psi at the filter outlet.

3.1 Description of Test Equipment

The test arrangement is shown in Figure 8 along with the location of the test instrumentation. Since it was desirable to obtain the overall attenuation from the source to the filter outlet, the in-pile tube was included as an integral part of the test hardware. The in-pile tube and tube internals were constructed to half-scale. Pressure transducers were located as shown in the figure to determine the pressure attenuation through the system. Transducer PXD-4 was moved to several different positions during the tests. In addition, strain measurements were made of the deformation of the in-pile tube in the region of the explosive source.

FIG. 8 PRESSURE-ATTENUATION-TEST HARDWARE AND LOCATION OF THE TEST INSTRUMENTATION.

Several types of filter designs were evaluated in the investigation. Details of these designs will be discussed in the results.

3.2 Experiment Procedures

Three types of explosives were used in the experiments to investigate the response of the filter to pressure pulses having a range of rise times and durations. These explosives are identified in Table IV.

TABLE IV
CHARACTERISTICS OF EXPLOSIVES

	<u>Detonation Velocity</u>	<u>Heat of Reaction</u>
Pyrocore ^[a] (4 gr/ft)	21,000 ft/sec	475 cal/ft
Black Powder (Type fffg)	≈ 1,300 ft/sec	≈ 700 cal/gram
Firecracker	-----	≈ 2000 cal

[a] DuPont trade name for a detonating cord which contains a mixture of pentaerythritol tetranitrate (PETN), aluminum, and beryllium oxide as an explosive core.

For the experiments, the in-pile tube, filter, and piping were water filled. All air in the system was bled off before the experiments were conducted. Charges were placed in the lower portion of the in-pile tube and fired using electrical detonators.

3.3 Discussion of Results

Over 130 tests were conducted before an acceptable filter design was established. Attempts were made to analyze the test data using linear techniques; however, the system exhibited nonlinear characteristics, apparently due to cavitation effects established in the system by rarefaction waves. The system was subjected to pressure pulses having a range of rise times and durations that bracketed the region of interest.

In evaluating the influence of the various filter components upon the test geometry, use was made of an electrical analog network in which the system components were described in terms of equivalent electrical circuits. Figure 9 shows the network representation of a section of pipe. In acoustics, inertance is the element that opposes a change in volume velocity, and acoustical capacitance opposes a change in the applied pressure. Therefore, voltage and pressure, current and volume velocity, inductance and inertance, and capacitance and acoustical capacitance are analogous in electrical and acoustical systems.

Many observations could be made regarding the results of the various tests; however, this discussion will be confined to three test configurations which had a critical influence in the final design selection.

3.31 Test Series No. 1. The first filter to be tested consisted of two expansion chambers series-coupled, as shown in schematic and analog form in

FIG. 9 ELECTRICAL TRANSMISSION LINE ANALOG OF A SECTION OF PIPE.

Figure 10. These expansion chambers are equivalent to an increase in the pipe diameter; in this case, the ratio of the chamber cross-sectional area to that of the loop pipe was about 13. Filters of this type have definite resonate frequencies where the attenuation is decreased. However, resonance occurs at different frequencies for the two filters tested, and overlapping of attenuation results[10].

Table V summarizes the response of series-coupled filter components. All pressure readings are maximum recorded values. The locations of the pressure transducers listed in Table V are shown in Figure 8. For this series of tests, the duration of the pressure pulses into the filter ranged from 0.6 to 0.8 msec, while the duration of the initial pressure pulses out of the filter was approximately 10 msec. An example of the filter inlet and outlet pressures is

FIG. 10 ELECTRICAL ANALOG OF TWO EXPANSION CHAMBERS IN SERIES.

TABLE V^[a]

RESULTS OF TEST SERIES NO. 1

Test Number	Pressure Transducer (psi)				Type of Explosive	Remarks
	PXD-1	PXD-2	PXD-3	PXD-4		
14	1400	165	4,600	2700 (3)	1-ft Pyrocore (2)	-----
15	850	225	39,000	1700 (3)	3-ft Pyrocore (1)	1 percent yield occurred in-pile tube at location of explosive
16	1850	160	3,200	260 (4)	1-ft Pyrocore (2).	PXD-4 failed -- replaced Test 18
18	690	190	4,250	150 (5)	1-ft Pyrocore (2)	-----
19	1050	345	4,250	150 (6)	1-ft Pyrocore (2)	-----
20	1100	230	4,150	165 (7)	1-ft Pyrocore (2)	-----
21	770	320	4,000	240 (8)	1-ft Pyrocore (2)	-----
22	650	175	4,250	145 (8)	1-ft Pyrocore (2)	-----

[a] The number in parenthesis following the pressure data from PXD-4 denotes the location of this pressure transducer, which is also indicated in Figure 8. The location of the pressure source within the in-pile tube was alternated between two positions: No. 1 -- 18 inches from the bottom of the in-pile tube; No. 2 -- at the top of the in-pile tube, opposite the filter inlet.

given in Figure 11. Although this filter arrangement reduced the filter outlet pressures considerably, the observed attenuation was not sufficient for the design to meet the specified criterion in the low-frequency region.

3.32 Test Series No. 2. In this series of tests, the two expansion chambers (Figure 10) were tested separately. Tests were conducted both with the chambers completely full of water and also with an air expansion space contained in the chambers. The results of the tests with the chambers water-filled were not significantly different from the results of the tests with the chambers series-coupled (test series No. 1). However, when the chambers were tested with about half of the expansion volumes filled with air, a large decrease in the

FIG. 11 PRESSURE HISTORIES RECORDED AT THE INLET AND OUTLET OF TWO SERIES-COUPLED EXPANSION CHAMBERS. TOTAL CHARGE: 1-FOOT LENGTH OF 4-GR/FT PYROCORE.

transmitted pressure was observed. The effect of the air space was to reduce the inertance and increase the capacitance of the expansion chamber, thereby increasing the attenuation of low-frequency pressures passing through the chamber. The effect of the increased capacitance of the air volume is illustrated in Table VI by the inlet and outlet pressures that were recorded during the tests. As a result of this series of tests, a filter design was proposed in which the

TABLE VI

RESULTS OF TEST SERIES NO. 2

Test Number	Pressure Transducer (psi)				Type of Explosive	Remarks
	PXD-1	PXD-2	PXD-3	PXD-4		
30	885	260	---	300 (9)	1-ft Pyrocore (2)	8-inch chamber, water filled
31	940	255	---	320 (9)	1-ft Pyrocore (2)	8-inch chamber, water filled
32	880	25	---	---	1-ft Pyrocore (2)	8-inch chamber, with air space
33	1050	20	---	---	1-ft Pyrocore (2)	8-inch chamber, with air space

filter would consist of a water-filled expansion chamber, to attenuate high-frequency pressure pulses, and an air plenum, which would attenuate low-frequency pressure pulses without interfering with the coolant flow through the loop.

3.33 Test Series No. 3. The third series of tests consisted of a water-filled expansion chamber followed by an air dome, as shown in Figure 12. The dome had a volume of about 225 cubic inches. As the charge levels were increased, the number of air domes required to attenuate the low-frequency pressures was also increased.

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FIG. 12 THE ELECTRICAL ANALOG OF AN EXPANSION CHAMBER FOLLOWED BY AN AIR DOME.

The results of the test series are shown in Table VII. Black powder charges were used to evaluate the response of the filter to large-magnitude pressure pulses. For the largest charges to which the system was subjected, it was found that four air domes were required to provide the desired amount of attenuation through the filter. The quantities of explosives used were increased up to a maximum of 400 grams of black powder. For this size charge, the pressure at the inlet of the filter was 11,000 psi, and the pressure at the outlet was 560 psi. Since the data indicated that the filter was providing the amount of attenuation required to protect the loop pumps from damage, the test series was terminated and a final design adopted consisting of a water-filled expansion chamber, to attenuate high-frequency pressures, followed by four air domes, to attenuate low-frequency pressures.

The pressure-time characteristics of the adopted design are illustrated in Figure 13, which shows the measurements recorded in front of the expansion chamber, between the expansion chamber and the air domes and beyond the air domes for a 190-gram black powder charge.

RESULTS OF TEST SERIES NO. 3

Test Number	Pressure Transducer (psi)				Type of Explosive	Remarks
	PXD-1	PXD-2	PXD-3	PXD-4		
118	7,700	1250	10,000	5100 (7)	120 grams black powder (1)	One air dome
125	8,580	1500	9,600	4850 (7)	120 grams black powder (1)	One air dome (twice the air volume of that of test No. 118)
127	6,860	780	10,000	5950 (7)	150 grams black powder (1)	Three air domes
128	6,550	235	10,000	3620 (7)	190 grams black powder (1)	Four air domes
129	7,020	310	12,000	4220 (7)	240 grams black powder (1)	Four air domes
136	11,000	560	12,700	7500 (7)	400 grams black powder (1)	Four air domes

FIG. 13 PRESSURE ATTENUATION CHARACTERISTICS OF ADOPTED FILTER DESIGN, TOTAL CHARGE: 190 GRAMS OF BLACK POWDER.

The in-pile loop filter arrangement is shown in Figure 14. The loop piping that is located between the filters and the pumps is 4-inch, Schedule 120, Type 321 stainless steel. The expansion chambers are eight feet long and have an inner diameter equal to four times the inner diameter of the loop piping. The expansion chambers and connecting piping between filters are designed to contain an internal static pressure of 3750 psi at 650°F. The air domes associated with each of the expansion chambers have a total air volume of 1.0 ft³.

4. SUMMARY AND CONCLUSIONS

It has been found that for one given set of test conditions, extra-low-carbon stainless steel test cylinders possessed considerably greater strength in an explosive environment than similar steel cylinders having a higher carbon content. The results of a series of tests conducted using selected materials dictated the use of Type 304L stainless steel in capsules that are being designed for use in the investigation of reactor fuel samples which are subjected to transient power bursts.

An equation has been developed which correlates the deformation response of a variety of materials to a range of Primacord explosive charges. The agreement of the data obtained using this correlation indicated that the complex phenomena which occur in capsule materials subjected to large-magnitude transient pressure fields do not preclude the description of the macroscopic responses of capsules using standard classical techniques.

Analysis of cylinder rupture patterns indicates the possibility of wave phenomena being established which contribute to the rupture of the cylinders. A model is being developed which describes the observed rupture patterns in terms of longitudinal standing waves in the cylinders.

Tests conducted in conjunction with the development of a transient pressure filter for an in-pile loop facility resulted in a design consisting of an expansion chamber, to attenuate high-frequency pressure pulses, and a series of air-expansion volumes, to attenuate low-frequency pressure pulses.

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TESTING AND APPLICATION OF DETECTORS FOR MEASUREMENT OF DYNAMIC PHENOMENA DURING BURST IRRADIATION CAPSULE EXPERIMENTS

1. INTRODUCTION

One of the primary purposes of capsule testing performed with the Capsule Driver Core (CDC) is to determine the consequences of test fuel failure resulting from the test fuel being subjected to high-energy, short-period nuclear transient irradiations. This information is needed to properly assess the hazards associated with reactor operation and siting decisions. Measurements of mechanical [a] and nuclear energy release within the capsule are made under adverse environmental conditions of large-magnitude transient radiation fields and in the presence of explosive pressure fields. Measurements of cladding temperature and strain and fuel temperature are required to determine the nature of fuel rod failure.

The anticipated radiation environment may be as high as 8×10^{16} nv (thermal) peak with about 10^{15} nvt integrated dose and 10^{12} R/hr peak with a total integrated dose of about 10^6 rads. The pressure environment may exceed 20,000 psi under shock pressure conditions with rise times on the order of 10 μ sec or less. The maximum temperature environment in the water is limited to less than 100°C. However, peak fuel temperatures may be in excess of 3000°C, and claddings are expected to melt. The asymptotic period of the nuclear transient may be as short as 2 msec with a burst half-width of 10 msec and a total test duration of 500 msec.

The aforementioned environs pose many unusual instrumentation problems, and the transducer devices are selected to withstand these environs and respond to the signal rates which are generated. The electronic systems are selected to be compatible with the response requirements of the detector whenever possible. The following discussions are primarily concerned with the detector measurement problems, and no further discussion of the electronic systems is made except that a system description is included in the appendix for general information. An instrumented capsule is illustrated in Figures 1, 2, and 3 with a cutaway view of the same capsule illustrated in Figure 4. These figures

FIG. 1 INSTRUMENTED CAPSULE AND FUEL ROD PREPARED FOR USE

[a] Mechanical energy is defined here as the combination of kinetic and potential energies arising from pressure pulses.

FIG. 2 CLOSEUP OF THE TOP END OF THE CAPSULE.

FIG. 3 CLOSEUP OF THE BOTTOM OF THE CAPSULE

illustrate the instrumentation employed for a typical CDC capsule experiment. Capsule instrumentation typically consists of a pressure transducer, strain gages, a cobalt wire, thermocouples, and a growth ring on the fuel rod. A Conax fitting is provided in the head of the capsule for exit of the thermocouple and growth ring leads through a pressure seal. A two-way valve is provided to allow bleed-off of any pressure buildup as a result of a test and to allow static testing of the strain gages and pressure transducers. For normal test conditions, the capsule and instrumentation are assembled as shown, and the capsule is filled with water to about 55 inches leaving about 12 inches of air space in the top of the capsule.

