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Westinghouse Advanced Reactors Division

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URANIUM PLUTONIUM CARBIDE DEVELOPMENT

QUARTERLY PROGRESS REPORT

FOR THE PERIOD

ENDING DECEMBER 31, 1967

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LIST OF PREVIOUS PUBLICATIONS

The following is a list of previous publications prepared for the U. S. Atomic Energy Commission under AEC Contract AT(30-1)-3791.

WCAP 3791-1	Uranium-Plutonium Carbide Development Work Program for the Period October 3, 1966 to June 30, 1967.
WCAP 3791-2	Uranium-Plutonium Carbide Development Quarterly Progress Report for the Period Ending December 31, 1966.
WCAP 3791-3	Compatibility of Stabilized Hypostoichiometric Uranium-Plutonium Monocarbide Fuels with Stainless Steel.
WCAP 3791-4	Uranium-Plutonium Carbide Development Quarterly Progress Report for the Period Ending March 31, 1967.
WCAP 3791-5	Uranium-Plutonium Carbide Development Quarterly Progress Report for the Period Ending June 30, 1967.
WCAP 3791-6	Uranium-Plutonium Carbide Development Irradiation Test Program.
WCAP 3791-7	Vanadium Cladding Alloy Development Work Program for the Period July 1, 1967 to June 30, 1968.
WCAP 3791-8	Uranium-Plutonium Carbide Development Work Program for the Period July 1, 1967 to June 30, 1968
WCAP 3791-9	Vanadium Cladding Alloy Development Quarterly Progress Report for the Period Ending September 30, 1967.
WCAP 3791-10	Uranium-Plutonium Carbide Development Quarterly Progress Report for the Period Ending September 30, 1967.
WCAP 3791-11	The Analysis of Mixed Carbide Reactors for the AEC LMFBR Task Force Studies Final Report
WARD-3791-13	Vanadium Cladding Alloy Development Quarterly Progress Report for the Period Ending December 31, 1967.

SECTION 1

INTRODUCTION

This is the fifth quarterly progress report and covers the work performed during the second quarter of FY-1968.

1. PROJECT OBJECTIVE

The overall project objective is to develop a basic fuel element design that is suitable for application in a commercial, liquid metal-cooled, fast breeder reactor (LMFBR). The investigations will encompass both vented and sealed element designs, because vented elements are considered preferable for long-endurance cores, and sealed elements may have to be used initially as an expedient to avoid reactor system design difficulties. The fuel used in both types of elements is a stabilized, uranium-plutonium monocarbide (U,Pu)C solid solution, though microstructures will also include small amounts of second phases required for stabilization. For heat transfer reasons, the (U,Pu)C will be sodium-bonded to the fuel element claddings.

The project is expected to culminate with definition of processes for manufacture, for quality control, and for final product characterization of a (U,Pu)C fuel element that is suitable for commercial LMFBR use. Also, by that time, the performance characteristics of such elements will have been sufficiently established to support detailed engineering design of entire cores for commercial, liquid metal-cooled, fast breeder reactors.

2. PRIOR WORK

All work related to the present investigations and performed prior to the present period has been fully reported in:

WCAP-3791-2, "Uranium Plutonium Carbide Development Quarterly Progress Report for the Period Ending December 31, 1966,"

WCAP-3791-4, "Uranium Plutonium Carbide Quarterly Progress Report for the Period Ending March 31, 1967,"

WCAP-3791-8, "Uranium Plutonium Carbide Quarterly Progress Report for the Period Ending June 30, 1967," and

WCAP-3791-10, "Uranium Plutonium Carbide Quarterly Progress Report for the Period Ending September 30, 1967."

3. SUMMARY OF CURRENT PROGRESS

A summary of current progress is shown below. Detailed progress is reported in the individual task sections of this document.

Under UPC-100, estimated expenditures for the period January through June, 1968 were prepared and submitted in December. A contract modification providing interim FY-1968 funding was received and accepted during the period.

Under UPC-200, the production of fuel pellets for GETR Phase I irradiation test capsules were initiated and continued throughout the period. Process and equipment problems have delayed the completion of the three capsules; however, sufficient unmodified and chromium-modified pellets for the first of the three capsules and most of the pellets for the other two capsules were manufactured. A limited investigation on process variables was conducted during the period.

Under UPC-300, the three pins for the first GETR irradiation test capsule were fabricated. In addition, work progressed on EBR-II irradiation specimen fabrication. Two fuel pin specimens and five vanadium containers were bonded and evaluated, and modifications to the sodium glovebox for the handling of the 40-inch long capsule were completed. A program was started and completed during the period to provide data to indicate an optimum process for bonding of GETR irradiation pins. Because sufficient space was not available in the glovebox used for sodium bonding development, test apparatus for sodium deposition studies were modified to make the apparatus self-contained.

Under UPC-400, the three Phase I fuel pins, the pre-assembly machining of the capsules, and the WARD review of the data package were completed. The instrumentation console was shipped to GETR. Design analysis of the Phase I capsule resulted in a change in the pellet hold-down device. This change was required to circumvent difficulties encountered during sodium bonding. Design analyses were initiated for Phase II capsules. The purchase orders for irradiation services, special services, and V-RAFTS were approved by the AEC. The fabrication of V-RAFTS and shipping containers were completed.

Under UPC-500, the design of the eleven Phase I capsules was completed and fabrication development begun. Preparation of the design and hazards report for experiment approval is in progress.

Under UPC-600, the TREAT experimental program is being developed. Capsule configuration studies, nuclear analysis, and thermal analysis continued during the period.

SECTION 2

UPC-100 - PROJECT ADMINISTRATION

W. B. Heubel, A. E. Duval

1. OBJECTIVE

The objectives of this task are as follows:

1. To assure timely completion of the full scope of the project within the financial plan
2. To provide day-to-day project planning, coordination, and control
3. To satisfy USAEC contract administration requirements
4. To coordinate this project with other USAEC-sponsored Westinghouse projects in the Commission's Liquid Metal-Cooled Fast Breeder Reactor (LMFBR) Development Program.

2. PRIOR WORK

Prior work in project administration included the following:

1. Finalization of the contract.
2. Establishment of project controls.
3. Preparation and submittal of FY-1967 and FY-1968 Work Programs, (WCAP-3791-1 and WCAP-3791-5).
4. Preparation and submittal of Quarterly Reports, (WCAP-3791-2, WCAP-3791-4, and WCAP-3791-8).
5. Preparation and submittal of Topical Report, WCAP-3791-3, "Compatibility of Stabilized Hypostoichiometric Uranium-Plutonium Monocarbide Fuels with Stainless Steel."
6. Preparation and submittal of (U,Pu)C Development Irradiation Test Program, (WCAP-3791-6).

7. Initiation and submittal of monthly letter progress reports.
8. Initiation and submittal of monthly reports of actual costs.
9. Application for and receipt of license to operate the plutonium fuels development laboratory.
10. Order for and receipt of initial requirements of plutonium metal and natural uranium.
11. Negotiation of thermal irradiation services at GETR.
12. Negotiation and placement of subcontract with GETR for design and fabrication of three irradiation capsules for irradiation of fuel pins.
13. Expansion of Task UPC-300 to include Fuel Column Expansion Evaluation and Sodium Corrosion Behavior of Fuel Element Cladding. This work is incorporated into the approved work program for FY-1968, (WCAP-3791-5). Cost and price analysis, Form NY-272, was submitted for this scope.
14. Receipt and acceptance of Contract Modifications 1, 2, and 3.
15. Request and receipt of Commission approval to insert vanadium alloy samples in Phase I, Fast Flux Irradiation Carbide Capsules (UPC-500).

3. CURRENT PROGRESS

The principal administrative accomplishments during the calendar quarter September 1, 1967 to December 31, 1967 were as follows:

1. Contract Modification 4 was received and accepted.
2. The Quarterly Progress Report for the period July 1 to September 30 was prepared and submitted, (WCAP-3791-10). Monthly progress reports for September, October, and November were prepared and submitted.
3. Revisions to the FY-1968 monthly forecast of expenditures were prepared and submitted to the Commission.

SECTION 3

UPC-200 - FUEL MATERIAL FABRICATION DEVELOPMENT

W. R. Jacoby, M. G. Bale, P. M. French, M. H. Hodge, R. M. Horgos
T. W. Wallace, N. G. Wolcott, Jr.

1. OBJECTIVES

1. To develop suitable (U,Pu)C fuel materials, stabilized either by the addition of iron or chromium carbide (Cr_{23}C_6) to the hypostoichiometric carbide, or by addition of chromium to the hyperstoichiometric carbide. Such fuel must be compatible with cladding materials in a sodium-bonded fuel element.
2. To develop process techniques for the fabrication of prototypical fuel pellets that will be inserted in fuel test assemblies. The test assemblies will be utilized to evaluate the performance of these stabilized fuel materials under conditions similar to anticipated liquid metal-cooled, fast breeder reactor conditions.

2. PRIOR WORK

2.1 Contract Work

The initial work on the fabrication of GETR Phase I fuel pellets commenced in the second quarter of 1967 after the fabrication line gloveboxes in the Plutonium Laboratory had been activated. Prior to this, procedures for arc melting, crushing, cold pressing and sintering the material had been prepared. During the early work, it was not possible to obtain the characterization data required since the evaluation glovebox line had not been activated. It was not until the later part of the third quarter before chemical analyses were available; it was then determined that the carbides worked on had extremely high oxygen contents (up to 5,000 ppm). To reduce oxygen pick-up rates, argon-purged tents were constructed and installed in the gloveboxes, and alcohol with trichlorethylene was used as a solvent.

The final leak-checking of the evaluation line was completed, and shakedown trials on the metallographic equipment undertaken. Plutonium-containing carbide samples were first accepted during September and conventional metallographic techniques were applied in preparing these for phase identification.