2. MECHANICAL ENERGY RELEASE MEASUREMENTS

One of the quantities of major interest is the percentage of the nuclear energy which is converted to mechanical energy in the capsule environment as the result of test-fuel failure. Mechanical energy may be calculated from simultaneous measurements of capsule pressure, strain, and water velocity. The following discussion deals with the out-of-pile testing performed to evaluate transducers and measurement techniques and some of the results obtained from the in-pile tests performed in the CDC.

2.1 Pressure Measurements

A series of out-of-pile tests was performed to aid the development of techniques for measuring explosive pressures in the radiation environment. Over the past three years, considerable effort has gone into radiation testing of transducer devices for use in radiation environments. The results of these tests are described in IDO-16914, Effects of Transient Nuclear Radiation on Transducers and Electrical Cables, and in IDO-17103, Transient Nuclear Radiation Effects on Transducer Devices and Electrical Cables. These radiation tests limited the choice of pressure transducers to strain gage types. The out-of-pile test program was then undertaken to find a pressure transducer

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FIG. 4 CUTAWAY VIEW OF A CDC CAPSULE.

which could respond accurately to explosive pressure. These results further limited the choice of pressure transducers to bonded, strain-gage-type transducers.

2.11 Out-of-Pile Testing. Out-of-pile tests were performed in a water environment in a CDC prototype capsule (similar to the capsule shown in Figure 4) using the detonation of Primacord, Pyrocore, or black powder as a source of pressure. The approximate risetime and pressure pulse duration times of the explosives are listed in Table I.

TABLE I
CHARACTERISTICS OF EXPLOSIVES
USED FOR OUT-OF-PILE TESTS

Explosive	Rise Time	Duration Time (free-field)
Primacord ^[a]	3 to 10 μ sec	200 to 300 μ sec
Pyrocore (4 gr/ft) ^[b]	3 to 10 μ sec	200 to 300 μ sec
Pyrocore (20 gr/ft) ^[b]	$\approx$ 10 μ sec	$\approx$ 300 μ sec
Black Powder	3 to 100 μ sec	300 to 1000 μ sec

[a] Trade name for detonating cord manufactured by the Ensign Bickford Company, which contains an explosive core of pentaerythritol tetranitrate (PETN). When used in this report, the unit gr/ft means grains of PETN/ft of Primacord.

[b] DuPont trade name for detonating cord which contains a mixture of pentaerythritol tetranitrate (PETN), aluminum, and beryllium oxide as an explosive core.

Tourmaline crystal pressure transducers manufactured by Crystal Research, Inc., were selected as the standard for the out-of-pile tests. These transducers are designed for measurement of explosive pressures in a water environment. Radiation tests (IDO-17103) previously indicated, however, that the radiation-induced signal from these transducers would be of the same order of magnitude as the signals expected from the pressure pulse itself; therefore, their use would be unacceptable in the anticipated CDC radiation environment. Figure 5 shows the response of a typical tourmaline crystal transducer which was positioned 6 inches from a 28-inch length of 40 gr/ft Primacord. The resulting pressure has a peak magnitude of 4000 psi and a duration of 250 μ sec. The rise time of the recorded pressure pulse was limited by the 16- μ sec rise time of the signal conditioning and recording system. The Crystal Research pressure transducer has a rise time of 1 μ sec as specified by the manufacturer, and the electronics used with the transducer limits the low-frequency response to 5 Hz. Figure 6 shows a Crystal Research transducer that was used for these out-of-pile evaluation tests.

FIG. 5 RESPONSE OF A CRYSTAL RESEARCH PRESSURE TRANSDUCER TO A PRIMACORD DETONATION.

FIG. 6 CRYSTAL RESEARCH PRESSURE TRANSDUCER PREPARED FOR MOUNTING IN A CAPSULE.

A bonded strain gage, catenary diaphragm pressure transducer, similar to the one shown in Figure 7, was selected as a result of the out-of-pile test for use in the in-pile capsules. The previous radiation sensitivity tests indicated that this particular transducer model would have radiation-induced output of only 1 to 5 percent of the transducer's rated output for the expected radiation environment in the CDC. These transducers are manufactured by the Advanced Technology Laboratories Division of American Standard.

Comparisons between data from the two types of pressure transducers are illustrated in Figures 8 and 9. They are, respectively, (a) the output from a bonded strain gage transducer and (b) the output from a tourmaline crystal pressure transducer. The pressure results from the detonation of a one-foot length of 20-gr/ft Pyrocore in a CDC

FIG. 7 CUTAWAY VIEW OF A BONDED STRAIN GAGE PRESSURE TRANSDUCER.

prototype capsule. The capsule was water-full, with an open top, and the transducers were threaded into the capsule bottom. As can be observed from the figures, the output of the crystal transducer is well behaved; whereas, the strain gage transducer shows some oscillations in its output. The peak pressures agree within 10 percent, and the integral of the pressure pulse also agrees within 10 percent. The output of the strain gage pressure transducers was better behaved when exposed to pressure pulses resulting from the detonation of charges of black powder. The rise-time capabilities of the strain gage pressure transducers has been bracketed between 10 and 30 μ sec and, for best results, should not be used to measure explosive pressures which have rise times less than 10 μ sec.

Strain gages have been fastened to the capsule wall in an attempt to measure the wall strain and to compare the strain with the internal pressure. Two types of strain gages have been used thus far: (a) a wire gage in a weldable sheath manufactured by Microdot, Inc., and (b) a foil gage manufactured by Dentronics, Inc. Both strain gages have an upper strain limit of about 6000 μ in./in.

Initial testing was performed using small charges of black powder as a source of pressure. The resulting pressure pulse had a peak pressure of about 3000 psi. The capsule was made from a piece of 2-inch, Schedule 40 aluminum pipe. The results from these tests were encouraging since the gages responded to the strain and could be calibrated to give an accurate indication of the time history of the pressure pulse.

FIG. 8 RESPONSE OF A BONDED STRAIN GAGE PRESSURE TRANSDUCER TO A PYROCORE DETONATION INSIDE A CAPSULE.

Further testing was done using Primacord as a pressure source and using a capsule made from a 2-inch, Schedule 80, Type 304 stainless steel pipe. For these tests, both types of strain gages failed to give any usable data when the internal pressure exceeded about 2000 psi due to failure of either the strain gage or the bonding. However, strain gages mounted on a 4-inch, Schedule 160, Type 304 stainless steel pipe did respond to strains resulting from the detonation of a Primacord charge placed inside a CDC prototype capsule located inside the 4-inch pipe. The strain gage output indicated a permanent deformation of 13,000 $\mu\text{in./in.}$, as compared with about 18,000 $\mu\text{in./in.}$ permanent set in the pipe as measured with a micrometer. Since the gages are nominally limited to strains of 6000 $\mu\text{in./in.}$, the error was considered to be well within reason.

2.12 In-Pile Tests. Figure 4 is a cutaway view of a CDC lead-rod capsule showing the pressure transducer, the Conax fitting for exiting leads from the capsule interior, and the fuel rod and associated hardware. Four pre-formed circumferential strain gages, similar to the one illustrated in Figure 10, were mounted on the exterior of the capsule wall.

Figure 11 shows the response of the capsule pressure transducer during a power burst with an asymptotic period of 7.9 msec. The pressure pulse is due to the rupture of a fuel rod in which the fuel had previously been intentionally "waterlogged". The recorded pressure pulse had a peak magnitude of 5400 psi and a rise time of 40 μsec . Figure 12 shows the response of a similar pressure transducer of higher sensitivity during a test wherein no fuel rod rupture occurred. In this test, the peak pressure was 90 psi with a rise time of about 75 μsec . These figures show that the bonded strain gage transducers are

FIG. 9 RESPONSE OF A CRYSTAL RESEARCH PRESSURE TRANSDUCER TO A PYROCORE DETONATION INSIDE A CAPSULE.

FIG. 10 PRE-FORMED WELDABLE STRAIN GAGE.

capable of measuring not only explosive pressures but also pressures of small magnitudes (10 to 20 psi) during a nuclear excursion. For these tests, the pressure transducers had a radiation-induced output of less than one percent of the full-scale range of the transducer as a result of a flux field of about 10^{16} nv and 10^{11} R/hr.

The output from the capsule strain gages on the 7.9-msec "waterlogged" rupture test is shown in Figure 13. Compared with the output of the pressure transducer, the indicated pressure pulse from the strain gages is about a factor of 2 smaller in magnitude. A possible explanation for this discrepancy lies in the fact that acoustic theory predicts pressure doubling at the bottom boundary due to the reflected pressure pulse. The integral of the two curves is essentially the same, indicating that the strain gages may see the reflected pulse displaced in time with respect to the incident pulse.

2.2 Water Velocity Measurement

As has been stated previously, determination of the amount of mechanical energy generated as a consequence of fuel rod failure is one of the objectives

FIG. 11 RESPONSE OF A BONDED STRAIN GAGE PRESSURE TRANSDUCER DURING A FUEL ROD RUPTURE TEST.

FIG. 12 RESPONSE OF A BONDED STRAIN GAGE PRESSURE TRANSDUCER DURING A TEST WITHOUT A FUEL ROD RUPTURE.

FIG. 13 RESPONSE OF A WELDABLE STRAIN GAGE DURING A FUEL ROD RUPTURE TEST.

of the CDC capsule test program. Part of this mechanical energy release may cause a column of water to be accelerated in the capsule which contains the test fuel rod. Measurement of the velocity of this water slug in conjunction with measurement of pressure at the bottom of the capsule would permit determination of the magnitude of the mechanical energy contained in the water slug. Therefore, a device has been designed to measure water-slug velocity, and out-of-pile testing has been completed using chemical explosives to accelerate the water.

The velocity transducer consists of magnets embedded in a moveable, buoyant, polyethylene piston which slides along a stationary armature. The armature is wound around a 1/2-inch-OD tube, and a hole is drilled in the center of the polyethylene piston to allow it to slide freely up and down the tube. The direction of the armature winding is reversed after each 1/2-inch length of the tube so that the output signal from the armature induced by the motion of the magnets is also reversed, approximating a sine wave for each inch of travel of the polyethylene piston and embedded magnets. Figure 14 is a photograph of the velocity transducer attached to the top of a prototype CDC lead-rod capsule.

Preliminary testing of the velocity transducer was performed using a 2-inch-long piston. Small chemical explosives ("cherry-bombs") were detonated in the lower portion of the capsule. For these tests, the transducer was mounted on the inside of the capsule. The velocity of the piston was measured over 16

inches of travel, and the pressure at the bottom of the capsule was measured using a bonded strain gage pressure transducer. Results of these tests are listed in Table II together with a comparison of the momentum of the water slug (the average velocity over 16 inches of travel times the mass of water above the explosive) with the momentum as determined from the pressure transducer at the capsule bottom (the integral of the pressure pulse times the inside cross-sectional area of the capsule). These momenta agree within 20 percent for each test. The kinetic energy of the water slug listed in Table II is one-half the product of the average velocity of the water slug and the momentum indicated by the pressure pulse.

TABLE II

RESULTS FROM INITIAL WATER VELOCITY TRANSDUCER TESTS

Test Number	Maximum Velocity (ft/sec)	Average Velocity (ft/sec)	Momentum from Velocity Transducer (slug ft/sec)	Momentum from Pressure Transducer (slug ft/sec)	Kinetic Energy in Water Slug (ft/lb)
1	133	128	8.9	10.5	670
2	136	126	10.7	10.6	670
3	104	99	8.4	10.1	500
4	95	89	7.5	8.2	365
5	142	132	11.2	12.8	845
6	144	136	11.5	12.6	855
7	169	113	9.6	8.9	505
8	156	138	11.7	12.3	850
9	157	108	10.8	11.5	620
10	125	106	10.5	11.4	610

Additional tests were performed with the capsule top open, as shown in Figure 14, and high-speed motion pictures of the cylinder and water slug were taken. Detonations of one-foot lengths of 4-gr/ft Pyrocore were used in addition to the "cherry bombs", and the capsule pressure was measured using a tourmaline crystal pressure transducer. The motion pictures indicated that the velocity transducer gave good data for the "cherry bomb" explosives, which had rise times of 30 to 100 μ sec; however, for the Pyrocore explosives which had 3- to 10- μ sec rise times, the piston separated from the main water column, ie, "spalled", resulting in errors of 50 percent or greater.

Further tests were performed with Pyrocore explosives where the piston lengths were varied from 2 to 16 inches. Motion pictures taken during these tests indicated that the piston ceased to spall for lengths in excess of 8 inches. Therefore, for pressure pulses having rise times of 3 to 100 μ sec and magnitudes up to 10,000 psi, it has been concluded that the velocity transducer should provide a good measure of the water-slug velocity, with an overall accuracy of 20 percent.

3. NUCLEAR ENERGY RELEASE MEASUREMENTS

The measurement of energy per gram of fuel released in a test fuel element or elements is accomplished by irradiation of a cobalt wire attached to the outside of the capsule wall. The energy is calculated by multiplying the total nvt, as measured by the cobalt wire, by the energy per gram per nvt of test fuel, as previously determined by a fission product analysis of a similar test fuel element. The time history of the energy release in the test fuel is determined currently by multiplying the time history of the core energy (which is measured by integration of data from ion chambers located exterior to the core) by the product of the peak-to-average thermal neutron flux in the core and the figure-of-merit. Figure 15 illustrates the time history of the core energy during a 6.9-msec-period test. It should be noted that 70 percent of the energy is released within 80 msec; an additional 20 percent is released to time of scram, approximately 300 msec later. The remaining 10 percent arises after scram.

FIG. 15 TIME HISTORY OF THE CORE ENERGY AS DETERMINED BY AN ION CHAMBER.

In an attempt to provide a more accurate measurement of the total energy release in the test fuel, a calorimetric capsule is being designed. As illustrated in Figure 16, the calorimetric capsule is essentially a Thermos bottle constructed of a double-walled stainless steel vessel. To minimize heat losses, the adjacent surfaces will be polished; an aluminum foil will be placed between the walls; the top plug will include a teflon insulation; and the space between the walls will be evacuated to 10^{-3} mm Hg or better. A vacuum gauge and mechanical bellows valve will be located at the capsule bottom to monitor the vacuum prior to testing and provide access for re-evacuation as required between tests. A vertical profile of water temperature will be measured by

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FIG. 16 CUTAWAY VIEW OF A CALORIMETRIC CAPSULE.

thermocouples before and after each test. The accuracy of the calorimetric capsule energy measurement should be 3 to 5 percent, as compared with an expected accuracy of 8 to 9 percent for the cobalt wire technique.

4. FUEL ROD PERFORMANCE MEASUREMENTS

To determine the nature of fuel rod failure resulting from a short-period nuclear excursion, a number of fuel rod performance measurements are required. The required measurements are: cladding surface temperature, fuel temperature, and cladding strain.

4.1 Temperature Measurement

The requirements for surface temperature measurements are to determine the peak temperature and the time history of the surface temperature. Measurements of the fuel temperature are also of considerable importance since the cladding temperature gives no direct indication of the fuel temperature. The development of oxide fuel temperature measurement during a nuclear excursion is still in the experimental stages.

4.11 Surface Temperature Measurement. The surface installations are made from standard manufactured lines of 30-gauge (0.010 inch diameter), Type K, Chromel-Alumel, high-temperature, fiberglass-insulated thermocouple wire. Figure 17 is a photograph of an instrumented oxide fuel rod. The leads are held in place along the rod with 0.005-inch stainless steel tabs laid over the wire and resistance-welded to the rod. The ends of the 0.010-inch wires are flattened to a thickness of approximately 0.003 inch and then resistance-welded to the rod. It should be noted that the thermocouple is actually composed of two thermoelectric junctions: Chromel to stainless steel and Alumel to stainless steel, with the junction, in each case, at the interface of the stainless steel and the Chromel or Alumel wire.

FIG. 17 CLOSEUP OF AN OXIDE FUEL ROD THERMOCOUPLE INSTALLATION.

Work performed at Spert and other projects indicates that the response time (0 to 63 percent time constant) of a 0.003-inch thermocouple on stainless steel lies between 3 and 7 msec. For a single time-constant system, the dynamic error can be expressed as

$$\epsilon = \dot{e}_o(t) \tau$$

where

ϵ = error ($^{\circ}\text{C}$)

$\dot{e}_o(t)$ = derivative, the output ($^{\circ}\text{C}/\text{second}$)

τ = 0 to 63 percent time constant (seconds).

For a maximum error of 10°C and a τ of 5 msec, the maximum allowable rate of change of the temperature is about $2000^{\circ}\text{C}/\text{sec}$.

Figure 18 shows the response of a 0.003-inch thermocouple during a 7.9-msec-period transient. (The capsule pressure time history is depicted in Figure 12.) The thermocouple output indicates a maximum temperature rise of 140°C , with a maximum slope of $2000^{\circ}\text{C}/\text{sec}$ which approaches the allowable slope previously described for a 10°C error.

FIG. 18 RESPONSE OF A 0.003-INCH THERMOCOUPLE DURING A 7.9-MSEC-PERIOD TRANSIENT.

4.12 Fuel Temperature. To date, no fuel temperature measurements have been made in the CDC; however, measurement of fuel temperatures during a nuclear excursion using tungsten/tungsten-26 percent rhenium thermocouple wire encased in a magnesium oxide insulated stainless steel sheath have been attempted in other Spert facilities. The tests have indicated that several problems must be solved before the tungsten/tungsten-26 percent rhenium thermocouples can be used in a high-temperature environment with any degree of reliability. Typical problems encountered have been

- (1) The output of grounded hot-tip junctions became erratic during tests, indicating intermittent open circuitry within the thermocouple. This failure was probably due to the difference in thermal expansion rates between the thermocouple wire and the sheath material.
- (2) Ungrounded bare-wire junctions have given excessively large outputs. These large outputs are believed to be caused by radiation effects between the junction and ground. These effects are probably due to a radiation-induced, common-mode impedance change and/or voltage source and have been observed with other ungrounded thermocouples in a reactor during

nuclear excursions. To alleviate this radiation effect, bare-wire junctions with a ground wire attached to the junction have been tested. Although these grounded thermocouples have shown a well-behaved time history, the indicated temperature was well below what post-test metallurgical analysis established for a peak fuel temperature.

Other problems are also anticipated, and further development is required before the tungsten/tungsten-26 percent rhenium wires can be used for transient temperature measurements. These include the following:

- (1) The surface contact between the thermocouple junction and the oxide is difficult to control and will make repeatability between thermocouples very difficult. Post-test examinations have indicated a compaction of the oxide around the junction following a transient test. This means the degree of surface contact may vary as a result of testing, and a "breaking-in" period may be required.
- (2) The mass of the junction, particularly that of the grounded hot-tip probe, in conjunction with the low heat conductivity of the oxide fuel will create rather long time constants, probably 0.1 second or longer.
- (3) The insulation provided by mineral oxides used for high-temperature thermocouples deteriorates to less than 100 ohms at temperatures above 2000°C. This has the effect of shunting the thermocouple and reducing the actual emf output thereby resulting in an erroneous indication of the fuel temperature.

It appears that something other than the usual thermometry techniques should be investigated for this measurement area. We have considered (and have ready for testing) a sapphire, light-pipe-coupled, optical pyrometer detector. The sapphire rod serves as the light pipe from the fuel to an external fibre-optic light pipe which is connected to a fast-response (10 μ sec) optical pyrometer located external to the reactor. The detector will be limited to 2100°C for steady-state operation, since this is the melting point of sapphire; but it may be capable of going to higher temperatures on a transient basis.

4.2 Fuel Cladding Strain Measurements

Direct measurement of fuel cladding strain during a nuclear transient has been attempted using weldable strain gages attached directly to the cladding. Interpretation of these measurements with the weldable strain gages has been complicated by the following problems:

- (1) Static temperature compensation techniques are invalid during transient temperatures
- (2) Strain gages attached to the cladding will insulate the cladding and result in perturbation of the cladding temperature.

In order to reduce the difficulty in interpreting the results of direct cladding strain measurements, a device called a growth-ring was designed to measure

total radial fuel rod growth. The cladding strain is then determined from the differences between the measured total radial fuel rod growth and the thermal expansion of the cladding, which is calculated from the cladding surface temperature.

The construction of the growth-ring is illustrated in Figure 19. The growth-ring is machined from Invar, which has a low coefficient of thermal expansion, to reduce any thermal growth of the growth-ring. The fuel rod growth is transmitted to the growth-ring through three small areas of contact with the fuel rod. Four strain gages on the ring, which are connected in a full-bridge circuit to minimize temperature and nuclear radiation effects, sense the expansion of the ring caused by the fuel rod growth. Out-of-pile tests with the growth ring have determined that the temperature sensitivity is $0.08 \mu\text{in./in./}^{\circ}\text{F}$; the natural resonant frequency of the ring was also determined to be about 4000 Hz and limits the dynamic response capabilities to about $80\text{-}\mu\text{sec}$ rise time.

Growth ring data have been obtained from in-pile capsule tests. For all tests, the growth-ring indicated an overall growth in excess of that due to clad temperature expansion alone but indicated less than the upper limit of growth calculated for the UO_2 fuel expansion.

FIG. 19 PROTOTYPE FUEL ROD GROWTH TRANSDUCER.

APPENDIX A

TRANSIENT DATA INSTRUMENTATION SYSTEM

The following discussion is a description of the conditioning, transmission, and recording electronics system used in Spert I during the Oxide Core "destructive tests" series and, except for total channel capability, is typical of the electronics used at all the Spert reactors. The Spert IV system presently provides 70 channels of recording oscillograph data and 39 channels of magnetic tape data. In the near future, the number of magnetic tape channels will be expanded to a total of 70 channels through the use of a frequency-modulation technique of multiplexing multiple data channels on one magnetic tape track.

TRANSIENT DATA INSTRUMENTATION SYSTEM

The Spert I transient data are obtained from detector devices in the reactor and recorded as analog signals in real time at the control center. The low-level signals from the detectors are amplified and otherwise conditioned in the instrument bunker, which is located about 100 feet from the reactor, prior to transmission to the control center through 3000 feet of coaxial cable. The signals are then recorded on both optical oscillographs and magnetic tape at the control center.

The discussion of the transient data instrumentation is presented in two parts: (a) the conditioning and recording equipment description and (b) a measurements discussion. The conditioning and recording equipment description is presented first to familiarize the reader with those tools available for handling the data once it had been acquired. The measurements discussion, which is a description of the detector devices and their calibration or sensitivity determination, is then presented to show how the data was acquired. A brief performance evaluation also is presented at the conclusion of the discussion on each measurement.

The instrumentation system was capable of providing 105 channels of recording oscillograph and 65 channels of magnetic tape information on nuclear power, fuel rod temperatures, strain, and pressure measurements during an oxide core destructive test. Five decades of power to 100,000 MW and fuel rod cladding surface and internal fuel temperatures to 1500 and 2500°C, respectively, were covered with the necessary system capabilities to follow this information for a 1.5-msec-period test. Coverage to 100,000 psig with system rise time capabilities of 1 and 20 μ seconds was provided for the pressure measurements. The system performed to within ± 10 percent in determining the value of primary quantities of power, surface temperature, and pressure resulting from the test. The precision of the peak strain data was ± 30 percent.

1. CONDITIONING, TRANSMISSION, AND RECORDING ELECTRONICS

The following section is devoted to a description of the conditioning, transmission, and recording electronics used during the tests conducted upon the Spert I oxide core. This portion of the overall instrumentation system is discussed at this time to familiarize the reader with the conditioning and recording electronics and will not be further included in the measurements section except where the limitation of a measurement is determined by this equipment.

Figure A-1 is a block diagram of the complete layout of a typical channel of information showing the signal conditioning amplifiers, the transmission amplifiers, and the oscillograph and magnetic tape recorders. The signal conditioning amplifiers convert the detector signal to a usable voltage magnitude. This voltage signal is converted to a current signal by the transmission amplifiers and then transmitted to the control room where it is recorded by the recording oscillographs and the magnetic tape recorders.

FIG. A-1 BLOCK DIAGRAM OF A TYPICAL DATA CHANNEL.

Calibration of the electronics consisted of applying a calibration signal to the input of the conditioning amplifiers and determining the output of the recording electronics in terms of the input to the conditioning amplifier, eg, millimeters per millivolt for the recording oscilloscope and volts per millivolt for the magnetic tape.

The frequency response requirements of the quantities being measured and the bandwidth of the conditioning, transmission, and recording electronics were taken into consideration in allocation of the recording channels. The combination of signal conditioning amplifiers and recording method (oscilloscope or magnetic tape) determined the frequency response. In general, the frequency response fell into three categories: (a) 0 to 5 kcps, (b) 0 to 10 kcps, and (c) 0 to 20 kcps.

1.1 Signal Conditioning Amplifiers

Signal conditioning amplifiers were used to convert the low-level voltage, current or charge detector signals to a usable voltage magnitude. The signal conditioning amplifiers were housed in the instrument bunker and shielded from the reactor by several feet of earth and concrete to prevent generation of radiation-induced outputs in the amplifiers and to prevent radiation damage to the amplifier components.

The voltage input amplifiers were differential, low-level, wide-band dc amplifiers. The amplifiers used were manufactured by two different companies:

Video Instruments Company and United Electrodynamics Incorporated. The Video Instruments amplifier had gains of 25, 50, 100, 200, 400, and 800 and full-scale output of 1.0 volt with a noise level, referred to the input, of $30 \mu\text{V}$. The bandwidth was from dc to 10 kcps; at which frequency the output showed a loss in gain of 3dB (this frequency is commonly referred to as the -3 dB frequency). The United Electrodynamics amplifier had gains of 10, 25, 50, 100, 200, 400, and 1000; and the bandwidth was dc to 20 kcps ± 0.3 dB. These amplifiers were capable of 10 volts, full scale, and had a noise level, referred to the input, of 3 to 5 μV .

The type of amplifier, used for a specific measurement, was determined by the frequency requirements of the quantities being measured; ie, typically the Video Instruments amplifiers were used with thermocouples, and the United Electrodynamics amplifiers were used with strain gauges and pressure transducers.

Current measurements were made using two different techniques: (a) the standard method, using an electrometer with current ranges of 10^{-4} through 10^{-9} amperes with outputs of 1, 5, and 20 times the input range; and (b) two resistors were placed in series with the signal lead from the current source, and amplifiers were used to amplify the voltage drop across each resistor. This method provided coverage from 2×10^{-8} to one ampere. The second system would provide slightly more than 3 decades of current information from a single source. This was accomplished by attenuating the 10-volt outputs of the United Electrodynamics amplifiers by factors of 1, $3\frac{1}{3}$, and 10 and by the appropriate choice of amplifier gain and resistor sizes.