Stainless steel immersion density standards were fabricated, calibrated, and used to standardize density measurement utilizing ethylene dibromide. Immersion densities were undertaken on approximately 50 UC and (U,Pu)C pellets using this dibromide. As anticipated, densities thus obtained were 2-3 percent higher than those determined geometrically. The pellets were found to have typical densities of approximately 90 percent theoretical, with 6 percent open porosity and 4 percent closed.

A limited amount of wet sieving was undertaken on (U,Pu)C powders after the milling operation for both UC and (U,Pu)C powders. Depending on the particular batch, it was found that:

1. 30-40 percent by weight passed through the 10 μ sieve.
2. 40-45 percent passed through the 20 μ sieve.
3. The remainder (~30 percent) was retained by the 20 μ sieve.

Oxygen, nitrogen, and carbon analyses were not available until early September since a variety of problems were encountered with the analytical chemistry equipment. These problems included leaks in the Nitrox analyzer, insufficient water cooling in the high-frequency coil, and "shorting" from the leads onto the box shell.

For use in U-Pu determinations, X-ray fluorescence standards having a wide range of U-Pu ratios were prepared.

2.2 Related Work Not Charged to the Contract

A Fisher Sub-Sieve Analyzer was received and modified for glovebox installation. Torque-versus-pressure calibration curves were obtained for the analyzer loading mechanism, and the average particle size of a variety of ZrO₂ samples was determined.

3. CURRENT PROGRESS

Work performed during this reporting period is reported in two parts: Contract Work, and Related Work Not Charged to the Contract.

3.1 Contract Work

During the first weeks of the quarter, it was decided that the use of trichloroethylene, or any halogen-bearing gas, would have to be discontinued since the decomposition of the halogenated vapors resulted in the formation of acid fumes which affected the process atmosphere purification system. Therefore, alcohol was returned to use as the primary solvent.

During the quarter, work in carbide pellet production has progressed in three main areas:

1. Reduction of the oxygen contents of the powder and pellets.
2. Production of GETR Phase I irradiation specimens.
3. General investigations on the effects of process variables (e.g., milling time, particle size, pressing procedure, and firing cycle) and on fuel evaluation.

The following discussion describes this work in detail.

3.1.1 Reduction of the Oxygen Contents

A decrease in the powder oxygen content was noted while using the argon-purged tent and it was decided to minimize the working volume even further by using small containment vessels. Plastic containers, large enough to take all necessary equipment, were obtained and these were purged with argon of <100 ppm oxygen. An appreciable decrease in the oxygen contents of powder has resulted, but unprotected powder was found to oxidize rapidly. In one test, samples of carbide were pressed without binder addition and the oxygen contents of the pellets were found to be extremely high. It is proposed that the binder (carbowax) affords some protection for the powder through much of the processing. Typical oxygen contents at various process stages are given in Table 200-1.

TABLE 200-1
TYPICAL OXYGEN CONTENTS OF CARBIDE MATERIAL
AT VARIOUS PROCESS STAGES

Carbide Form	Typical Oxygen Contents (Ppm)	Minimize Oxygen Contents (Ppm)
Arc-Melted Button	100	15
Carbide Powder Prior to Binder Addition	400-800	100
Pellets Taken Within 24 Hours of Firing	1,400-2,000	1,000
Pellets Taken 2-4 Weeks After Firing	3,000-5,000	

Further work is required to determine the source of oxygen contamination between the powder and pellet stages.

3.1.2 GETR Fuel Pellet Fabrication

Delays have occurred in the completion of the three GETR test irradiation capsules due to process and equipment problems. However, sufficient unmodified and chromium carbide modified fuel and insulator pellets have been manufactured to satisfy requirements for the first test capsule which will contain three fuel elements. In addition, most of the pellets required for the other two capsules have been made.

Additional pellets for backup work on sodium purity analysis, library samples, compatibility samples, etc., have also been provided. One-hundred-percent dimensional checking of all these pellets ensured that material met acceptable specification before final loading.

3.1.3 General Investigations on Process Variables and on Fuel Evaluation

During the course of irradiation test specimen fabrication, it has been possible to undertake very limited research on process variables.

Arc-melted buttons were prepared by a standard procedure and process variations were made after this stage. Alterations in milling time and sieve sizes have been investigated and the results correlated with the Fisher sub-sieve grain sizes (see Table 200-2). Finer particles may be obtained by sieving the powder through finer screens or by milling for longer times. Regarding the fired pellets, with decreasing mean particle size an increase in sintering rate was noted.

Results from one recent batch of powder indicate that the fired pellet densities are not greatly affected by applied pressures ranging from 20 ton/inch² to 60 tons/inch² nor by sintering times in excess of one hour when fired at 1550°C. Decreasing densities have been noted with increase in carbon and chromium carbide additions -- the reverse is believed true of iron additions.

During this reporting period, considerable difficulty was experienced in obtaining a good, lasting polish on UC and (U,Pu)C metallographic samples, even

TABLE 200-2

THE EFFECT OF MILLING TIME AND SIEVE SIZE USED ON THE AVERAGE PARTICLE SIZE
AND RESULTANT PELLET DENSITIES FOR UC AND (U,Pu)C SAMPLES

Material	Milling Time (Hours)	Sieve Size Used (Mesh)	Fisher Reading (Microns)	Typical Fired Pellet Densities (% Theoretical)
UC	16.0	-325	1.9	88-90
(U,Pu)C	2.5	-325	2.7	87-87
(U,Pu)C	2.5	-200	3.7	86-88
(U,Pu)C	2.5	-200	3.7	89.5-90.5
UC	1.5	-325	3.8	88-90
(U,Pu)C	1.5	-325	3.8	86-88
(U,Pu)C	1.5	-200	3.7	Defective pellets

though all solvents were dried prior to use in the glovebox.

Analysis of the Metadi lubricating fluid used in the final polishing operation showed the fluid to have high water content (>500 ppm) despite assurances from the vendor to the contrary. Breakdown of solvents, moreover, has occurred in the argon purifier feeding the metallography box, causing a massive increase of impurities in the argon. Shutting down the metallographic operation for several days during December allowed the purifier to "clean" itself. The impurity content of the argon is now acceptable (150 ppm O_2 and < 5 ppm H_2O). However, if further shutdowns are to be avoided, a solvent vapor trap will have to be installed in the argon line between the box and the purifier.

The Metadi fluid used in the polishing operation has been replaced with a silicone oil-kerosene mixture which has proved successful. Following the atmosphere cleanup, both UC and (U,Pu)C samples have been successfully ground and polished to 1/4 μ diamond finish and the polish has lasted approximately 40 minutes before surface oxidation became noticeable.

A variety of samples, both cast and sintered, have been metallographically examined during this period. With some sintered samples, an "orange peel" effect has been observed after polishing. Gross pressing defects have been observed in a number of pellets, ground and polished, produced during the early part of this period. Figure 200-1, showing a UC sample, (4.69 percent C, 2800 ppm O_2 , 125 ppm N_2) illustrates the worst case observed; not only were both ends of the pellet capped, but pressure cracks were also generated through the sample. These were caused by a faulty pressing technique (resulting in uneven flow of powder). Since then, the pressing technique was improved and these types of defects were eliminated. Figure 200-2 showing a UC sample (4.44 percent C, 2283 ppm O_2 , 38 ppm N_2) is an example of this. (Note that this sample also exhibits the "orange peel" effect referred to above.)

Since the cathodic etcher had not been installed in the glovebox and some chemical etching development work has had to be undertaken to reveal microstructures of the carbide pellets, other means of etching had to be used. This was accomplished by using 85 percent orthophosphoric acid solution which

Figure 200-1. Photomicrograph (10.5X) of UC Pellet
Showing "Capping" and Other Pressing Defects

Figure 200-2. Photomicrograph (10.5X) of UC Pellet
Showing No Pressing Defects

proved satisfactory for etching Cr_{23}C_6 stabilized UC and (U,Pu)C.

Figure 200-3 shows the typical structure of one of the first Cr_{23}C_6 stabilized (U,Pu)C pellets produced in the WARD Plutonium Laboratory. This pellet has been etched for six seconds in 85 percent orthophosphoric acid. Gross segregation of the $\text{Cr}_{23}\text{C}_6 + \text{Cr}$ phase (white areas) has occurred in the (U,Pu)C matrix. Around the white "islands" are dark grey areas, which are (U,Pu)C produced by the stabilizing addition of Cr_{23}C_6 . Farther from these "islands" are light grey areas, which comprise the original (U,Pu)C matrix.

In recently produced Cr_{23}C_6 stabilized pellets, a more uniform distribution of the $\text{Cr}_{23}\text{C}_6 + \text{Cr}$ phase has been achieved in the grain boundary regions. This is illustrated in Figure 200-4 where the average (U,Pu)C matrix grain size is 25 microns. Porosity distribution for this type of material is satisfactory and pore size is uniform.

No major problems have been encountered with the analytical chemistry equipment during this reporting period, and oxygen, nitrogen, and carbon determinations have been successful on samples taken from arc-melted buttons, milled powders, and sintered pellets. There is now sufficient confidence in the oxygen and nitrogen analyses that duplicate samples are not routinely run. However, carbon determinations are still being duplicated.

U-Pu ratios in sintered pellets have been determined by an X-ray fluorescence technique. The U-Pu ratio in a variety of sintered batches ranged from 0.2439 to 0.2661, the desired ratio being 0.2500. Using the fluorescence technique, chromium determinations have also been made on Cr_{23}C_6 stabilized UC and (U,Pu)C pellets.

Investigations have been initiated on the effects of process variables on the densities of fired pellets of UC and (U,Pu)C. Variations in milling time and sieve size through which the powder feed is forced are correlated with Fisher Sub-Sieve readings listed in Table 200-2.

Thus, increasing the milling time from 1.5 to 2.5 hours and varying the sieve size from -200 to -325 mesh has apparently little effect on the average particle size, except in the one instance. There is no obvious correlation between final density and other variables listed.

Figure 200-3. Photomicrograph (300X) of CR_{23}C_6 Stabilized (U,Pu)C Pellet
(4.34%C, 1179 ppm O_2 , 140 ppm N_2 , 0.243 U-Pu)

Figure 200-4. Photomicrograph (300X) of CR_{23}C_6 Stabilized (U,Pu)C Pellet
(4.53%C, 1136 ppm O_2 90 ppm N_2 , 0.250 U-Pu)

A typical run for a (U,Pu)C powder sample tested in the Fisher Sub-Sieve is illustrated graphically in Figure 200-5 as a plot of average particle size versus compacting pressure and powder porosity. With this type of instrument, the "true" average particle size is that value where the curve of particle size versus pressure plateaus. Chemical analysis of the powder was 4.65 percent C, 1156 ppm O_2 , 592 ppm N_2 .

Investigations have also been undertaken to determine the O_2 pickup in the carbide samples while in the Fisher Sub-Sieve. This is illustrated in Table 200-3. Four runs were made on the same batch of (U,Pu)C, and oxygen samples taken immediately after particle size determination.

These figures indicate that a crust of oxide forms on each particle during the test, and that above 1700 ppm O_2 buildup this crust falls off, causing a reduction in the observed particle size. Furthermore, under these conditions, a realistic figure for the average particle size can be obtained from two runs in the Fisher without oxidation of the carbide affecting the result.

Because of solvent contamination of the argon purification units, only a limited quantity of immersion density determinations have been made and no wet sieving operations started. Also, because halogen ions were also detected in the argon atmosphere, it was decided to replace ethylene dibromide as the immersion liquid with dried xylene (this was standardized with the stainless steel cylinders, as previously described). Typical immersion densities of four (U,Pu)C pellets from a single sintering batch versus geometric densities are shown in Table 200-4.

3.2 Related Work Not Charged to the Contract

The vacuum cathodic etcher has been received from NUMEC and used outside the glovebox line for etchings on a variety of samples. One example of this is illustrated by Figure 200-6 which shows the microstructure of a Type 316 stainless steel sample. Grain boundaries and twinned areas have been well defined by this technique. The etcher is currently being installed in the glovebox line.

Figure 200-5. Average Particle Size versus Applied Pressure and Porosity for a (U,Pu)C Powder Sample Tested in a Fisher Sub-Sieve Sizer

TABLE 200-3

EFFECT OF OXYGEN PICK-UP DURING RUNS IN THE FISHER SUB-SIEVE
ON THE AVERAGE PARTICLE SIZE OF A (U,Pu)C SAMPLE

Average Particle Size of (U,Pu)C Powder Sample (Microns)	Oxygen Content of Powder After Particle Size Determination (Ppm)
2.7	1,655
2.7	1,700
2.3	2,109
2.1	3,061

Note: The original O_2 content of the powder was 500 ppm and bottle argon (400 ppm max impurity) was "fed" to the Fisher.

TABLE 200-4
DENSITY AND POROSITY OF FOUR PELLETS FROM
A BATCH OF SINTERED (U,Pu)C

Geometrical Density (% Theoretical)	Immersion Density (% Theoretical)	Open Porosity (%)	Closed Porosity (%)
90.5	91.8	3.8	4.4
90.4	91.7	3.1	5.2
93.0	92.3	1.9	5.8
91.6	92.2	2.5	5.3

Note: The analysis of this batch of pellets was 20.2% Pu, 4.65% C, 2,223 ppm O₂, and 382 ppm N₂.

Figure 200-6. Photomicrograph (Cathodically Etched, 250X)
of Type 316 Stainless Steel Sample

The water in the Fisher Sub-Sieve Analyzer was replaced with silicone oil (density 1.05 g/cc) before this apparatus was installed in the fabrication glovebox line.

The Fisher was then recalibrated in the glovebox argon atmosphere and used to determine the average particle size of zirconia, chromium carbide, and iron samples. The chromium carbide and iron were taken from the batches and held to be added as stabilizers to carbide powders. Within the detection limits, no difference was observed between in-box and out-of-box measurements of average particle size of the batch of zirconia.

Typical average particle size of the chromium carbide powder was found to be 8.7 microns and that of the iron powder 5.3 microns. Both these figures are considerably higher than those found for UC and (U,Pu)C powders examined to date, and as reported above.

3.2.1 Compatibility Capsule Design

The engineers involved in this task of capsule design and development are P. M. French, G. A. Whitlow, J. C. Cwynar, and R. C. DeKlever.

The proposed compatibility capsule design is illustrated in Figure 200-7. Five different cladding materials are each to be tested in intimate contact with five UC and five (U,Pu)C fuel compositions, and one (U,Pu)O₂ composition. To avoid any mass transfer complications in the sodium environment, the inner liner, cladding discs, extension rod, and hold-down rod will all be of a single material in any single capsule. In certain instances, nine pellets will be loaded into a capsule; so that the same dimensions can be used for all inner liners, housings, and outer jackets, an extension rod will be inserted in each capsule when only six pellets are required for the capsule. A conventional method of maintaining contact between pellet-disc couples has been adopted. This is accomplished by using Inconel 600 as the liner housing material, and relying on its differential thermal expansion relative to those of the various potential cladding materials in the liner assemblies.

It is also proposed to heat treat the capsules in furnaces outside the glovebox area. As a safeguard, during the heat treatment period, the capsule will be contained in a heavy outer stainless jacket with a welded end closure.

Figure 200-7. Cladding Material-Carbide Fuel Compatibility Capsule

SECTION 4

UPC-300 - FUEL ELEMENT FABRICATION DEVELOPMENT

W. R. Jacoby, W. B. Bargerstock, W. C. Bowen, J. C. Cwynar,
R. C. DeKlever, R. K. McGeary

1. OBJECTIVES

The objective of this task is to develop techniques for fabrication and characterizing sodium-bonded and helium-bonded, mixed carbide fuel elements, both vented and nonvented, that are suitable for use in commercial, sodium-cooled, fast breeder reactors.

2. PRIOR WORK

Procedures were developed and documented for GETR irradiation pin fabrication. These include end plug welding, sodium loading, pellet loading, sodium bonding heat treatment, and characterization test methods. It was indicated that additional refinement of the eddy-current test and extensive correlation with destructive examination was required.

3. CURRENT PROGRESS

3.1 Contract Work

3.1.1 GETR Irradiation Pin Fabrication

Irradiation pins are being made using the process flow diagram illustrated in Figure 300-1. Detailed specifications have been written for each operation and test. Three pins have been completed and none have been lost or recycled during fabrication.

Figure 300-1. Flow Diagram for GETR Irradiation Pin Fabrication

3.1.2 EBR-II Irradiation Specimen Fabrication

EBR-II irradiation specimen process development involves three main hardware components: 1) 28-inch long simulated fuel pin fabrication, 2) vanadium container fabrication and sodium filling, and 3) 40-inch long capsule fabrication and sodium bonding of fuel pin and vanadium container within the capsule. Progress in these tasks is as follows:

1. To date two fuel pin specimens have been bonded and completely evaluated. Full length (12-inch) dummy pellet stacks were used. Eddy-current test traces indicated extreme lateral staggering of pellets, but only one significant void. Peeling of the cladding so as to expose the sodium annulus for visual examination revealed the void to be about 3/16-inch in frontal diameter. These two specimens had the smallest annulus of the three required (i.e., 0.260-inch diameter pellets, 0.280-inch inside diameter tubing, resulting in a nominal 0.010-inch radial gap). Specimens of the other two combinations (0.244-inch diameter pellets in 0.276-inch inside diameter tubing, and 0.203-inch diameter pellets in 0.230-inch tubing) are also being fabricated for testing.
2. The vanadium container is a short, plugged tube used to contain the vanadium alloy tensile specimen during irradiation. A requirement for the container is that it be filled with sodium and have at least one hole for communication between the sodium in the container and the sodium in the outer annulus. Five containers have been fabricated, bonded and evaluated. A satisfactory vacuum loading process has been identified which results in the complete filling of the container with molten sodium, absence of gas voids, and the complete wetting of metal surfaces. Cladding removal revealed an isolated shrinkage void in the thick region of the annulus. This condition is acceptable because such cavities will fill when the sodium remelts. However, an additional set of containers is provided in which cooling will be refined in an attempt to obtain directional solidification and minimize cavities.

3. Modifications to the sodium glovebox have been completed for the handling of the 40-inch long capsules. All components for capsule fabrication have been received. Fabrication processes and sodium bonding evaluation will be completed during the next quarter.

3.1.3 End Plug Welding Development

In the last quarterly report it was shown that satisfactory end plug closure welds could be achieved in about 90 percent of the attempts. To increase this percentage, it was proposed then that during the following quarter, study would be directed to improve the precision of the welding process, specifically: electrode positioning, tip geometry, alignment of the plug during insertion, uniform loading during insertion, welding current stability and programming, and heat balance. This program has been completed. Blow-holes are now avoided through use of improved welding process parameters and tooling. Tube wall thinning in the weld zone (undercut) has also been minimized.

A summary of results for the variables studied is presented below:

1. Electrode Positioning

The rack and pinion holder for the welding torch was replaced with a compound two-dimensional slide with graduated divisions. This equipment allows for consistent positioning of the weld to within ± 0.001 inch of the desired location. It has been demonstrated that to prevent blow-holes and excessive tube melting, the electrode should not be directly over the seam, but be slightly displaced onto the end plug. The new equipment can maintain this precise positioning.

2. Electrode Geometry

The importance of electrode tip geometry was not apparent until variations in weld nugget shape were noticed when using identical welding parameters. Many of the nuggets were shallow and wide instead of the desired narrow deep penetration welds.

Lower arc voltage was found to contribute to desirable welds as shown in Figure 300-2. At D/W ratios (ratios of weld penetration [D] to weld width [W]) of less than about 0.25, tube wall melting and shrinkage (also called I.D. undercut) becomes more pronounced. Variation in ratio was found to be caused by corresponding variations in the effective plasma length of the arc. This condition is illustrated in Figure 300-3 for two different tip shapes with equal arc gaps, but unequal effective plasma lengths. Measured arc voltage represents the sum of anode and cathode drops plus an incremental voltage drop proportional to plasma "climb" at the electrode tip. This climb increases the effective plasma length which increases the total arc voltage.⁽¹⁾ Figure 300-4 shows this effect on weld penetration where all welding parameters were constant. A freshly ground tip was used and produced a shallow nugget while a well-used blunt tip produced the deep penetration nugget.