The input cable capacity limited the frequency response of both systems on the low-current ranges. The bandwidth of the electrometer amplifiers (dc to the -3 dB frequency) was 100 cps for 10^{-9} A, 1 kcps for 10^{-8} A, 7 kcps for 10^{-7} A, and 10 kcps for 10^{-6} through 10^{-4} A. The accompanying noise level was less than 50 mV on the electrometer output for 10^{-6} through 10^{-4} A and 100 to 200 mV on 10^{-9} through 10^{-7} A. The amplifier-resistor combination had a bandwidth of dc to a -3 dB frequency of 15 kcps with a 5000 ohm resistor and was limited to the amplifier frequency response of -0.3dB from dc to 20 dcps for resistors of 1000 ohms or less. The noise level was essentially amplifier noise for resistors of 1000 ohms or less and was less than 100 mV at the amplifier output for a 5000 ohm resistor. The better frequency response and lower noise level of the amplifier-resistor combination led to a preference of this technique over electrometers for current measurements.

Log coverage of current was obtained by amplifying the voltage drop across a resistor with a Kane Engineering Laboratories dc log amplifier. The log amplifier input range of 10 mV to 10 V produced a useful log current coverage of 10^{-6} A to 10^{-3} A with a response capability of dc to 20 kcps.

Measurement of the charge produced by the crystal pressure transducers and accelerometers was accomplished by using a United Aerotronics Corporation cathode follower in conjunction with a United Electrodynamics low-level amplifier. The cathode follower was used to convert the charge produced by the crystal transducer to a voltage which was then amplified to the desired amplitude by the low-level amplifier. The frequency response of this system was

from 5 cps to 20 kcps. The input sensitivity of the system was determined by the input cable capacity, the capacity setting on the cathode follower, and the amplifier gain setting.

1.2 Transmission and Recording Electronics

The basic philosophy used in this portion of the overall data gathering system was that the output of all conditioning amplifiers would be converted to a current signal by the transmission amplifiers and then transmitted to the control center via 3000 feet of RG-8/U coaxial cable. Since the data was transmitted as current, impedance change of the coaxial cable did not affect the magnitude of the signal that was recorded.

All data channels were recorded at the control center on recording oscillographs, with 52 of the channels also recorded on magnetic tape. Thirteen additional channels of magnetic tape information were provided in the instrument bunker. The signal for the control center magnetic tape recorder was taken across a resistor placed in series with the galvanometer. An isolation amplifier with a high common-mode rejection was required to isolate the magnetic tape electronics' ground from the circuit ground at the instrument bunker.

The transmission amplifiers were current driver amplifiers, manufactured by Offner Division of Beckman. These amplifiers nominally convert a one-volt input into a hundred-mA output. Normally the driver amplifiers had a frequency response of dc to ≈ 12 kcps; however, for the data channels that were also recorded on the magnetic tape recorders, the upper limit of the frequency response was improved to 20 kcps ± 0.5 dB. The output impedance of the driver amplifier was sufficient so that the load impedance could change from 50 to 70 ohms with less than a one percent change in the output current. A variable gain control on the driver amplifier was used to adjust the deflection of the galvanometers to three inches full scale, for a one-volt input to the driver. The drivers had a further characteristic of limiting the maximum galvanometer current to 110 mA and thereby preventing damage to the galvanometers.

Model 7-326 galvanometers, manufactured by Consolidated Electrodynamics Corporation (CEC) which had a sensitivity of 31.8 mA/inch for an 11.5-inch optical arm, were used. These units had a natural resonant frequency of 5 kcps and are fluid damped to 95 percent of optimum damping. These galvanometers were used in three 36-trace Model 5-119's and one 26-trace Model 5-114 CEC recording oscillographs. Figure A-2 shows the installation of these machines (one 5-119 not shown) in the Spert I control room. These machines used an incandescent light source and required a wet process for development of the film. For the destructive tests, the Model 5-119's were run at a paper speed of 160 inches/sec and the Model 5-114 at 115 inches/sec. The resulting traces on Kodak Ektaline 16 paper are illustrated in Figure A-3. Timing lines were provided every 10 msec with the 10th line accentuated. The overall trace width does not exceed 0.01 inch; wider traces are caused by instrument noise.

As previously stated, the signal for the magnetic tape recorder was developed across a resistor in series with the galvanometer. Since the input to the tape recorder electronics was single-ended, an isolation amplifier with high common-mode rejection was required to reduce ground loops between the reactor area and the control center. The isolation amplifiers were manufactured by the Offner Division of Beckman Electronics and had a common-mode rejection of 60 dB with a frequency response of dc to 20 kcps, ± 0.1 dB.

FIG. A-2 RECORDING OSCILLOGRAPH INSTALLATION.

FIG. A-3 OSCILLOGRAPH RECORD FROM THE SPERT I OXIDE DESTRUCTIVE TEST.

The tape recorder installation at the control center consisted of the three Ampex FR-600 transports, shown in Figure A-4, located in the tape recording room and one Ampex FR-1300 located in the Spert I control room. The transports used frequency modulated record amplifiers which change the input analog signal to frequency deviation about a center carrier frequency. At a tape speed of 60 inches/sec, the FM record amplifiers deviate up to ± 40 percent about a frequency of 108 kc, which provides a response from dc to 20 kcps, ± 0.1 dB. The FM signal is then recorded on one-inch wide magnetic tape. The magnetic tape accommodated 13 FM channels and one 100 kcps time channel which utilized direct record electronics.

FIG. A-4 MAGNETIC TAPE RECORDER INSTALLATION.

One additional FR-1300 transport was used in the Spert I bunker area. This machine provided five FM channels and eight direct record (250 cps to 250 kcps) data channels plus the 100-kcps timing track. The direct record channels were used for a qualitative rather than a quantitative measurement and were used in parallel with a FM channel. The direct record channels were to observe the high-frequency characteristics of the pressure pulse, should the rise time exceed the limits of the FM channels. This tape recorder was located in the instrument bunker so that the direct record input could be taken directly from the output of the conditioning amplifier and thus eliminate the need for a 250-kcps transmission amplifier.

The complete recording system consisted of 105 channels of galvanometer information. Fifty-two of these channels were also recorded on FM magnetic tape at the control center, with an additional five FM channels and eight direct record channels being recorded on magnetic tape at the instrument bunker.

Figure A-5 is a relative frequency response plot for the various combination of recorders and amplifiers; and, in summary, the overall system capabilities were as follows:

Linearity and accuracy	1 to 5 percent
Dynamic range	36 to 44 dB
Frequency response	
Galvanometers	3 kcps
Tape recorders	10 kcps (Video LL Amps) FM Channels
	20 kcps (UED LL Amps) FM Channels
	250 cps to 250 kcps -- Direct Record Channel

FIG. A-5 RELATIVE FREQUENCY RESPONSE OF THE VARIOUS COMBINATIONS OF RECORDING METHODS AND AMPLIFIERS.

1.3 Timing Information

Timing information for the recording system consisted of a real time reference and a sequence of events recorded on each of the oscillograph and magnetic tape records. The real time reference consisted of timing lines on the oscillograph records every 10 milliseconds and a 100-kcps crystal clock signal recorded on the 14th track of each magnetic tape. Loss of contact of the transient rod, the first event of interest, started the 100-kcps clock and was referenced as zero time. This event plus transient rod seat, each control rod loss of contact, and each control-rod seat event were recorded as unique steps on a rod drop trace on each recording oscillograph.

1.4 Calibration Procedure

The calibration of the detectors has been separated from the calibration of the electronics and the recording systems since the detectors were relatively inaccessible once they were installed in the reactor vessel. Following is a discussion of the calibration of the electronics and the recording system, while the calibration of the detectors will be covered in another portion of this report.

The primary reason for calibrating the electronic system was to determine the input sensitivity of the channel in terms of the output of the recording system, eg, millimeters per microampere or millivolt for the galvanometers and volts per microampere or millivolt for the magnetic tape. This was done on weekly basis by applying a one-second ramp of voltage or current to the input of the conditioning amplifier on each channel. The end of the ramp was clamped to within a known percentage of a specified value, allowing the full-scale output resulting from the ramp to be determined in terms of the input level. The ramp also allowed the linearity of the entire channel to be checked following each calibration. The accuracy of the calibration sources used is shown in Table A-I. In addition, it should be noted that the low-level voltage amplifiers were calibrated on one gain setting only, normally a gain of 100, since the gain stability between fixed gain steps was within 1/2 percent for extended periods of time. The current input devices were calibrated on all ranges to be used.

The desired full-scale deflections (ie, 3.0 inches from the recording oscillographs and -1.0 to +1.0 volt for the magnetic tape recorder) were adjusted by means of a variable gain control on the current driver amplifier. Between calibration periods, any change in channel sensitivities was accomplished by the fixed gain steps of the conditioning amplifiers.

Periodically, 3 to 6 months, the frequency characteristics of the conditioning and transmission electronics were measured. Typically, the frequency characteristics would remain constant within 3 to 5 percent; and thus the weekly static calibrations, in conjunction with the periodic dynamic calibration checks, provide the overall system calibration. The frequency response of all channels was determined in the month prior to the destructive test.

TABLE A-I

CALIBRATION SOURCE ACCURACIES

<u>Calibration Source</u>	<u>Range</u>	<u>Percent Error</u>
Current	10^{-4} Amps	+2 to 0 percent
	10^{-5} Amps	+0.9 to +0.7 percent
	10^{-6} Amps	0 to -0.1 percent
	10^{-7} Amps	0.2 to -0.6 percent
	10^{-8} Amps	-0.97 to -1.17 percent
	10^{-9} Amps	-1.15 to -1.35 percent
Voltage	1 mV	+2.4 to +0.4 percent
	3 mV	+1.16 to -0.84 percent
	10 mV	± 0.1 percent
	30 mV	-0.117 to +0.017 percent
	100 mV	+0.1 to 0 percent

APPENDIX B

DATA REDUCTION AND PROCESSING

The transient data as recorded by the instrumentation system was presented in two forms: (a) frequency modulated data recorded on magnetic tape and (b) optically recorded light traces on oscillographic paper. For the destructive tests, all magnetic tapes were recorded at a speed of 60 inches/sec. The oscillograph records were recorded at 160 inches/sec or 115 inches/sec depending upon the model of the oscillograph.

Reduction of the magnetic tape is accomplished by a Digital Data system which can provide analog plots of the data or digital magnetic tapes for computer input. The oscillograph records were reduced in two fashions; analog to analog by means of a curve follower or analog to digital by means of a manually operated curve follower.

DATA REDUCTION

The transient test data from both the FM magnetic tapes and the oscillograph records were reduced, as described in the following sections, to convenient physical units of a form useful for processing and analysis of the data. Figures B-1 and B-2 show a block diagram of the reduction procedures for both forms of recorded data.

FIG. B-1 BLOCK DIAGRAM OF THE REDUCTION OF THE MAGNETIC TAPE DATA.

FIG. B-2 BLOCK DIAGRAM OF THE REDUCTION OF THE OSCILLOGRAPHIC RECORD DATA.

1. MAGNETIC TAPE DATA

The FM magnetic tapes recorded on the Ampex F. R. 600 recording transport are reproduced one at a time on a similar transport. This unit converted the FM tape data into analog voltage signals in real time and in time base expansions of 2, 4, and 8. Next, the analog signals were amplified 2.5 times and filtered with a time-delay filter (approximating a Bessel response) by means of the filter-amplifier unit. The filter had an effective cutoff frequency of $1/2 \Delta t$, where Δt was the time increment to be used in digitizing the analog data.

The Analog-to-Digital (A/D) Converter accepted the analog signals from the Filter-Amplifier and converted them to digital units in a binary coded decimal (BCD) format--a three-decimal digit with a sign. The minimum digitizing time increment used was one point every 20 μ sec. The maximum digitizing time increment was one point every 990 msec. The digitizing rate depended on the amount of data needed from each transient. The digitized data were then stored in the Magnetic Buffer Unit.

The Magnetic Buffer Unit has a capacity for storing 2048 BCD words, where the first 16 words were used for control and index information, the next 32 words were used for the initial conditions of the data channels, and the remaining 2000 words were used for storing the transient data. The output of the Magnetic Buffer Unit consisted of an oscilloscope display, analog plots, and digital magnetic tape. For oscilloscope display and plotter output, the BCD data were processed through a digital-to-analog converter. In practice, the data from the entire transient were digitized and observed on the oscilloscope to determine the region of specific interest, and then the data were re-digitized over the limited area of interest to a new 2000-word section with the appropriate time increment. A Digital Magnetic Tape Recorder transferred the data from the Magnetic Buffer Unit to a magnetic tape for eventual use as computer input.

The combined errors of the analog-to-digital conversion system, the digital-to-analog conversion, the inherent error of the analog plotter, and operation of the analog plotter were less than two percent, provided the output of the digitized magnetic-tape data was greater than 0.1 volt. The general criterion for useful data is that which has a maximum output from 0.1 to 2.0 volts.

2. OSCILLOGRAPH RECORD DATA

Shortly after the transient tests, the maximum value of each transient test variable was determined by measurement of the oscillograph traces with a millimeter rule, and the measurement was converted to physical units by the appropriate calibration factor. These values were entered into a table used for comparison and reference of all the experimental data for each transient. Next, the data were reduced through an analog-to-analog and analog-to-digital conversion system. In both of the conversion systems, the X-Y coordinates could be varied in such a way that the processed data were available in the desired physical units.

The analog-to-analog conversion system consisted of a manually operated analog curve reader that transferred the data directly from the oscillograph

records to an analog X-Y plotter. Most of the data was transferred linearly; however, some of the reactor power data was transferred logarithmically by employing a function generator. Depending upon the magnitude of the oscillograph traces, the operational errors of the analog-to-analog curve reader and the inherent error of the analog plotter combine to a total overall error of less than two percent.