Contour gages have been designed for accurately measuring and controlling the electrode tip shape and thereby controlling the precision of weld penetration.

3. Plug Alignment During Insertion in Tube

Several plugs were deliberately started into tubes in a cocked position during insertion. Radiographs were taken before and after welding to correlate the amount of interface separation with weld quality. Several blow-holes occurred that could be identified with areas exhibiting tube deformation and interface separation. Good correlation was obtained between interface separation and undercut after welding. This is the result of poor heat transfer across the separated interface which caused excessive heat build up and melting of these portions of the tube.

(1) Savage, W. R.; Struck, S.S.; and Ishikawa Y, "The Effect of Electrode Geometry in TIG Welding," Welding Journal 45 (12) Research Suppl., 4895-4965 (1965).

Figure 300-2 Effects of ARC Voltage on Weld Nugget Geometry

Figure 300-3. Schematic of Arc Characteristics as Influenced by Electrode Tip Geometry

NOTE THE UNDESIRABLE WIDE,
FLAT WELD NUGGET RESULTING
FROM FRESHLY GROUND ELECTRODE
(LEFT)

Figure 300-4. Weld Nugget Geometry as Influenced by Electrode Tip Configuration in 0.010-Inch Stainless Steel Tube to End Plug Welds

4. Copper Chill for Improved Heat Balance

Removing heat more rapidly from the tube surface during welding prevents excessive melting of the tube. Several sets of split-ring copper chills were made for fitting on 0.300 and 0.250-inch diameter tubing. The edge of the attached chills were located about 0.050 inch from the weld nugget region. It was found that current stability during welding was improved and excessive tube melting was controlled.

3.1.4 GETR Irradiation Pin Welding Process

Final GETR irradiation pin welding parameters have been documented and applied to the welding of three pins for the initial GETR (Phase I) capsule. All top and bottom end-plug weldments have been nondestructively inspected and found acceptable. The inspection tests used were: helium leak test, radiography, dye-penetrant examination, and visual examination at 10X magnification.

3.1.5 Eddy-Current Test Development

Sodium bonding is a heat treatment applied to fuel pins for the purpose of completely filling the annulus between the pellets and the tube wall with sodium and wetting these components with the sodium. The influence of time at temperature, type of vibrational energy, and type of pellet were not known. A program was started and completed this quarter which provided data to indicate an optimum process for bonding of GETR irradiation pins. This information is summarized in this section.

Specimen evaluation included radiography and eddy-current testing prior to and after the bonding treatment, followed by the removal of the cladding by peeling for visual examination of sodium adherence to the pellets.

Specimens were fabricated to reference GETR irradiation pin design except that alumina and Type 304 stainless steel were substituted for carbide fuel pellets. The bonding temperature for all specimens was $1000 \pm 50^{\circ}\text{F}$ since previous experience indicates that this temperature is sufficient to obtain wetting of stainless steel, alumina, and (U,Pu)C. All pins were subjected

to a bottom quench to minimize the chance of shrinkage voids which would form upon solidification of the sodium. Bonding cycle times longer than two hours duration was to be investigated only if the initial work indicated that two hours were not sufficient.

Twenty pins were fabricated and bonded at 1000°F according to the schedule shown in Table 300-1. The disposition of the pins after each bonding treatment was based on eddy-current test results. All pins were eventually destroyed by cladding removal as indicated by the letter S in the table. Correlation of eddy-current data and destructive examination was excellent.

Table 300-1 shows that after the first bonding treatment only one of ten pins not vibrated was satisfactory, while four of the ten pins vibrated was satisfactory. Vibration was, therefore, used for all rebonding. After the second two-hour bonding treatment, ten of the twelve pins were reclaimed. A third bonding treatment of these two pins produced good bonding. Thus, it appears that all pins can be satisfactorily bonded at 1000°F provided that they are held at sustained temperature. Vibration apparently accelerates bonding; vibration where the specimen is allowed to bounce in its holder is the best type. Finally, there is an indication from the tabulated data that stainless steel pellets may bond slightly faster than alumina pellets. When sufficient UC pellets become available, their bonding rate will be compared to the simulation pellets now being used.

GETR pins 1H, 1M and 1L, for insertion in GETR under UPC-400 have been fabricated and bonded. The bonding cycle is described below.

1. Place pins in magazine. Allow for a maximum amplitude of free flight of 1/8-inch.
2. Heat to 1000°F and hold for 15 minutes without vibration.
3. Vibrate at 1000°F for two hours.
4. Hold at 1000°F without vibrating for 15 minutes.
5. Remove magazine from furnace and immerse bottom in water to a depth of one inch until magazine cools to room temperature.

TABLE 300-1

TEST SCHEDULE FOR SODIUM BONDING PROGRAM

Pin Number	Pellet Stack Material	Spacer Rod and Hold-Down Spring	Initial Bonding Treatment	Disposition	2nd Bonding Treatment	Disposition	3rd Bonding Treatment	Disposition
M-105	Al ₂ O ₃	No	A	X	D	√S		
M-106	Al ₂ O ₃	No	A	X	C	√S		
M-108	Al ₂ O ₃	Yes	A	XS				
M-120	Al ₂ O ₃	Yes	A	X	C	X	D	√S
M-109	Al ₂ O ₃	Yes	B	XS				
M-110	Al ₂ O ₃	Yes	B	XS				
M-111	Al ₂ O ₃	No	B	√S				
M-112	Al ₂ O ₃	No	B	X	C	X	D	√S
M-113	Al ₂ O ₃	No	B	X	C	√S		
M-118	Al ₂ O ₃	No	B	X	D	√S		
M-115	Al ₂ O ₃	No	C	X	D	√S		
M-116	Al ₂ O ₃	No	C	√S				
M-114	Al ₂ O ₃	Yes	C	X	D	√S		
M-128	Al ₂ O ₃	Yes	C	√S				
M-117	304 SS	No	A	X	C	√S		
M-125	304 SS	No	A	X	D	√S		
M-126	304 SS	No	B	X	C	√S		
M-127	304 SS	No	B	X	D	√S		
M-121	304 SS	No	C	√S				
M-124	304 SS	No	C	√S				

TABLE 300-1 (Cont)

- A - 1/2-hour @ 1000°F without vibration
- B - 2-hours @ 1000°F without vibration
- C - 2-hours @ 1000°F with vibration, pins clamped in holder
- D - 2-hours @ 1000°F with vibration, pins loose in holder
- ✓ - Pin meets GETR criteria for Na bond quality as determined by eddy-current test
- X - Pin does not meet GETR criteria for bond quality as determined by eddy-current test
- S - Stripped of cladding for visual examination

The pins were radiographed and eddy-current tested at room temperature prior to and after bonding. Eddy-current tests were performed on a Magnetics Analysis Heli-Tester using a probe type coil. Traces were made using a Brush Mark 220 oscillograph recorder.

The eddy-current traces indicated absence of voids after the one bonding cycle described above. Correlation of radiographs with the eddy-current traces verified a random shifting of some pellets against the cladding. This condition is acceptable.

3.1.6 Sodium Deposition Studies

The engineers involved in this task are J. L. McKinney and J. Sutherland.

As reported in WCAP-3791-4, it was intended to install the sodium deposition studies equipment in a glovebox also used for sodium bonding development. Since it was found that sufficient space was not available in the box without hampering other operations, a decision was made to redesign the deposition studies apparatus to make it self-contained. Major modifications to the original design include the addition of oxygen and water content analysis equipment and an argon purification and recirculation system.

Figure 300-5 schematically indicates the revised experimental setup. Purified argon enters through the top of the bell-jar, sweeps downward, exits at the bottom, and is then purified and recirculated. Continuous sampling of the bell jar atmosphere for oxygen and water content determinations is provided for. A flowmeter is shown in the cover gas circulating loop so that data on sodium deposition versus gas flow can be obtained. Any sodium which is carried out in the cover gas will condense in the sampling trap, and the amount can be measured. The sodium pot and sodium dump tank are separated by a valve. This allows the liquid sodium to be pushed from the dump tank, via a pressurized argon supply, into the pot and held at any given level. The vapor trap prevents sodium plugging of the dump tank gas line.

The heater modules were changed to a new type of flexible heating tape that easily conforms to a non-uniform surface. These tapes will be wrapped around the sodium pot, dump tank, and all sodium and sodium vapor lines.

Figure 300-5. Sodium Deposition Apparatus, Schematic

Figure 300-6 shows the controlling circuit for the heating systems. There will be two such control circuits, one on the sodium pot and one on the sodium dump tank. The sodium and gas line temperatures are not critical, therefore, they are not controlled automatically.

Figure 300-7 is an illustration of the control and instrument rack. The temperature controllers are located at the top along with the control switches. The recorder will monitor temperatures at all critical points throughout the test apparatus. The ammeters give visual display of power input to the various heat systems. The O_2 and H_2O monitors, along with the thermocouple vacuum gauge are just under the ammeters.

Analysis of the sodium vapor transport problem was expanded to include a range of pool sizes and also the use of nitrogen and helium as cover gases.

Construction of the test apparatus is approximately 50 percent complete. Installation of heaters, insulation, and some electrical wiring is yet to be done. Figure 300-8 shows the stage of construction presently attained. Parts have been ordered to construct a controlling instrument rack for the test apparatus. Some of the parts have been received and a wiring diagram for them is presently underway.