The analog-to-digital conversion system consisted of a manually operated analog curve reader and a digital converter with tabular and punched-card outputs. The data on the punched cards could be plotted on a digital X-Y plotter or processed through the IBM-7040 computer.

The operation of the analog curve reader introduced an error estimated to be $\pm 1/2$ mm (or about ± 0.7 percent of the full-scale deflection of 76 mm) due to the parallax encountered in positioning the curve readers on the oscillograph traces. The estimated errors of the analog-to-digital conversion, the inherent error of the digital plotter, and the operation of the digital plotter all combine with the above error to an estimated overall error of less than five percent, provided the deflection of the original analog curve was greater than 10 mm. The general criterion for useful data is that which has a maximum deflection from 10 to 76 mm.

UNITED KINGDOM ATOMIC ENERGY AUTHORITY

REACTOR GROUP

THE ORGANISATION AND ADMINISTRATION OF THE
IRRADIATION PROGRAMME FOR THE FAST REACTOR
DOUNREAY EXPERIMENTAL REACTOR ESTABLISHMENT

G.M. JONES
U.K.A.E.A.

"We must (indeed all hang together, or, most assuredly we shall all hang separately"

Benjamin Franklin

Remark to John Hancock at the signing of the Declaration of Independence, 4th July, 1776.

INTRODUCTION

1. I think that the above quotation by Benjamin Franklin could be a most appropriate 'motto' for all concerned with the carrying out of irradiation experiments. The mounting of an irradiation experiment is one of the most costly experiments to carry out and involves theoretical scientists, reactor physicists, design engineers, production engineers, assembly engineers, reactor operators and post irradiation examination scientists, all with their own particular view point. If an irradiation experiment is to proceed smoothly through all these many sections, then very detailed planning and co-ordination must be done and clear short lines of communication must be working throughout.

2. Because of the complexity of any irradiation experiment one must also be absolutely sure that the experiment one is considering carrying out is the correct one for once the decision is taken to carry out the experiment it can take up to nine months to prepare, two years to irradiate and then a further six months before the results are analysed.

3. At the Dounreay Experimental Reactor Establishment, using our present organisation, we find that for

- (a) An experiment using existing equipment it takes up to eight weeks to prepare for irradiation
- (b) A new experiment from conception stage to commencement of irradiation takes up to nine months depending on the complexity of the experiment
- (c) Priority experiments can be prepared for irradiation within three - four months from the final specification being achieved

4. The present programme of irradiation experiments in support of the Prototype Fast Reactor Development Programme commenced in September, 1961. The first experiments were loaded into the Dounreay Materials Reactor in December, 1961 with further experiments being loaded into the Fast Reactor in April, 1962. Since this date the two irradiation programmes have been co-ordinated by the Assistant Director (Fuels) through the Irradiation Project Office to the changing requirements of the scientists carrying out the development work.

5. In order to carry out such a large programme of development, design, manufacture and assembly, involving many specialist skills, a flexible organisation was evolved to co-ordinate the work of the many sections

involved in order to meet the preset target dates for reactor loading. At the Dounreay Experimental Reactor Establishment we have evolved a Project Organisation which we believe gives the maximum utilization of the specialised skills of the staff available.

AIMS (OR TERMS OF REFERENCE) OF THE PROJECT OFFICE

6. For a Project Co-ordinating Office to be able to carry out its function efficiently, it must have clearly defined terms of reference or aims. The following are the terms of reference of the Dounreay Irradiation Office:-

- (a) To co-ordinate the mounting of experiments from the initial enquiry by a Sponsor through all the stages of the experiment until the final report is written on the results. The co-ordination is such as to do all necessary to meet pre-set irradiation dates.
- (b) To use such management techniques as may be of value to improve the efficiency of the mounting of experiments e.g. Work Study Techniques, Critical Path Analysis Networks for sub projects.
- (c) To ensure smooth communication between all concerned within the Project.
- (d) To examine, at regular intervals, procedures and organisation in use to ascertain what modifications if any, could be incorporated to improve the efficiency of mounting experiments.

OUTLINE OF THE ORGANISATION

Preparation of the Irradiation Experiments for a Specific Reactor Date

7. A Sponsor requiring to carry out an irradiation experiment in the Fast Reactor first of all writes to the Assistant Director (Fuels) outlining the need for the experiment and the conditions required for the experiment. The request is passed to (a) the Project Manager Fast Reactor for his information and approval in principle for the experiment, and (b) the Feasibility Group (a section of the Fast Reactor Technical Services Group) to explore the best way of carrying out the experiment (after consideration that the Fast Reactor is the most suitable facility for doing the experiment). A summary of the conclusions reached is sent to the Assistant Director (Fuels) for his consideration and approval.

8. If approved, a meeting of the Technical Evaluation Panel is convened by the Project Office which consists of the following members nominated by the various departments:-

Assistant Director (Fuels)
Fast Reactor Operations Group
Experimental Rig Design Office
Experiment Assembly Section
Post-Irradiation Examination Section
Feasibility Group
Central Project Office
Sponsor of Experiment

The purpose of the Panel is to:-

- (a) Discuss the experiment proposed in detail from their specialist viewpoint.
- (b) Enable each group involved in the experiment to be fully informed of the experiment from its initiation.
- (c) Obtain a coherent plan of action to build the experiment for an agreed date with all sections concerned.

Action lists produced from such meetings start the experiment off for a particular reactor period.* The experiment is then given a D.F.R. number to identify it throughout its life, this number being quoted on all documents and drawings referring to the experiment.

9. The Central Project Office maintains a master register of D.F.R. numbers issued for all experiments, and also numbers given to the various development experiments of the driver charge fuel. This register acts as a condensed "history" of the Fast Reactor since its inception as a reactor (i.e. driver fuel development) and as an irradiation facility (experimental programme development).

10. Having been noted on a given reactor period loading schedule the experimental design manufacture and testing is progressed through the Central Project Office at fortnightly Progress Meetings along with other experiments schedules for the same reactor period. Also running concurrently with the manufacture of the experiment, the safety clearance for the irradiation of the experiment is prepared and submitted through the Central Project Office to the Fuel Element Approval Panel. The proposal for an irradiation experiment is a summary document of the Design Report and is appended to the Proposal. If the experiment is approved the proposal is signed by the Assistant Director (Operations) and the Director, D.E.R.E., the signed Master Proposal then being held in the Central Project Office for reference.

11. In order to be able to plan experiments satisfactorily for a given reactor start-up it is necessary to issue a timetable for each period at least NINE months ahead of the reactor start up (i.e. three reactor periods). This timetable is issued throughout the Authority for information and action and the initial dates are strictly adhered to for planning purposes.

To Maintain Full Records of Each Experiment

12. Each experiment has an identification number, i.e. a D.F.R. number and a pin number (if a fuel experiment) or a D.F.R. number only (if a materials test). This number is placed on a file and the aim is that the file should go everywhere with the experiment so that all recorded information, concessions and other paperwork associated with the experiment can be placed in the folder immediately and kept in one place. After final inspection and clearance of the experiment the file is sent to the Central Project Office for filing in the D.F.R. number experiment file.

13. In the case of fuel pin manufacture, set forms have been devised to record easily the data on the fuel pin assembly. The data recorded are that which has been agreed with all concerned as being information that warrants recording. The data sheets are devised in such a way as to enable a clerk to transfer the information on to a punched card for reference.

*I should like to explain here that the Dounreay Fast Reactor operating period lasts for approximately ninety days at full power with thirty days shut down for changing fuel and replacing experiments.

14. Initially the original documentation was compiled into Data Compilation reports and these were issued to agreed 'centres' within the U.K.A.E.A. Centres were agreed to be more useful than issuing to persons as the tendency with the latter was to bury the files in their own filing system, other people complaining that they should have their own copies!

15. Because of the rather large bulk of each data compilation it was agreed to condense the data on each pin and vehicle on to a punched card and thus issue this to each centre. The punched card is so devised that it can be used against the Master D.F.R. number system and be a complete summary of the development of the Fast Reactor.

DISCUSSION

16. As can be seen from the above, the organisation is projectwise making use of each discipline as required from the various groups within the site, the Project Office, co-ordinating and progressing the experiment from section to section i.e. from design to manufacturing orders to delivery of components and so on.

17. At one time the Project Office used to make out detailed bar charts for each experiment but now it issues target dates only for the various stages for all experiments after an assessment of the whole work load has been carried out. This method has been found more convenient in that it allows each section (a) to organise its own work within its section in order of priority, (b) the target dates contain a 'slack' which allows for the non completion by the target date of the previous activity (c) using a target date rather than a rigid time scale allows for more flexibility between two dates, knowing that the Sponsors sometimes change their minds even when the preparation of the experiment has commenced.

18. An irradiation experiment can be broken down into five major activities.

- (i) The scientist finding the need for irradiation information and detailing his requirements.
- (ii) Preparation of the irradiation experiment.
- (iii) Irradiation of the experiment.
- (iv) Post Irradiation Examination.
- (v) Formal Report on the experiment written by the Sponsor.

Each of the above major activities can be broken down into a large number of minor activities necessary to complete the final activity. It follows that the lines of communication or communication channels can be very tortuous and wasteful if each Sponsor has to try and follow his own experiment through all stages. The Project Office responsibility is to act as a link between a Sponsor and the 'workers' keeping up to date checks on all aspects of all experiments in their various stages and informing Sponsors when necessary. Thus the use of a Project Office reduces to the minimum the lines of communication and therefore I think improves the efficiency of the organisation. In this context of communication, the Project Office can be used by all to obtain any information that they require. Correct use of a central office for co-ordinating irradiation experimental work i.e. having a "live project" office can be a most useful tool to keep the impetus and interest 'alive' on the part of all staff. In this context the office must pass on information on policy to the lower grades of staff i.e. keeping them informed, and also pass 'up the line' information and ideas from the shop floor as it were. Also, information obtained from other sources which is considered

relevant must be cross fertilized through the organisation.

SUMMARY OF THE EXPERIMENTS DEVELOPED AND USED
FOR FAST REACTOR WORK TO DATE

The Facility

19. The plan of the reactor is shown in Figure 1.

In general the use of replaced fuel element positions (R.F.E.) requires that the experiment must externally resemble the fuel element it replaces so that there are no location problems and standard lifting gear can be used. The five types of position available are listed in Table 1.

	Number	O.D. (ins.)	Useful length (ins.)
Core R.F.E.	342	0.752	21
Reflector R.F.E.	294*	1.36	84
Central Hexagon	1	5.2	21
B.10	6	0.9	84
Control Rod P.30	4	0.752	21

* Inner Breeder Only

Table 1. Irradiation Facility Dimensions

20. Table 2. gives the main reactor parameters:-

$$\text{Central Flux} = \frac{\text{Table 2 selected reactor data}}{2.5 \times 10^{15} \text{ n/cm}^2 \text{ sec. full power}}$$

Cross Sections and Reaction Rate

Material	Fission Cross-Section barns	Capture Cross-Section barns	Reaction Rate Core Centre w/gm	Reaction Rate Reflector w/gm
^{235}U	1.43	0.25	280	210
^{238}U	0.09	0.16	17	6
^{239}U	1.78	0.16	360	230

Coolant Parameters

Mixed mean inlet	230°C
Reactor vessel outlet	320°C
Main driver charge outlet	390°C
Core pressure drop	10.2 lb/in ²
Reflector Pressure drop	0.27 lb/in ² 0.38 lb/in ²

Experimental Elements

21. Attached in Appendix 1. are a few examples of designs which have been used for irradiation experiments in the Fast Reactor. To date it is estimated that over sixty different designs of experiments have been produced to suit the various requirements of sponsors.

DEVELOPMENT OF THE ORGANISATION

22. There are a number of tools of management which I think should be examined to ascertain how they might assist in the management of irradiation programmes. I will simply list my suggestions with one or two comments.

- (a) Use of Critical Path Networks for development sub projects
- (b) Use of Critical Path Analysis for resource planning of available effort - this has been used at the Dounreay Experimental Reactor Establishment for planning the transfer of the welding laboratory from one site to another. Could such a system be used for allocation of scientists to various projects within a site?
- (c) Value Analysis - this science is in its infancy but I think it has possibilities in the early stages of discussion on experiments to ascertain whether the results one will obtain are worth the money to be invested in the experiment.
- (d) Finally, I should like to say that all concerned in the mounting of irradiation experiments should examine their procedures at regular intervals and alter accordingly in order to be as efficient as possible, for irradiation experiments are amongst the most costly experiments carried out for the results obtained.

23. As the policy of the United Kingdom Atomic Energy Authority is now to make available space for sponsors throughout the world to irradiate their experiments in the Dounreay Fast Reactor the experience and organisation built up over the last few years will be made available to facilitate the sponsor's requirements.

FIG. 1. PLAN OF REACTOR

FIG. 2. VEHICLE FOR IRRADIATING THREE EXPERIMENTAL PINS IN ONE FUEL ELEMENT

FIG. 3. INSTRUMENTED ZIRCALOY TUBE EXPERIMENT

84-3320-1

FIG. 4. EXPERIMENTAL FUEL SPECIMEN MOUNTED WITHIN A 'DRIVER CHARGE' FUEL ELEMENT.

FIG. 5 . SPRING CREEP IRRADIATION RIG

8.4.11

EXPLODED VIEW OF 77 PIN SUB-ASSEMBLY

CAPSULE IRRADIATION EXPERIMENTS AT CHALK RIVER

by

A.W. Boyd

INTRODUCTION

This paper surveys work done at Chalk River Nuclear Laboratories since the meeting in this subject in Maryland in October 1963 [1]. Two other reviews [2] of in-core instrumentation at CRNL were given in June 1964.