The controlling phenomena on sodium vapor mass transfer is the diffusion of sodium atoms through the boundary layer and natural convection upwards caused by the temperature difference between the sodium pool and the hot cell atmosphere. Using dimensionless parameters, the general form of the equation relating mass flux to system variables can be expressed⁽¹⁾

$$\begin{aligned} \text{where: } Nu' &= f(Gr' \cdot Pr') \\ Nu' &= \text{Nusselt number prime} = \frac{m\ell}{D(\rho_s' - \rho_\infty')} \\ Gr' &= \text{Grashof number prime} = \frac{g\ell^3}{\nu^2} \left(\frac{\rho_s'}{\rho_\infty'} - 1 \right) \\ Pr' &= \text{Prandtl number prime} = \frac{\nu}{D} \end{aligned}$$

(1) E. G. R. Eckert, Introduction to the Transfer of Heat and Mass, p. 253.

Figure 300-6 Control Circuit for Sodium Pot & Pump Tank

Figure 300-7 Sodium Deposition Apparatus Control Panel

Figure 300-8. Status of Sodium Deposition Apparatus

and:

- m = mass flux of sodium vapor leaving pool
- ℓ = diameter of pool
- D = diffusion coefficient of sodium through atmosphere
- ρ_s = partial sodium density at surface
- ρ_∞ = partial sodium density at infinity
- ρ'_s = mixture density at surface
- ρ'_∞ = mixture density at infinity
- g = acceleration of gravity
- ν = kinematic viscosity of gas

More specifically, the empirical correlation of Boelter, et al.⁽²⁾

$$Nu' = 0.643 (G'r \cdot Pr')^{0.25}$$

which was developed for evaporation of water from a horizontal surface, has been used by G.E.⁽³⁾, with good results, for predicting mass transfer from small diameter pools.

Using this expression, the sodium transport rate was calculated as a function of sodium temperature and pool diameter. The atmosphere over the sodium pool was assumed to be argon at a bulk temperature of 100°F. The results are plotted in Figure 300-9.

Inspection of this figure indicates the sodium transport rate to be strongly a function of temperature. For a 1-inch diameter pool, the transport rate at a sodium temperature of 400°F is approximately 1×10^{-8} lb/hr. At 1000°F, it becomes approximately 2×10^{-4} lb/hr, an increase of over four orders of magnitude.

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- (2) L. M. K. Boelter, et al., "Free Evaporation into Air of Water from a Free Horizontal Quiet Surface:", Industrial Engineering Chemistry, Vol. 38, No. 6, 1946.
 - (3) Southwest Experimental Fast Oxide Reactor Development Program, Second Quarterly Progress Report, GEAP-4742, July-September, 1964.

Figure 300-9. Sodium Transport Rate vs Temperature
for Various Size Pools in Argon at 100°F

The transport rate is also quite obviously a strong function of diameter. The predicted transport rate from a 12-inch diameter pool at 1000°F is nearly 2×10^{-2} lb/hr , leading to approximately 2 pounds of deposited sodium during a 100-hour run.

Similar calculations were performed for helium and nitrogen atmospheres. The results were essentially the same as for argon, indicating no advantage, from a vapor transport standpoint, to their use.

3.1.7 Sodium Corrosion Studies of Cladding

The engineers involved in this task are E. C. Bishop, J. Baysden, and S. Schrock.

One mass transfer loop simulating the primary system of the W LMFBR (cladding in the hot leg, IHX tubing in the cold) was started at 1200°F. However, failure of a heater after 150 hours of operation forced abandonment of the run. The next run, which will contain reference primary system materials, is scheduled to start early in the next reporting period.

SECTION 5

UPC-310 - SODIUM BOND HEAT TRANSFER PERFORMANCE

A. A. Bishop, E. H. Novendstern, Hans Van Seuren, S. E. Majka

1. OBJECTIVE

The objective of this task is to determine the thermal and hydrodynamic behavior of the cladding-to-fuel bond in a sodium-bonded fuel rod under both steady state and transient conditions.

Tests will be initiated to study the hydrodynamic effects of fission gas release in narrow annuli. One of the tests will include "simulated pellets" fabricated with solid and/or porous materials. The narrow annuli will be filled with various fluids in order that the important properties be varied. As gas is released in the small annulus, bubble growth rate, agglomeration, and motion will be studied. These data will be used to calculate hot spot temperatures caused by the fission gas bubbles. Tests using stainless steel cladding and electric heaters inserted in a sodium filled annulus will be initiated upon completion of initial tests. An analytical program will be conducted concurrently with the experimental effort.

2. PRIOR WORK

The test section used to simulate the hydrodynamics effects of the bond was designed and assembled. Analytical work was also performed to determine the effects of stationary bubbles of various sizes on fuel rod temperatures.

3. CURRENT PROGRESS

3.1 "Simulated Pellets" Test

After successfully starting tests on this test assembly and receiving photographs of good quality, the test section was dismantled, cleaned, and reassembled.

Upon reassembly, the glass tube cracked badly. Several other tubes also cracked when reassembly was tried again. Failure of the glass tubes is believed to have been caused by overstressing of the glass in the end fittings.

This problem has been solved by installation of a tube with glass flanged ends. New end fittings were matched with this flange. The improved assembly has been disassembled and reassembled satisfactorily on several occasions. Experiments should begin in the next quarter.

3.2 Flat Plate Tests

Experimental work has begun this quarter to determine bond performance between two flat plates; one is glass and the other Type 304 stainless steel. The spacing between these flat plates can easily be controlled by using shim stock. The flat plate test assembly is shown in Figure 310-1.

The small space between the plates should closely approximate the annulus in a fuel rod. Cameras and a stroboscope are being used to investigate bubble motion. A typical photograph of the bubble motion, using multiple exposures on one photograph, is shown in Figure 310-2. Tests have been conducted using water, ethanol, methanol, acetone, and ethanol-water mixtures. Tests are currently being planned using NaK. Spacing between plates have been 0.025 inch and smaller.

Figure 310-3 illustrates some results of bubble motion in ethanol. The results indicate that a higher velocity is attained with bubbles having a larger bubble frontal area. This is due to their larger buoyancy force. Results also indicate that for smaller spacing, bubble velocity is lower.

3.3 Effect of Fission Product Contaminative of the Bond

During the lifetime of the reactor fuel, the sodium filled gap between the fuel and cladding will close due to swelling of the fuel under irradiation. Simultaneously, fission products change the thermal conductivity of the sodium bond.

ASSEMBLED TEST SECTION

TEST SECTION COMPONENTS

Figure 310-1. Sodium Bond Flat Plate Test Section

Figure 310-2. Photograph of Bubble in Motion

Figure 310-3. Velocity of Bubbles in Ethanol

Cesium is a fission product which can be of major importance in establishing the thermal conductivity of the bond. The formula proposed to cover the thermal conductivity change of the sodium bond with cesium content is:*

$$k_m = k_1 w_1 + k_2 w_2 - c(k_1 - k_2)(w_1 w_2)$$

$$c = +4.28 \text{ for } 200^\circ\text{C (392}^\circ\text{F)}$$

$$c = +2.68 \text{ for } 500^\circ\text{C (932}^\circ\text{F)}$$

The constant, c , is derived from the change in thermal conductivity of sodium with potassium content as shown in Figure 310-4. Since no data were found for Cs-Na solutions, it is assumed that these constants are applicable.

Additional assumptions made are:

1. The final Cs content at end of life is estimated to be approximately 20 wt percent.**
2. The change in thermal conductivity is rectilinear from k_1 to k_m .
3. The gap closing is rectilinear from 13 to 1 mill at end of life.

For 20 wt percent Cs and 200°C , the thermal conductivity is:

$$k_m = 47.4 \times 0.8 + 11.6 \times 0.2 - 4.28 \times 0.62 \times 0.16 = 16.2 \text{ BTU/h ft}^\circ\text{F}$$

For 20 wt percent Cs and 500°C , the thermal conductivity is:

$$k_m = 38.7 \times 0.8 + 11.6 \times 0.2 - 2.68 \times 27.1 \times 0.16 = 22.0 \text{ BTU/h ft}^\circ\text{F}$$

The conductivity as a function of time is:

$$k_t = k_1 - (k_1 - k_m) \left(\frac{t}{t_o} \right)$$

* Refer to nomenclature at the end of this Section 3.3.

** Personal communication, WARD, December 1967.

Figure 310-4. Thermal Conductivity of Sodium as a Function of K Content

The gap width is:

$$x_t = \frac{0.013 - 0.012 \left(\frac{t}{t_o} \right)}{12}$$

Using these equations, the gap conductance may be determined by:

$$h = \frac{k_t}{(a_1 - x_t) \ln \left(\frac{a_1}{a_1 - x_t} \right)}$$

Figure 310-5 shows the ratio of gap conductance to conductance at the beginning of life, as a function of dimensionless time, t/t_o . As it can be seen, the gap conductance increases with burnup even though the thermal conductivity of the bond decreases. This is caused by the progressive reduction of gap thickness.

These results illustrate that conductance cannot be predicted by using just the thermal conductivity of pure sodium and the gap width. As shown in Figure 310-5, the gap conductance predicted using the thermal conductivity of pure sodium is considerably higher than the Cs-Na alloy. Thus, the WARD design procedure of using 50 percent of this value, as is currently done, will be retained to correct the fission product contamination as well as other uncertainties such as non-wetability at the interfaces between the sodium, cladding and fuel.

Figure 310-5. Ratio of Gap Conductance to Conductance at the Beginning of Life as a Function of Burnup

3.4 Rate of Fission Gas Production in the Bond

The purpose of this phase of the sodium bond study was to determine the maximum growth rate of bubbles formed in the sodium bond by the release of fission gas.

The analysis was confined to radially expanding, stationary (with respect to elevation) bubbles. Fission gas generation rates were calculated from the local linear power rating and a value of 30 stable gas atoms produced per 100 fissions. The fission gas that is produced is not all released and some will remain trapped within the porosity of the fuel, thereby contributing to fuel swelling. The fraction of gas released, f , is the ratio of gas released to gas produced and it is calculated by the following equations*:

$$f = 1 - (1 - e^{-vt})/vt$$

$$v = 5.60 \exp [0.216 (\Delta\rho/\rho) - 26,000/RT]$$

where: $\Delta\rho/\rho$ = percent density change

T = centerline fuel temperature, °K

R = gas constant, 1.987 cal/gm mole °K

t = time

Figure 310-6 shows the gas release fraction versus burnup for several fuel centerline temperatures.