Opposite each item discussed is given the name of the person primarily responsible and references giving the names of all those concerned are given wherever possible.

I NOVEL CAPSULE DESIGN

(1) Capsule for Continuous Measurement of In-Reactor Creep [3] -

V. Fidleris

The Mk IV in-reactor creep machines [1] have been used for 13 creep tests of zirconium alloys at times to 3000 hr, fast neutron fluxes to $1.5 \times 10^{13} \text{ n cm}^{-2} \text{ sec}^{-1}$, and temperatures from 200 - 400°C. Creep rates of the order $10^{-6} - 10^{-5} \% \text{ hr}^{-1}$ have been measured. Four of these machines are now operating in the NRX reactor.

The Mk VI creep machines shown in Figure 1 have operated at 300°C at a fast flux of $5 \times 10^{13} \text{ n cm}^{-2} \text{ sec}^{-1}$ for 3000 hr. In these instruments temperature control to within

$\pm 2^{\circ}\text{C}$ is achieved by simultaneous electrical heating and water cooling, aluminum and copper fins around the specimen and enclosing the specimen and the loading bellows in helium. The strain is measured by a pneumatic gauge with a sensitivity of 10 micro-inches. These instruments are designed to operate at temperatures up to 500°C and to measure strains up to 30%.

A computer controlled automatic readout system is now being made which should increase the accuracy by a program for statistical reduction of the data.

These instruments were manufactured by Hawker-Siddely Canada Limited.

(2) Biaxial Creep Measurements - E. Ibrahim

Zirconium alloy tubes stressed by pressurized water are to be irradiated at fast neutron fluxes up to $5 \times 10^{13} \text{ n cm}^{-2} \text{ sec}^{-1}$ as shown in Figure 2. The bore will be measured during reactor shutdowns at two month intervals. An air gauge accurate to within ± 50 micro inches over the range 0.90 - 0.91 inches has been designed and tested [4].

(3) Pressure Measurements in Fuel Capsules [5] - J.R. MacEwan
M.J.F. Notley

The pressure due to fission gas release in irradiated UO_2 has been measured using a Booth-Cromer null balance switch. The details are shown in Figure 3. The diaphragm is maintained in a null position by pressure applied through the tubing. The sensitivity of this switch is 1 psi with a bias of less than 3 psi. The free volume on the fuel element side of the diaphragm is about 100 microliters. These assemblies have operated satisfactorily at 250°C and pressures up to 100 atmos over irradiations of 6 months at $4 \times 10^{13} \text{ thermal n cm}^{-2} \text{ sec}^{-1}$.

(4) Gaseous Corrosion Facility - B. Cox

This consists of an annular furnace mounted at the end of a tube extending above the reactor shielding. Samples are inserted through a seal at the top of this tube and suspended in the furnace. Different gases can be circulated and the temperature maintained at $300 - 350^{\circ}\text{C}$. This assembly has been used inside annular uranium rods at fast neutron fluxes up to $1 \times 10^{13} \text{ n cm}^{-2} \text{ sec}^{-1}$.

(5) Thermoelectric Fuel Element Capsule - R.D. MacDonald

Work at Chalk River [6] has shown that an electromotive force can be generated between uranium oxide pellets and the molybdenum sheath of a central thermocouple. To study this effect the capsules shown in Figure 4 have been made and are to be irradiated at a thermal neutron flux of $6 \times 10^{13} \text{ n cm}^{-2} \text{ sec}^{-1}$. The molybdenum disks and the UO_2 pellets form a series of thermocouples with the hot junctions at the centres and cold junctions at the surface. Voltages and resistances will be measured by the leads shown. Temperatures will be determined by simultaneous irradiation of capsules with thermocouples containing similar Al_2O_3 tubes, UO_2 pellets and Mo disks.

(6) Organic Coolant Irradiation Capsules - A.W. Boyd

The stainless steel capsules shown in Figure 5 with re-entrant stainless steel clad thermocouples have been used for irradiating 5 g terphenyl samples. The samples are out-gassed and sealed in vacuo by crimping the 0.125-inch diameter filling tube. The thermocouples agree within $1 - 2^\circ\text{C}$ at dose rates of 1 W/g .

In these irradiations temperatures of $350 - 450^\circ\text{C}$ were desired. These were achieved by radiation heating of the capsule and the insulation afforded by the air gap between the capsule and the inner wall of the rod. Different temperatures in this range were obtained by wrapping thin sheets of Zircaloy around the capsule. Because of its low thermal neutron cross section Zircaloy did not change the γ -ray dose rate in the capsule.

(7) Capsule for Fission Yield Measurements [7] - R.C. Hawkings

To measure the yield of fission product gases the capsule in Figure 6 has been used. The fissionable element is deposited as the nitrate on an aluminum foil and covered with 30 mg/cm^2 of barium stearate, which absorbs the energy of the fission recoils and allows the fission gases to escape. Irradiation times have been of the order of 100 seconds at $1 \times 10^{12} \text{ thermal n cm}^{-2} \text{ sec}^{-1}$. The capsule can be removed from the reactor and the gas collected for counting 50 - 100 seconds after the end of the irradiation.

Figure 3 - Capsule for Fuel Element Pressure Measurements

Figure 4 - Capsule for Thermoelectric Measurements of UO_2 - Mo Elements

Figure 5 - Capsule for Organic Coolant Irradiations

Figure 6 - Capsule for Fission Gas Yield Measurements

FLUX DETERMINATION AND CONTROL

(1) Dosimetry [8] - A.W. Boyd

Adiabatic calorimeters have been used to measure dose rates from 0.02 - 0.7 W/g in aluminum, beryllium, graphite, polyethylene and polystyrene and Zircaloy-2. In one series of tests graphite was used in place of aluminum for the jacket. The dose rate values for the sample were within 5%. The thermocouples are inserted in the plastic samples by melting and in the graphite by packing with powdered graphite. They are attached to the metals by peening or spot welding. It was found that these two methods gave the same temperatures (in Zircaloy-2) at a dose rate of 0.7 W/g.

Measurement of ion chamber currents is easy and rapid. Ion chambers have been developed for measuring γ -ray and fast and thermal neutron fluxes. The γ -ray chambers have graphite electrodes and CO_2 at around one atmosphere and have been used from 0.1 to 500 m W/g. Good saturation characteristics at 500 - 1000 v up to 0.5 W/g have been obtained with a 0.06 inch gap between concentric graphite electrodes with an inside diameter of 0.5 inches. The most satisfactory envelope is quartz. Epoxy cement may be used as a seal for low temperature low dose rate measurements and quartz-molybdenum seals for higher temperature measurements. Stainless steel envelopes have been used with the high voltage lead brought in with stainless steel sheathed magnesium oxide insulated wire. However, comparison of these with the quartz sealed chambers has shown that the response to thermal neutrons is anomalously high.

A chamber with stainless steel electrodes and 200 psi of nitrogen has been used for measuring relative thermal neutron fluxes. Up to 75% or more of the current is due to thermal neutrons - largely the $\text{N}^{14}(\text{n},\text{p})\text{C}^{14}$ reaction and the γ -ray response can be corrected for by using a graphite- CO_2 chamber.

For relative fast neutron flux measurements a chamber with graphite electrodes and 100 psi of helium has been used. Inside an annular uranium fuel element about half the current is due to elastic scattering of fast neutrons by the helium.

The radiolytic yield of hydrogen from cyclohexane is the same within 5% for fast neutrons and γ -rays at temperatures from 0° to 150°C. Measurements using both cyclohexane and cyclohexane-d₁₂ have been made at dose rates from 0.001 - 3 W/g using 2 - 10g samples sealed in quartz capsules. As ~10 times more energy is deposited by fast neutrons in cyclohexane than in graphite comparison of ion chamber and cyclohexane measurements gives both fast neutron and γ -ray fluxes. Unfortunately, the difference between fast neutron contributions in normal and deuterated cyclohexane is not great enough to give an accurate measure of fast neutron and γ -ray fluxes.

The results from calorimetry, ion chambers and cyclohexane dosimetry agree within 5%.

(2) Capture γ -Ray Fields in Absorbers - A.W. Boyd

The dose rates due to thermal neutron capture γ -rays inside annuli of stainless steel, silver and cadmium have been calculated and measured [9]. The measurements were made with graphite-CO₂ ion chambers. The capture γ -ray dose rate inside cadmium in a thermal flux of 1×10^{13} n cm⁻² sec⁻¹ is 0.1 W/g.

(3) Decay of the γ -Ray Dose Rate After a Shutdown - A.W. Boyd

The γ -ray dose rate has been measured from one second to several days after shutdown of NRX using graphite-CO₂ ion chambers [10]. Measurements were made in annular uranium fuel elements in empty lattice positions and in the graphite reflector in the NRX reactor. These values provide a measure of the relative contributions from fission, fission product decay and capture γ -rays.

(4) Self-Powered Thermal Neutron and γ -Ray Detectors [11] -
J.W. Hilborn

In-core neutron detectors depending on the short-lived β activation of rhodium ($t_{1/2} = 42$ s) or vanadium ($t_{1/2} = 3.8$ min) are being used in the NRU and NRX reactors at Chalk River. These are shown in Figure 7. The current produced by the escape of the β particles from the rhodium or vanadium to the sheath is measured by an electrometer. The neutron sensitivities are 2.3×10^{-6} amp per 10^{14} n cm⁻² sec⁻¹ for a 20 cm

rhodium detector and 1.4×10^{-6} per 10^{14} n cm^{-2} sec^{-1} for a 300 cm vanadium detector.

These detectors have been exposed to neutron fluxes up to 3×10^{14} n cm^{-2} sec^{-1} for more than a year. Over this period out of a total of 64 Rh and 5 V detectors only 3 Rh detectors failed.

Where several detectors are used the current signals are scanned by a multiplexer and a PDP-5 on line computer is used to correct for background and detector burn up and to calculate absolute thermal neutron flux.

Self-powered neutron detectors with a prompt response are being developed based on the electrons from scattering of capture γ -rays. Prototypes using cobalt and cadmium have given encouraging results.

(5) Factors Producing Flux Variations with Time [12] - G.C.Garrow

Continuous measurements of the thermal neutron flux using self-powered detectors have been made for several months at different positions in the NRX and NRU reactors. In NRX ten detectors spaced over the height of the moderator (10 feet) were used in two different empty lattice positions. In NRU eight rods each with eight detectors were used.

These measurements show the effect of moderator height, adjacent fuel rods and cobalt rods on the neutron flux, over the height of the core of the reactor.

IV CAPSULE COMPONENTS AND FABRICATION

(1) Thermocouples - A.J. Otter

Much of the in-reactor work at Chalk River is concerned with materials behaviour in pressurized water or steam. In these conditions alumina insulated ISA type K thermocouples sheathed in 347 stainless steel are used as the 347 resists corrosion better than 304 s.s. and the Al_2O_3 , providing it is of high purity, does not swell as much as MgO in contact with water. Diameters of 0.0625, 0.04 and 0.02 in. are used although

the last tend to fail due to grain growth in the sheath. These thermocouples are brazed to Zircaloy using a Zirconia-5% beryllium alloy.

For measuring temperatures inside UO_2 fuel elements W - W Re thermocouples insulated with alumina and sheathed in molybdenum are used. These have operated satisfactorily to 1700°C though some have opened up intermittently at the junction at this temperature.

Time-domain reflectometry [13] has been applied to location of faults in thermocouples. If comparison with an identical sound thermocouple is available then the fault can be located within 10% for circuits with up to 75 feet of unshielded extension wire and 25 feet of 0.0625 inch sheathed thermocouple wire.

V REACTOR CHARACTERISTICS AS THEY AFFECT EXPERIMENTS

(1) Facilities for Short Irradiations - A.W. Boyd

There are often short irradiations where the sample size or the need to control and measure temperature do not allow pneumatic insertion and removal. Facilities for installing and removing such samples in vertical and horizontal access holes to the core with the reactor at full power have been built. One of the vertical facilities is shown in Figure 8.

ACKNOWLEDGMENTS

The author wishes to thank all those whose names are listed in the text and references for their generous help in preparing this review.

Figure 7 - Self- Powered Thermal Neutron Flux Detectors

Figure 8 - Facility for Short Irradiations

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CAPSULE IRRADIATIONS IN THE HANFORD PRODUCTION REACTORS

by

C. E. Bowers

DOUGLAS UNITED NUCLEAR, INC.

Richland, Washington

CAPSULE IRRADIATIONS IN THE HANFORD PRODUCTION REACTORS

INTRODUCTORY NOTES

The role of the Hanford reactors in the production of plutonium has been well publicized in recent years, and there are few people remaining who believe that reactors make nylon stockings. Not so well known, however, is the fact that a program of test irradiations has been conducted at Hanford almost since Day One. My purpose today is to acquaint you, within the limitations of security restrictions, with that program--past and current. My presentation will not be highly technical, but it may be noteworthy in the sense that it illuminates a problem that may be unique to Hanford--that of co-existence of production and experimental programs.

Several channels were provided by farsighted early designers for future materials testing. These penetrate the reactor core horizontally, and provide a wide variety of irradiation conditions. They may be used for experimental purposes under AEC approval. Cooled channel sizes range from 3/4" to 3-1/4" ID, and up to about 35 feet long; capsule "trains" up to about 15' length have also been accommodated. Uncooled bare channels (up to 4-3/16") are also available. Practically any material may be irradiated if encapsulation requirements are met.