The total quantity of gas released at any time can be calculated by integration of the equation of fraction of gas released with respect to time. The volume of the bubble is then calculated from the equation of state.

Figure 310-7 shows, for two centerline temperatures, the time required for a bubble to completely fill the pin-to-cladding gap versus the burnup at the start of bubble formation. It is assumed that all the gas released goes into the formation of the bubble. The curves are based on a 14-mil cold gap width which decreases with burnup due to fuel swelling. The results shown in

* B. L. Harbourne, "Fission Gas Release From Carbide Fuels", AMS-542, November 1967.

Figure 310-6. Gas Release Fraction vs Burnup

Figure 310-7. Time Required for Bubble to Close Remaining Gap versus Burnup

Figure 310-7 indicate, for centerline temperature of 2650°F, a bubble which starts to form at a burnup of 100,000 MWD/Tonne will fill the remaining 4-mil gap in 0.045 days, or 65 minutes.

These results illustrate that the sodium bond may be replaced by fission gas in a short period of time. The bond will be replaced if the fission gases accumulate and do not rise.

3.5 Bubble Velocity

The work of White and Beardmore*, which discusses bubble motion in cylindrical tubes was reviewed for applicability to bubble hydrodynamics in the bond. Their work showed that for Eötvös numbers less than 4, the bubble remained stationary. However, when compared with some experimental data obtained from tests between flat plates, bubbles did not remain stationary for Eötvös numbers less than 4, when based on the gap thickness. Work is continuing to try to relate results in tubes to results in annuli.

* E. T. White, and R. H. Beardmore, "The Velocity Rise of Cylindrical Air Bubbles Through Liquids Contained in Vertical Tubes", Chem. Eng. Science, 1962, Vol. 17, pp. 351-61.

NOMENCLATURE

a_1 = inner radius of cladding (ft)

h = thermal conductance $\left(\frac{\text{BTU}}{\text{hr-ft}^2-\text{°F}} \right)$

h_i = initial thermal conductance $\left(\frac{\text{BTU}}{\text{hr-ft}^2-\text{°F}} \right)$

k_1 = thermal conductivity of Na

k_2 = thermal conductivity of Cs

k_m = thermal conductivity of mixture

k_t = thermal conductivity of mixture at time t

t = time (years)

t_o = lifetime of the fuel (years)

w_1 = weight percentage of Na

w_2 = weight percentage of Cs

x_t = gap width at time t (ft)

$\left(\frac{\text{BTU}}{\text{hr} - \text{ft} - \text{°F}} \right)$

SECTION 6

UPC-400 - THERMAL FLUX IRRADIATION STUDIES

A. Biancheria, J. R. F. Hayes, E. V. Khan, P. J. Levine
A. J. Lipps, D. J. Rop, J. A. Rylatt

1. OBJECTIVE

The objective of this task is to evaluate the in-pile performance of sodium-bonded, stabilized (U,Pu)C fuel elements. In-pile tests will be performed in currently available thermal flux facilities. The post-irradiation test results, which will include evaluations of fuel swelling, fission gas release, and fuel-cladding interaction, will be employed in fast breeder reactor design programs.

2. PRIOR WORK

Specific test objectives and irradiation exposure parameters for individual capsules have been defined based on the results of a statistical analysis of the overall irradiation program variables. The Phase I test parameters and design details have been summarized in previous reports. Fuel pin and capsule drawings were completed and fabrication commenced. Plans for the instrumentation console and fuel pin shipping containers were completed and the required items for each were ordered. The Irradiation Services Purchase Order (ISPO), and WARD purchase orders for V-RAFTS and special services were submitted to the AEC for approval. Initial plans for pre-and post-irradiation were outlined.

3. CURRENT PROGRESS

3.1 Summary

Three Phase I fuel pins have been completed and are undergoing inspection. Pre-assembly machining of the capsules is complete. The WARD review of the data package was completed. The instrumentation console was shipped to GETR. Design analysis of the Phase I capsules resulted in a change in the pellet

hold-down device. This change was required to circumvent difficulties encountered during sodium bonding. Design analysis were initiated for Phase II capsules. The purchase orders for irradiation services, special services, and V-RAFTs were approved by the AEC. The fabrication of V-RAFTs and shipping containers is complete.

3.2 Fuel Pins

Three Phase I fuel pins--two Cr_{23}C_6 -modified (1H and 1M) and one unmodified (1L)--were assembled, welded, and bonded. Detailed inspection results are being evaluated. Some pellet diameters and lengths deviate slightly from the original requirements (see UPC-200 and 300 tasks) but are acceptable and do not compromise the objectives of the tests. These pins will be shipped in January, 1968- for assembly in Capsule 1 of GETR if the completed inspection reports are satisfactory. Fuel fabrication for the remaining six pins in Phase I is continuing.

3.3 Irradiation Capsules

Capsule fabrication for Phase I is progressing well. All pre-assembly machining is completed. The Data Package on the capsule design and experimental safety report prepared by GE was evaluated by WARD and found to be acceptable with minor changes.

3.4 Instrumentation

The multipoint recorder and all console components were received and installed into the cabinet. The system was designed initially for operation with only one capsule, but a spare electrical cable was included to convert the system for handling three capsules simultaneously. The console was checked and found to be satisfactory and shipped to GETR, where it was received in good condition. The wiring diagrams and operating procedures are being prepared for use by GETR test operators.

3.5 Design

3.5.1 Nuclear Design

Nuclear analyses were performed during the past quarter to support the selection of test parameters for the Phase II experiments. Most analyses were performed by GETR at the request of WARD.

The Phase II tests are intended to achieve maximum fuel burnups of the order of 100,000 MWD/T compared to about 50,000 MWD/T for the Phase I tests. Achieving this burnup will require a change in the Phase II test parameters. This is demonstrated by Table 400-1 which summarizes the calculated heating rates for specimen W2GH as a function of fuel burnup. Heating rates are listed for start-of-irradiation, at 50,000 MWD/T, at 100,000 MWD/T, and at various rod bank positions during the GETR cycle. All powers are based on 45 MW reactor power, a GETR billing flux of 1.1×10^{14} $\eta/\text{cm}^2/\text{sec}$, and the V-RAFT positioned to maintain the highest constant power during that cycle.

After 100,000 MWD/T, the power generation in this specimen would drop to only about 50 percent of its initial value. This reduction is due to a depletion of fissile material and a buildup of fission products. Since this decrease in specimen power is not acceptable, some means must be found to maintain specimen power during Phase II.

Increasing the U-235 enrichment is one means whereby the specimen power could be maintained during the Phase II tests. At the request of WARD, GETR has performed a parametric study to investigate this possibility. All calculations were based on the following conditions.

1. All specimen and capsule dimensions identical to those of Phase I specimen W2GH
2. 89 percent dense unmodified hypostoichiometric ($\text{U}_{0.8}\text{Pu}_{0.2}$)C fuel
3. A reactor power of 45 MW
4. An unperturbed billing flux of 1.1×10^{14} $\eta/\text{cm}^2/\text{sec}$

TABLE 400-1
POWER GENERATION IN SPECIMEN W2GH
0 PERCENT ENRICHMENT

Inches Above Core Bottom	Initial	Power, Kw/ft		Rod Bank Inches
		@ 50,000 MWD/T	@ 100,000	
9	28.0	27.5	14.0	18.6
10	28.2	28.1	14.1	
11	27.8	27.8	13.9	
11	28.0	25.6	14.0	23.8
12	28.2	25.8	14.1	
13	27.8	25.5	13.9	
13	28.0	24.1	14.0	25.8
14	28.2	24.2	14.1	
15	28.0	24.1	14.0	
14	27.4	23.0	13.7	27.8
15	27.6	23.1	13.8	
16	27.4	23.0	13.7	

Initial power for P/A = 1.0 = 18.8 Kw/ft RAFT full in

Final power for P/A = 1.0 = 9.4 Kw/ft RAFT full in

5. The V-RAFT positioned to maintain the highest constant power (full-in) during that cycle. Lower powers can always be achieved.
6. Uniform fuel pump.

Tables 400-2 through 400-5 contain the initial and final power (at 100,000 MWD/T) for a specimen located in the peak flux position at various rod bank positions. Powers are tabulated for U-235 enrichments of 5, 10, 40 and 93 percent respectively. Note that these powers represent maximum values with the V-RAFT full in. Lower values could be achieved to coincide with the Phase I conditions by positioning the RAFT further out from the core center.

Figure 400-1 contains a plot of the maximum power that could be achieved in the test specimen at a rod bank position of 27.8 inches (end of cycle). Heating rates are plotted for both zero and 100,000 MWD/T burnups. From this plot, it can be concluded that the effect of fuel burnup on specimen power generation is virtually eliminated at a U-235 enrichment of about 40 percent. Furthermore, increasing the U-235 enrichment beyond 40 percent produces very little increase in the specimen heating rate.

Increasing the U-235 enrichment has an adverse effect on the power distribution within the fuel specimen. This is demonstrated in Figure 400-2 which contains a plot of the power distribution within the fuel specimen as a function of U-235 enrichment (no burnup). The effect of power distribution on fuel temperature was demonstrated in earlier studies (Uranium Plutonium Carbide Development Quarterly Progress Report for the Period Ending March 31, 1967, WCAP-3791-4). For the conditions stated above, the ΔT within the fuel specimen would be about 36 percent less with 40 percent enriched uranium than with natural uranium. For a heating rate of 28 Kw/ft, this would represent a reduction in fuel ΔT of about 180°F. However, if the heating rate in the enriched specimen were increased to about 31 Kw/ft, its initial fuel center temperature would be about the same as a specimen with natural uranium at 28 Kw/ft.