And practically every material has been irradiated--we have irradiated everything from wheat to bone ash, from soil to sea water, from aluminum to plutonium. Our tests have been as small as a so-called "Quickie" irradiation of sub-microgram quantities of material, and as large as to require a 200 ton external assembly designed to capture the wily neutrino. We've made a few isotopes, too, at the request of the Isotopes Sales Department of Oak Ridge National Laboratory--in fact, about 1.5 million curies of 39 different isotopes. Among these have been more than a mega-curie of Cobalt-60, and virtually all of the Carbon-14 that has been produced in the United States.

GENERAL IRRADIATION DESCRIPTIONS

Experimental capsules generally fall into one or two categories:

1. Radiation Effects on Materials
2. Reactor Fuels Development

There is, naturally, a wide variety of capsule designs.

A. Radiation Effects on Materials

1. Considerable work is routinely done in graphite research (usually long-term irradiations). Test samples, residing in graphite "boats" are irradiated in uncooled channels. The sample boats sit free in the channels during irradiation, and are discharged during reactor shutdown.
2. Small samples of various materials are irradiated for radiation effect studies or, occasionally, for activation analysis of extremely short half-lived materials (for Battelle Northwest). Hazardous materials are doubly encapsulated, with the

sample in a quartz ampoule which is in turn encapsulated in aluminum; "benign" materials require aluminum encapsulation only. Samples may be charged and discharged during reactor operation. Capsules may be instrumented and equipped with variable gas atmosphere controls, and may contain fissionable materials.

B. Reactor Fuels Development

Fuel program experiments include anything from small, simple capsules to complex, multi-unit assemblies up to 15' long. Irradiation environment is widely flexible, and has included:

1. Pressurized coolant (up to 2200 psig)
2. Temperature control (specimen temperature up to 1600°C)
3. Gas pressure control
4. Complex instrumentation
5. "Tailored" flux
6. Charge-Discharge during operation
7. Coolant pH control

UNIQUE CAPSULES AND IRRADIATIONS

Some irradiations have been conducted at Hanford that may be considered of unusual interest or even unique; a few of these follow:

1. Biological and Medicinal

- A. Yttrium-90 was produced for use in transnasal hypophysectomy, an operation in which the pituitary gland is burned out to reduce pain in terminal breast cancer patients.
- B. Cancerous and non-cancerous tissue was irradiated for Reed College in support of a program to determine elemental differences using activation analysis techniques.
- C. Nine thousand fish tags were irradiated in one run, thanks to our high volume capacity. These tags were made of small Alnico magnets (chosen for closely controlled cobalt content) and were installed in the body cavities of herring for studies of migration habits.
- D. Small samples of urine were irradiated to determine fissionable material content.

2. Fuel Technology

- A. Candidate fuel specimens for space reactors were irradiated to determine performance and behavior in support of the ROVER program.
- B. Plutonium and uranium oxide fuels specimens have been irradiated to provide high temperature redistribution effects to permit heat transfer calculations at various exposure states.
- C. Several large, highly instrumented capsules containing SNAP-8 candidate fuel specimens have been irradiated for Atomics International for environmental proof testing. Some of these have been highly complex, incorporating multi-capsule assemblies, NaK heat transfer, and individual gas annulus temperature control for each of eight sample pairs.

- D. A fully instrumented capsule assembly has been irradiated successfully for NASA; the assembly, which contains prototype fuel plates, is unique in that it is movable during irradiation, and may be partially discharged and recharged for parametric evaluations or single element failures.
- E. An assembly which consisted of fuel pins surrounded by a graphite shell was designed and irradiated for the MGCR program.
- F. The large core volume of our reactors provides capability of exposure of many capsules simultaneously to uniform temperatures and fluxes. Prototype fuel capsules have been positioned and rotated during irradiation, thus providing variable or uniform temperature and flux conditions. And up to 20 instrumented and 130 non-instrumented fuel pins have been irradiated at one time and under nearly identical conditions.

REACTOR CHARACTERISTICS AND EXPERIMENTS

Although Hanford is not "in the business," and therefore performs only those irradiations which cannot be reasonably performed in commercial test reactors, there is little problem with either space availability or reactor scheduling for those irradiations that do qualify. There are over 50 individual facilities, of four or five different types; many of them have the capacity for charge-discharge during operation. And, as previously mentioned, the relatively large reactor cores provide the opportunities for uniformity of irradiation in many simultaneously run samples.

The capability to determine reactor flux distributions has been well developed at Hanford; this can be accomplished fairly accurately even during operation. This feature is, of course, very attractive in parametric measurements or in burn-up experiments in which accurate nvt exposure history is desirable. Flux mapping may be done practically at will.

The normal neutron spectrum in the reactor core and test facilities is highly thermalized, but fluxes "tailored to measure" can be supplied. These tailored fluxes could conceivably run from 95% thermal to 95% fast, if required flux magnitude is not too high. Collimated neutron beam facilities have been designed and operated.

TEST ANALYSIS

Being something like "senior citizens" in the irradiations game, we have developed a good body of operating and technical experience over the years. Facility fluxes are known to fairly close detail; this, coupled with the capability of determining reactor flux distributions (and relating the two), allows good control of test exposure. Similar experience in heat generation rate, pressure drop, reactivity equivalent, and other calculations, supported by subsequent verifying irradiation (and associated empirical data) is quite valuable in test analysis. And we have a rather complete set of monitoring equipment, using both ionization chamber and calorimetric techniques, for post-irradiation measurements.

CO-EXISTENCE

I promised to shed some light on a problem--that of co-existence of experimental and production programs. Well, I'm pretty sure that I didn't really dazzle anyone with that light, primarily because of well-known security regulations. However, I can and will say this: we exist quite nicely with production. Our facilities are in general independent of reactor scheduling and loading, to the extent that experiments cannot be displaced or even greatly influenced by normal reactor operation. Reactor charging procedures are such that the flux local to any given facility is fairly constant over any long period of time. And (again due to the large core volumes) our test irradiations may be moved practically at will, even during operation, without significantly affecting reactivity or flux distribution.

SUMMARY

In summary, we have conducted a large volume of test irradiations over the years; many of them have been highly complex. And this has been accomplished for the most part with virtually no interference with normal manufacturing activities. Thus, by-product Hanford neutrons have found their ways into many widely diversified research and development programs, from Pittsburgh to Puget Sound, and from California to ~~Canaveral~~.

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IRRADIATION FACILITIES IN THE
OAK RIDGE NATIONAL LABORATORY
BULK SHIELDING REACTOR

C. E. Klabunde and B. C. Kelley

SOLID STATE DIVISION
OAK RIDGE NATIONAL LABORATORY
Operated by
UNION CARBIDE CORPORATION
for the
U. S. Atomic Energy Commission
Oak Ridge, Tennessee

IRRADIATION FACILITIES IN THE
OAK RIDGE NATIONAL LABORATORY
BULK SHIELDING REACTOR*

C. E. Klabunde and B. C. Kelley

The core of the ORNL Bulk Shielding Reactor (BSR)¹ consists of an assembly of MTR-type fuel elements. It is suspended in a pool of water, 40 ft long, 20 ft wide, and 22 ft deep, which acts as the reactor coolant, moderator, shield, and reflector. Steel rails are located along the east and west walls of the pool to support the movable reactor bridge. An additional movable bridge called the instrument bridge is used for a working platform and to provide space for special equipment. Presently, the reactor is limited to a maximum power level of 1 Mw. However, construction is under way to increase the maximum power level to 2 Mw and provide for 24-hr-a-day operation. All of the experimental facilities described below will be compatible with the increased power level.

In this paper, five unique irradiation facilities[†] for materials testing and solid state research are described. These are:

- 1) The Thermal Neutron Irradiation Facility
- 2) The Liquid Helium Cryostat
- 3) The Position Five Facility
- 4) The Liquid-Nitrogen Cooled Cryostat
- 5) The Fast-Flux Irradiation Facility

* Research sponsored by the U. S. Atomic Energy Commission under contract with Union Carbide Corporation.

[†] The persons responsible for the design, construction, and operation of the facilities are indicated in footnotes in the sections to follow.

1. THERMAL NEUTRON IRRADIATION FACILITY*

The express need for a high-dose thermal neutron irradiation facility has long been recognized. There is no locale in the Oak Ridge Research Reactor or the Low Intensity Test Reactor wherein the cadmium ratio[†] is in excess of about 300. The cadmium ratio in the biological thermal neutron treatment chamber of the ORNL Graphite Reactor was in excess of 10^5 , however, the actual thermal neutron dose rate was only about $10^9 \text{ cm}^{-2} \text{ sec}^{-1}$ in this facility. The cadmium ratio in the vertical graphite thimble experimental positions outside the heavy water region of the Argonne CP-5 reactor has been measured as being in excess of 10^4 and the actual thermal neutron dose rate is in excess of $10^{12} \text{ cm}^{-2} \text{ sec}^{-1}$ in these locales.

Design requirements were calculated² for a facility of high thermal neutron flux with relative absence of both fast neutrons and gamma rays for the BSR, and such a facility was constructed at the BSR.

Figure 1 shows a sketch of the Thermal Neutron Irradiation Facility as installed on the east side of the BSR. This facility consists of a 46 x 30 x 32 in. aluminum tank filled with heavy water. Vertical sample tubes (2 in. ID) are located 19, 29, and 40 in. from the reactor face. One of the three tubes (tube C_L, Fig. 1) at 29 in. from the reactor has an inside diameter of 2-3/4 in. and is shielded from gamma rays by a 14 x 8 x 8 in. thin-walled aluminum container filled with bismuth.

The thermal and fast neutron flux have been measured³ for each of the experimental facilities. Table I indicates the data as normalized to 1 Mw reactor power. Only five of the six available fuel elements were in the first row of the reactor at the time of these detailed measurements. Later measurements were made in just the two holes nearest the reactor (NW and SW), after the addition of the sixth fuel element and after repositioning the entire tank to leave about

* J. W. Cleland and J. T. Howe.

[†] The "cadmium ratio" is the ratio of total (thermal plus resonance) induced activity to the cadmium shielded (resonance only) activity in a detector like gold. This serves as an empirical indication of thermal flux "purity".

TABLE I

Fluxes in Thermal Neutron Irradiation Facility
at 1 Mw BSR Power Level

Detector - Sulfur Position	ϕ Fast (n/cm ² sec)	Gold ϕ Thermal (n/cm ² sec)	Copper ϕ Thermal (n/cm ² sec)	Ratio Cu/Au	Gold Cd Ratio	γ (r/hr)
E	.251 x 10 ⁷	.129 x 10 ¹²	.152 x 10 ¹²	1.182	593	.754 x 10 ⁵
C _L	4.81	.488	.580	1.188	511	1.41
N	3.76	.408	.485	1.189	926	3.89
S	2.96	.372	.440	1.183	1100	3.41
NW	58.9	1.50	1.78	1.182	116	14.4
SW	75.2	1.33	1.62	1.214	136	12.9

1/4 in. water gap between the tank and the reactor. These show a 60% gain in thermal flux (which should hold for all holes), and an increase in the cadmium ratio for these two holes from 120 to 200. It is anticipated that the total flux will double when the reactor power is increased from one to two megawatts in the near future.

The gamma flux has also been measured⁴ for each of the sample tubes, and is indicated in the last column of Table I. The relatively large gamma flux inside the bismuth shield (tube C_L) is primarily due to capture gamma rays from the aluminum sample tube. Preliminary measurements⁵ taken inside a bismuth container within this tube indicate a gamma flux of less than 10⁴ r/hr.

2. THE LIQUID HELIUM CRYOSTAT*

Liquid helium temperatures are necessary to immobilize crystal lattice imperfections (damage) induced in pure metals by reactor irradiation. Reactor fluxes induce damage in metals not only by fast neutron collisions,

* R. R. Coltman, Jr., C. E. Klabunde and G. F. Fielder.

but also by recoils from the (n,γ) -reaction upon capture of thermal neutrons.⁶ Since the latter mechanism produces a relatively simple damage configuration, it can be a valuable basic research tool if properly isolated from fast neutron damage. The liquid helium cryostat irradiation facility described here was developed to meet these needs.

The sample chamber is a one-inch diameter, eight-inch long section at the bottom of an 18-foot tube. The chamber walls are cooled by liquid helium circulating in a closed-cycle refrigerator system. The sample chamber and associated refrigeration transfer lines and vacuum jacketing are located in a six-inch diameter housing pipe which is surrounded by a heavy-water filled tank at the level of the reactor core (see Fig. 2). This aluminum tank provides about 20 inches of heavy-water moderator between the reactor face and the sample chamber. Resulting thermal flux is 8×10^{11} n/cm² sec at full 1 Mw reactor power, while fast flux (per sulfur) is a factor of 2000 lower and the cadmium ratio for gold is 100. Gamma-ray heating is about 10^{-2} watts per gram.

Figure 3 shows the refrigerator schematically. In the main heat exchanger high pressure helium gas is cooled from room temperature (by returning flows) on its way to be expanded adiabatically in the expansion engines. Then before the gas returns to room temperature through the main exchanger it absorbs in the heat-sink exchanger the heat load from the high pressure liquefier flow which is enroute to the Joule-Thomson (J.T.) valve. This J.T. flow is further cooled (by returning liquefier flow) in the J.T. heat exchanger. After isenthalpic expansion at the J.T. valve the gas-liquid mixture proceeds through the cryostat, where part of the liquid is evaporated; but little, if any, temperature rise occurs. The returning flow is progressively warmed to room temperature in the three heat exchanger sections enroute to the liquefier suction compressor from which it is returned into the main compressor bank.

The refrigeration capacities of both engine and liquefier portions are quite adjustable by varying mass flows which are governed by low and high pressure settings, compressor series or parallel operation, and J.T. valve setting. In a test with an electrical heater simulating the effect

of the cryostat, a 20 watt load was absorbed by a 20 scfm flow in the liquefier circuit at a stable temperature of 3.6°K. Engine-circuit flow in the test was 180 scfm. From the general behavior of the entire circuit under these test conditions, it was clear that a heat load of about 40 watts could be absorbed at less than 5°K.

Since the radiant heat load is a large fraction of the total in this system, use of a refrigerated shield surrounding all liquefier circuitry is essential. A small flow (1-2 scfm) of high pressure $\sim 16^\circ\text{K}$ helium gas taken from the engine inlet line is routed through a small tube down to the sample chamber of the cryostat, where the line cools a heat trap and shield for the sample chamber (see Fig. 4). It then returns along a shield surrounding the liquid helium transfer lines and its own supply line. Flow rate is adjusted to maintain the warm end of the shield at 125°K after having covered all liquefier circuitry. It is estimated that shielding reduces the liquefier radiant heat load by a factor of 100. Additional refrigerated shielding of part of the main heat exchanger is obtained from the remaining heat capacity of the shield gas flow.

All cold portions of the system are vacuum jacketed, such that when cold, the vacuum is better than 10^{-6} torr. In addition, "super insulation" in the form of four to six alternate layers each of aluminum foil and .015" fiberglass sheets have been wrapped on all cold surfaces. In the irradiation zone only a single layer of aluminum is used since gamma-heating makes "floating" shields useless.

Liquid nitrogen pre-cooling is used to cool the large mass of the main heat exchanger and the engines so as to reduce both start-up and total running time. With an overnight pre-cool and four hours of running, the system is ready for normal liquefier operation. Reliability of the over-all system is such that runs of 3-weeks duration have been made without difficulty.

The U-shaped arrangement of the cryostat (Fig. 4) reduces the diameter of the outside vacuum jacket so that as small converter* as possible may be used (in the future) for fission-neutron bombardment. Above the 1 in.

* A ^{235}U containing sleeve which effectively converts thermal flux to fission flux.

diameter by 8-1/2 in. long thin-wall copper sample chamber itself are portions of the sample tube serving as: 1) heat barrier, 2) refrigerated trap, 3) transition region up to room temperature, and 4)unjacketed extension to top of pool. In operation the liquid-flow-cooled wall of the chamber allows helium to be condensed inside the chamber, so that the experiment is immersed in static non-boiling liquid helium.

A typical sample can assembly for study of electrical resistivity changes in wire specimens is shown in Fig. 5. Static liquid helium is condensed in the can so that during full power reactor irradiation a sample temperature of 3.6°K is maintained. After an irradiation, study of the recovery of the damage may involve extensive thermal cycling of the samples to successively higher temperatures with return to liquid conditions for resistivity measurements. The double wall (can plus chamber) separating samples from refrigerant stream permits uninterrupted refrigerator operation while rapidly and precisely varying the temperature of the can and its contents to as high as 100°C with the aid of the heater and controlled vacuum isolation between can and chamber.

Temperature Measurement

Four kinds of thermometry are used in these operations. Liquid helium vapor pressure (up to 5.2°K) may read slightly high when there is a heat load causing the top liquid surface to be warmer. Carbon resistors (1/2 or 1/10 W, nominal 1000 Ω) give instant response and high sensitivity up to about 50°K. They suffer slight radiation damage and are moderately reproducible under thermal cycling. Copper-constantan thermocouples are good above about 30°K and suffer negligible radiation damage even at far greater doses than available here. Helium gas-bulb thermometry (in a special two-volume system) can be very sensitive, accurate and reproducible within 0.1% of the absolute temperature up to about 40°K; it is radiation damage-proof.

A combination of all of these methods is usually required to cover the range of temperatures needed in our experiments and to provide both speed of response and ultimate reproducibility. An innovation was necessary to adapt the gas-bulb thermometer for use where some 30-35 feet

separate the cold bulb and the sensing meter. A capillary this long, fine enough to be of negligible volume, would give nearly zero response rate. Only two feet of capillary is necessary for the transition to room temperature, then 1/8 in. copper tubing (water jacketed for temperature stability) leads to a bulb and the pressure meter. The total warm volume is purposely made 20 times the cold-bulb volume, giving a nonlinear (but calculable) response which is very steep at the low end. It does require 3 to 4 minutes to come to equilibrium after a large temperature change, therefore a carbon resistor thermometer is used to arrive at and maintain an annealing temperature, which is accurately determined by the gas thermometer after 4 minutes.

Flux Monitoring

Auxiliary access tubes into both the rear D₂O tank and the 6 in. housing pipe are used for various flux monitoring schemes. A boron thermopile inserted in one of the tank-access tubes monitors thermal neutron flux above 0.1 Mw. In some experiments very accurate matching of doses from one run to another has been monitored by activating and counting fine, pre-weighed pieces of copper wire irradiated just outside the cryostat.

3. THE POSITION FIVE FACILITY^{*}

The Position Five Facility (PFF)⁷ is installed in the BSR to provide a chamber for in-pile radiation damage experiments or for post-irradiation experiments following reactor spectrum neutron irradiations. As shown in Fig. 6, the facility is located at the northwest corner of the reactor. It consists of a curved access tube, 4 in. in outer diameter, which is welded to a hollow core piece having the same outer dimensions as a BSR fuel element, roughly a 3 in. square cylinder 2 ft long. The top of the tube is sealed with a plug (Fig. 7) which has incorporated in it a port for pressurizing the system and several electrical feedthroughs. There is a neutron and gamma shield plug suspended from the seal plug. A side arm is also provided for evacuating the system. A lead ballast, sheathed

^{*} J. M. Williams, B. C. Kelley, W. E. Brundage and M. S. Wechsler.

in aluminum, surrounds the tube near the bottom end to compensate for buoyancy. As can be seen in Fig. 8, the top of the tube extends above the surface of the water. A large transfer cask has been constructed that may be lowered around the top of the tube and into which the irradiated assemblies may be removed.

The gamma heating in the facility was measured by placing an aluminum block containing wells for an internal heater and thermocouples in the tube. The block was prevented from making intimate contact with the inside wall of the facility by set-off rings of Nichrome wire. The total mass of the assembly approximated that of a typical experiment. With the heater power off, the temperature of the block was measured as a function of reactor power. At full power, the temperature reached 150°C. Then with the reactor off, the electrical power to the internal heater necessary to raise the temperature to 150°C was determined. In this way, the gamma heating at full power was found to be about 0.1 w/g.

As can be seen in Fig. 6, one of the three shim safety rods is located diagonally adjacent to the PFF. The question arose as to the sensitivity of the neutron flux to the shim rod position. Therefore, a survey was made of the thermal flux (using cobalt wires) and fast flux above 2.9 Mev (using nickel wires) as a function of vertical distance in the sample tube for several positions of this shim rod. The reactor power, as measured by a servo chamber on the south side of the core, was maintained at 0.1 Mw by adjusting the other control rods and the results scaled up to 1 Mw. The flux profiles shown in Fig. 9 indicate a change of only a few percent for shim rod positions of 16 and 22 inches withdrawn. However, if the No. 1 shim rod is dropped to a 10 in. withdrawn position, the flux may be decreased by as much as 10% along the upper end. During normal operations at 1 Mw the usual position of this rod is between 16 and 22 inches withdrawn. Figure 9 also indicates that in the region of the flux peak, the flux is uniform to within 10% over a span of 7 or 8 inches.

4. THE LIQUID -NITROGEN-COOLED CRYOSTAT*

Several experiments planned for the BSR require that the irradiation temperature be maintained well below room temperature. A cryostat has been constructed for this purpose, which is cooled by helium gas pre-cooled by heat exchange with liquid nitrogen. The liquid nitrogen does not enter the radiation zone.

The cryostat was constructed and mounted on the instrument bridge such that it may be moved to almost any spot in the reactor pool. Figure 10 shows a schematic layout of the cryostat. Helium gas pumped through a liquid nitrogen cooled counter-flow heat exchanger enters the bottom of the sample chamber and flows up, cooling the chamber. The gas then returns to the heat exchanger, flows through the inner annulus helping to cool the helium which is enroute to the chamber. Next the helium gas flows to the helium pump where the cycle begins again. The cold regions are insulated by a vacuum jacket surrounding the assembly.

In the configuration shown in Fig. 10, the samples were irradiated while in the helium flow. This arrangement somewhat complicates removal of samples while they are cold and requires a new helium charge each time a sample is removed. Since most irradiations will require cold removal of samples, a sample tube 1-1/8 in. outside diameter by 20 ft long is inserted in the 1-7/8 in. inside diameter chamber. Once a sample is inserted in the inner chamber, the chamber is evacuated and back-filled with helium gas for heat transfer.

When the irradiation is completed, the helium is pumped out and the chamber back-filled with nitrogen gas which is then liquefied, facilitating cold sample removal.

Since it was necessary to use an oil lubricated refrigeration compressor to pump the helium through the circuit, oil vapor problems were expected. To help alleviate this somewhat, oil separators and fume filters were installed in the helium circuit. In addition to this, connections

* J. T. Howe, W. E. Busby, J. T. Stanley, B. C. Kelley and W. E. Brundage.

and valving have been provided to periodically flush the system with a suitable solvent.

Flux determinations have been made in this facility for thermal flux (using cobalt wires) and fast flux above 2.9 Mev (using nickel wires). These data are shown in Fig. 11.

5. FISSION SPECTRUM FACILITY*

Studies of radiation damage in metals and alloys require a knowledge of both the number and energies of the neutrons. The determination of the spectrum in an experimental facility is a difficult and inexact process. Also, the flux and spectrum are subject to change as a result of variations in control rod position, fuel, and experiment loading in the reactor. Therefore, it is helpful to provide an irradiation chamber with a reproducible, known neutron spectrum such as the fission spectrum. Once the neutron spectrum and spacial distribution are determined, the required neutron flux measurements are limited to a single properly chosen monitor irradiated along with the sample. In the fast flux irradiation facility, the nearly-pure fission spectrum is provided by irradiating a neutron converter (a ^{235}U -bearing sleeve surrounding the sample chamber) in a high-purity thermal flux. Another motivation for providing a fast neutron flux environment, without moderated or thermal components, is in connection with the AEC's fast breeder reactor program for which information is needed on radiation damage effects to structural materials in fast neutron fluxes.

The general arrangement of the facility is shown in Fig. 12. The 12-in. thick D_2O moderating tank is mounted on the reactor grid plate and moves with the reactor to the converter which is fixed in the pool. The main housing of the converter is also filled with D_2O except the 14-in. diameter cylinder in the center which is void except for the cooling fins. These fins function as an emergency heat path in the event of coolant failure. Details of the converter element, located in the center of the box are illustrated in Fig. 13. It is fabricated with a thick

* W. E. Brundage and M. S. Wechsler.

aluminum cladding into which a series of longitudinal slots 0.038 in.-sq are machined to form the coolant passages. The fuel is Uranium enriched to 93% ^{235}U and alloyed with aluminum and is also clad on the inner surface with aluminum. The innermost portion of the assembly is a cylinder of cadmium-magnesium alloy which provides additional thermal neutron shielding for the sample. The converter element is at the lower end of a tube which extends to the surface of the pool so that samples may be lowered directly into the center of the converter. This tube will be closed with an atmospheric and shield plug except during loading and unloading. A shielding cask which was fabricated for use with the Position Five Facility will also serve the converter to aid in the removal of experimental rigs.

The anticipated flux within the converter space is a minimum of 10^{12} n/cm² sec. This will result in a fission heat level in the converter of 30 kw. Previous measurements by Coltman et al.⁸ for a similar geometry at the BSR indicate that a flux of 2×10^{12} n/cm² sec may be expected when the reactor is operated at a 2 Mw power level. The converter element and cooling system are being designed for operation at a reactor power level of 2 Mw.

The Fast Flux Irradiation Facility is presently in the detailed design stage. A criticality test on a mock-up of the facility has been carried out,⁹ and no undue hazard was found to exist.

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Figure 1. Bulk Shielding Reactor Thermal Neutron Irradiation Facility.

Figure 2. The Cryostat and Reactor Arrangement of the Liquid Helium Irradiation Facility at the BSR.

Figure 3. Schematic Flow Diagram of Liquid Helium Facility.

Fig. 4

Figure 4. Liquid Helium Irradiation Cryostat.

Fig. 5

Cadmium Sample Rig

Figure 5. Liquid Helium Specimen Irradiation Assembly.

Figure 6. Core Loading of the Bulk Shielding Reactor.

Figure 7. Schematic Drawing of the PFF.

Figure 8. Photograph of BSR Pool Showing Core and PFF.

Figure 9. Neutron Flux Profiles in BSR-PFF for Various Positions of No. 1 Shim Rod.

Figure 10. BSR Liquid Nitrogen Cryostat.

Figure 11. Flux Data from Liquid Nitrogen Cooled Cryostat.

Fig. 12

Figure 12. General Arrangement of the BSF Fission Spectrum Converter and the BSR.

Fig. 13

Figure 13. Converter Element of the BSF Fission Spectrum Converter.

IX. APPENDICES

APPENDIX "A"

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