Results of these parametric studies indicate that the U-235 enrichment in the Phase II specimens be increased to 40 percent. Furthermore, the design heating rates in the Phase II specimens should be proportionately increased

TABLE 400-2

MAXIMUM SPECIMEN POWER GENERATION
5 PERCENT ENRICHMENT

Inches Above Core Bottom	Power, Kw/ft		Rod Bank Inches
	Initial	Final	
9	29.4	19.5	18.6
10	29.6	19.7	
11	29.2	19.4	
11	29.4	19.5	23.8
12	29.6	19.7	
13	29.2	19.4	
13	29.4	19.5	25.8
14	29.6	19.7	
15	29.4	19.5	
14	28.8	19.1	27.8
15	29.0	19.3	
16	28.8	19.1	

Initial power for $P/A = 1.0 = 19.7$ Kw/ft RAFT full in

Final power for $P/A = 1.0 = 13.1$ Kw/ft RAFT full in

TABLE 400-3
MAXIMUM SPECIMEN POWER GENERATION
10 PERCENT ENRICHMENT

Inches Above Core Bottom	Power, Kw/ft		Rod Bank Inches
	Initial	Final	
9	30.2	22.9	18.6
10	30.5	23.1	
11	30.0	22.8	
11	30.2	22.9	23.8
12	30.5	23.1	
13	30.0	22.8	
13	30.2	22.9	25.8
14	30.5	23.1	
15	30.2	22.9	
14	29.6	22.5	27.8
15	29.8	22.6	
16	29.6	22.5	

Initial power for $P/A = 1.0 = 20.3$ Kw/ft RAFT full in

Final power for $P/A = 1.0 = 15.4$ Kw/ft RAFT full in

TABLE 400-4
MAXIMUM SPECIMEN POWER GENERATION
40 PERCENT ENRICHMENT

Inches Above Core Bottom	Power, Kw/ft		Rod Bank Inches
	Initial	Final	
9	33.8	33.4	18.6
10	34.0	33.6	
11	33.6	33.2	
11	33.8	33.4	23.8
12	34.0	33.6	
13	33.6	33.2	
13	33.8	33.4	25.8
14	34.0	33.6	
15	33.8	33.4	
14	33.1	32.7	27.8
15	33.4	33.0	
16	33.1	32.7	

Initial power for $P/A = 1.0 = 22.7$ Kw/ft RAFT full in

Final power for $P/A = 1.0 = 22.4$ Kw/ft RAFT full in

TABLE 400-5
MAXIMUM SPECIMEN POWER GENERATION
93 PERCENT ENRICHMENT

Inches Above Core Bottom	Power, Kw/ft		Rod Bank Inches
	Initial	Final	
9	35.2	35.2	18.6
10	35.5	35.5	
11	35.0	35.0	
11	35.2	35.2	23.8
12	35.5	35.5	
13	35.0	35.0	
13	35.2	35.2	25.8
14	35.5	35.5	
15	35.2	35.2	
14	34.5	34.5	27.8
15	34.8	34.8	
16	34.5	34.5	

Initial and final power $P/A = 1.0 = 23.6$ Kw/ft RAFT full in

Figure 400-1. Maximum Power at 27.8-Inch Rod Bank

Figure 400-2. Relative Power Distribution in the Specimen
as a Function of U-235 Enrichment

(28 to 31 Kw/ft in the bottom specimen) to maintain the same initial fuel temperature as in Phase I. This may necessitate some redesign of the thermal shims in the capsule.

3.5.2 Thermal Design

During the past quarter, GETR supplied WARD with the thermal conductivity data used in their calculations of capsule temperatures. They also supplied computer output sheets and related nodal diagrams showing the radial and axial temperature distributions within a capsule. Gamma heating rates were also included in the data package. Since two-dimensional temperature calculations performed at WARD differed slightly from those calculated by GETR, a more detailed analysis and comparison is presently in process to resolve the differences. Calculations are also in process to determine the fuel temperature in the Phase I specimens as a function of position and burnup.

None of the Phase I capsules contains heaters. Since the ambient GETR pool temperature is approximately 100°F, the sodium in the test positions would be frozen prior to reactor startup. During reactor startup, heat is generated non-uniformly along the axis of the fuel column. Consequently, the sodium in the bond would begin to melt in the vicinity of the hottest fuel near the bottom of the pellet column. During melting, sodium increases in volume by 2.17 percent. If no outlet were available for the expanding sodium, a pressure could be exerted on the fuel cladding.

A thermal analysis was performed to evaluate the rate of sodium melting in the bond of the hottest specimen. All meltdown calculations were performed in one-dimension using radial and axial temperature profiles generated with the two-dimensional TOSS computer program. During reactor startup, sodium melting would be initiated near the fuel column midplane in the bottom specimen. The results of the thermal calculation showed it would require about eight seconds to melt all the sodium in the bond.

Allowing for no axial expansion, the short term pressure buildup (if no Na outlet is assumed) could be sufficient to plastically deform the cladding. A cursory stress analysis showed the deformation to be marginal between elastic and plastic yielding. However, axial expansion or movement of the sodium, is expected during the melting process to alleviate this problem. Tests are currently in progress to verify that this is the case.

Analyses were also performed to estimate the fission gas release rate from specimens irradiated to burnups of 110,000 MWD/T. These analyses were performed using the gas release and swelling data developed from other carbide fuels programs.

Fission gas release was evaluated for each of the three types of fuel: unmodified (U,Pu)C iron-modified (U,Pu)C and chromium-modified (U,Pu)C. Evaluations were based on an average heating rate of 27 Kw/ft and a fuel center temperature of 2100°F. These conditions correspond to those which would be expected for a specimen located at the bottom of a capsule in the peak flux position.

The results of this analysis are summarized in Table 400-6. In all cases, the upper limits of the estimates (+2 σ) are greater than 80 percent. Consequently, the Phase II specimens will probably be designed for 100 percent gas release.

3.5.3 Mechanical Design

During the past quarter, it was necessary to revise the design of the fuel column hold-down mechanism in the Phase I specimens. The original design consisted of a tubular spacer device (WARD drawings 675C913 and 675C914) located in the upper gas plenum of each specimen. With this design, an unsatisfactory capillary action developed during the sodium bonding process. The design was changed to a dumbbell-shaped device and a new drawing was released (500B541). This new design has eliminated the problem.

It is intended that the Phase II irradiations be extended to maximum burnups of about 100,000 MWD/T. If 100 percent of the gases generated during fission were released from the fuel, the pressure in the bottom specimen (worst case) would react about 680 psi by the end of irradiation. Whether the bottom specimens could sustain this pressure without failure under the proposed Phase II conditions (i.e., cladding OD = 0.300-inch, cladding ID = 0.276-inch, cladding ID temperature = 1308°F, and cladding ΔT = 98°F) is currently being investigated. A parametric stress analysis is being performed to determine the pressure-temperature limitations for the Phase II specimens. In performing this analysis, the following assumptions have been made:

TABLE 400-6
FISSION GAS RELEASE FOR BURNUP
OF 110,000 MWD/T

	Unmodified	Fe Modified	Cr Modified
Average burnup, MWD/T	110,000	110,000	110,000
Fuel center temperature, °F	2,100	2,100	2,100
Fuel swelling rate, percent $8v/v / 10^4$ MWD/T	5.1	3.5	2.3
Integrated fission gas release, percent of gas generated	63	51	34
Percent FGR - 1 standard deviation	79	73	58
Percent FGR - 2 standard deviation	94	89	81

1. Cladding OD = 0.300 inch, cladding ID - 0.276 inch.
2. Clad T is constant for all cases and equal to 98° F.
3. Minimum life of 45 cycles with 15 irradiation periods of 672 hours each (nominal)
4. Type 316 Stainless Steel Cladding

Most of the Phase II tests are designed for Type 316 stainless steel cladding with the remainder using Type 304 stainless steel. At the present time, very little information is available for either irradiated or non-irradiated Types 316 or 304 stainless steel at the elevated temperatures involved. Consequently, this analysis was performed for Type 316 stainless steel only using the limited amount of data available.

Cladding failures are being analyzed for three conditions: thermal stress ratchet, low cycle fatigue, and creep rupture. In this analysis, each condition is being evaluated parametrically as a function of cladding pressure and temperature. Creep analyses are in progress, but analyses have been completed for thermal ratchet and low cycle fatigue as follows:

1. Thermal Stress Ratchet

Thermal ratchetting has been evaluated in this analysis using the techniques described by D. R. Miller in an ASME paper (No. 58-A-129) titled "Thermal Stress Ratchet Mechanism in Pressure Vessels". If the combination of pressure and thermal stresses in a pressure vessel are excessive, an increment of growth can occur with each cycle of temperature variation. The growth strain, y , due to one cycle of temperature variation is given by:

$$y = 2E_y \left[M - 2 (M (1 - n))^{1/2} \right] (1 - v) \quad (1)$$

where $E_y = \sigma_y / E$

E = Young's modulus of elasticity

σ_y = tensile yield stress

$$m = \sigma_t / \sigma_y$$

σ_t = maximum tensile thermal stress calculated on an elastic basis

$$n = \sigma_m / \sigma_y$$

σ_m = tensile membrane or average stress due to interval pressure or applied load

ν = Poisson's ratio in the elastic range.

At the onset of thermal ratchet, $y = 0$. For this condition, equation (1) reduces to:

$$n = 1 - m/4 \quad (2)$$

To avoid thermal ratchet failure, the above condition must be satisfied. Equation (2) can be rearranged to give the maximum pressure, Δp , that can be sustained across the cladding without failure due to thermal ratchet. The resulting equation is:

$$\Delta p = (\sigma_y - \sigma_t/4) \frac{2t}{D_m} \quad (3)$$

where

t = cladding thickness

D_m = mean cladding diameter

σ_y = 0.2 percent yield stress

$$\sigma_t = \frac{E\alpha\Delta T}{2(1-\nu)}$$

α = coefficient of thermal expansion

Both σ_y and σ_t are temperature dependent. Equation (3) has been evaluated for the Phase II specimens as a function of cladding ID temperature. The results are plotted in Figure 400-3 and represent the design limit for thermal stress ratchet.

Figure 400-3. Pressure-Temperature Limits of the Cladding Failure Analysis

2. Low Cycle Fatigue

A low cycle fatigue analysis has been performed for GETR specimens using techniques currently in practice at WARD. The stress-strain curves for type 316 stainless steel at various temperatures were modified to take account of material creep effects for a "hold" time of 672 hours. These creep-modified stress-strain curves were then used as input data for the PLATUDE computer program, using a wide range of internal pressures.

For each mean temperature, the maximum strain $\epsilon_{T_{MAX}}$ at each value of internal pressure was obtained, and the corresponding values of E noted. The maximum stress amplitude based on elastic theory is given by:

$$\sigma_A = 1/2 E \epsilon_{T_{MAX}}$$

A graph of stress, S, as a function of number of cycles to failure at that stress, N, was constructed from data available in Reference 1. This graph was then modified to allow irradiation effects, Reference 2, and residual stress effects. However, it was assumed that the residual stress was zero after 672 hours hold time, and on the final modified value of S, a factor of 2 was assumed, to allow for any material flaws. The S-N design curve is shown in Figure 400-4. (Although Reference 1 is for Type 316 stainless steel at 650°C [1202°], the design curve is used for all cases, because no further information is available at this time.)

Using the PLATUDE computer program, the results for each mean temperature value were checked to ensure that the highest value of

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1. UKAEA TRG Memorandum 2360(s)
 2. "Fatigue Design Basis for Zircaloy Components", Nuclear Science and Engineering, 20, 1-12 (1964)

Figure 400-4. Stress versus Number of Cycles to Failure at 650°C

pressure at which a stable result was obtained, σ_A , was sufficiently small to give a value of N greater than 45 in Figure 400-4. Based on this analysis, a relationship was established between temperature and pressure for a design "cycles to failure" life of 45. The results were plotted in Figure 400-3 and represent the strain cycle failure limit for the Phase II specimens.

The curves developed for Figure 400-3 represent failure conditions. The projected internal pressure and cladding ID temperature for the Phase II specimens is 680 psi and 1308°F, respectively, for the worst case. This is based on a fuel burnup of 100,000 MWD/T and a 100 percent fission gas release from the fuel. The curves in Figure 400-3 indicate these conditions could be applied to the Phase I specimens without anticipated failure. However, the results of the creep analysis are not yet completed. When this analysis is completed, a design envelope will be established and the Phase II conditions will be selected.

3.6 Test Site Negotiations

The ISPO and WARD purchase orders for V-RAFTs and Special Services were approved by the AEC. The three V-RAFTs for Phase I are completed. Assembly of Capsule I will proceed upon receipt of the first three pins and inserted into GETR. Based on a January delivery of pins, initial insertion into GETR should be in March, 1968, (Cycle 95).

3.7 Fuel Pin Shipment

Fabrication of three specification shipping containers was completed. The package is comprised of an AEC specification 2R flanged pipe saddled in an ICC specification 15A wooden crate. The three pins from any one capsule will be nested in grooves in pieces of polyurethane foam and inserted in the pipe for shipment under Fissile Class III regulations.

3.8 Post-Irradiation Examination

Hot cell facility requirements are being reviewed to determine what modifications to existing facilities are needed. Plans were initiated to design and build a recording, continuous diametral measurement instrument for fuel pin dimensioning. The equipment will have to be capable of hot cell operation.

SECTION 7

UPC-500 - FAST FLUX IRRADIATION STUDIES

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1. OBJECTIVE

The objective of this task is to evaluate the in-pile performance of sodium-bonded modified and unmodified (U,Pu)C fuel pin in fast flux. Special comparative tests on alternate concepts such as gas-bonded (U,Pu)C or (U,Pu)O₂ fuel may be included in the tests.

2. PRIOR WORK

EBR-II will probably operate at 45-50 MWt during 1968. The original Phase I objectives cannot be attained at this power level. Revised Phase I objectives were defined and the required adjustments of nuclear, thermal and mechanical design were started. The principle design change was to increase the fuel stack height to 9.5 inches. The remaining core height will be taken up by a vanadium alloy tensile specimen in each capsule.

3. CURRENT PROGRESS

3.1 Summary

The design of the eleven Phase I capsules has been completed and fabrication development begun. Fuel and pin fabrication will be initiated during the next quarter. Preparation of the design and hazards report for experiment approval is in progress and will be completed early next quarter.

The objectives of Phase II have been defined and the design of the pins has been initiated.

3.2 Phase I Design

The revised Phase I pin and capsule design has been completed. It had been shown previously (WCAP 3791-10) that the fuel stack height in the most highly rated pin could be up to 9.5 inches. This height was chosen for each of the eleven pins. The fuel stack is held axially in position with a compression spring and hold-down bar to prevent movement during transit. The spring load is too small to significantly restrain axial fuel swelling during irradiation. However, up to 10 percent axial expansion may be accommodated.

The transverse fuel and cladding dimensions of the pin were not changed in the revised design so that the pin design (as far as swelling accommodation is concerned) is now more conservative because of the lower ratings to be used.

The fission gas plenum was designed on the basis of 100 percent release for 50,000 MWD/T burnup, with the normal restrictions on cladding stresses. The overall pin length for the worst case, 28.5 inches has been used.

Engineering drawings for fabrication have been prepared and approved. The fuel pin operating conditions are shown in Table 500-1. Although operating conditions are lower than those originally envisaged in the statistical experimental design, the integrity and validity of the experimental design is not significantly affected.

Details of the design and hazards analysis is being prepared.⁽¹⁾

3.3 Phase I Fabrication

Fuel for EBR-II pins cannot be manufactured simultaneously with that for GETR pins due to cross-examination problems. The EBR-II fuel contains fully enriched and the GETR contains depleted uranium. Fabrication work is therefore confined to component development which is detailed under UPC-300.

In summary, fuel pin end plugs and capsule end fittings have been successfully welded, techniques for loading the vanadium alloy tensile specimens have been developed and a simplified hold-down device is being developed. Bonding and bond evaluation trials on long fuel stacks are proceeding.

(1) D. J. Rop, et al, "Experiment Description and Hazard Analysis for Eleven Phase I Experimental (U,Pu)C Fueled EBR-II Capsules," WCAP-7127.

TABLE 500-1

OPERATING CONDITIONS FOR PHASE I CAPSULES

Capsule #	Fuel Type ⁽¹⁾	Fuel Density ⁽²⁾ (%)	Cladding (SS)	Max. Rating ⁽³⁾ (Kw/ft)	Average Rating (Kw/ft)	Max. Fuel Temp. ⁽³⁾ (°C)	Max. Clad Temp. ⁽³⁾ (°C)	Fuel ΔT (°C)
W1X	C	89	304	24.5	22.2	1015	555	400
W2X	C	89	316	24.5	22.2	1020	555	400
W2XT	C	89	316	24.5	22.2	1020	555	400
W3X	F	89	316	24.5	22.2	1020	555	400
W4X	U	89	304	24.5	22.2	1015	555	400
W4XT	U	89	304	24.5	22.2	1015	555	400
W5X	C	94	316	26.5	24.0	1020	565	400
W6X	C	89	316	16.5	14.9	830	515	270
W7X	U	89	316	16.5	14.9	830	515	270
W8X	C	89	316	28.5	25.8	1070	544	460
W9X	C	89	304	28.5	25.8	1063	544	460

Notes:

(1) C = Cr - modified fuel
 F = Fe - modified fuel
 U = Unmodified fuel

(2) % theoretical density

(3) Excluding hot spot factors

Most of the pin and capsule hardware has been ordered or is available. Vanadium alloy stock for the tensile specimens has been melted and swaged to rod. Specimen machining will be completed by the end of January.

3.4 Phase I Hazards Analysis

The design and hazards report required by ANL for experiment approval is in preparation. The report will be completed by February 7.

3.5 Phase II Design

Target operating parameters for Phase II capsules have been defined and are shown in Table 500-2. Duplicate fuel pins for subsequent TREAT testing will be identified at a later date. Tentative special purpose tests are also shown in Table 500-2.

A parametric analysis relating reactor power, pin rating, fuel diameter and swelling has been initiated and will be completed early next quarter. This analysis will be used to determine the maximum permissible ratings attainable in Phase II, and to define transverse dimensions required to meet the target operating parameters shown in Table 500-2.

4. FUTURE PROGRESS

1. The first two irradiation capsules, W8X, and W9W will be fabricated during the next period.
2. The parametric analysis of Phase II capsules will be completed and critical dimensions of the fuel defined.
3. Available data on the irradiation performance of mixed oxide fuel pins will be examined to select critical parameters for the oxide-fueled special purpose pins W4S, W5S, and W6S.
4. Detailed programs for the post-irradiation evaluation of the Phase I pins will be prepared and costed with the help of WPIF personnel.
5. To isolate thermal and irradiation effects some out-of-pile studies of fuel pin behavior will be required.

TABLE 500-2

EBR-II PHASE II TEST PARAMETERS

Capsule	Fuel	Density	Kw/ft	Clad	MWD/T	Statistical Group
W10X	U	89	35	316	50,000	1
W11X	C	89	35	304	50,000	2
W12X	F	94	35	316	50,000	3
W13X	C	89	25	316	50,000	Variance
W14X	U	89	25	316	75,000	1
W15X	C	89	35	316	75,000	1
W16X	C	89	25	304	75,000	2
W17X	F	89	35	304	75,000	2
W18X	F	94	25	316	75,000	3
W19X	U	94	35	316	75,000	3
W20X	C	89	25	316	100,000	1
W21X	F	89	35	316	100,000	1
W22X	F	89	25	304	100,000	2
W23X	U	89	35	304	100,000	2
W24X	U	94	25	316	100,000	3
W25X	C	94	35	316	100,000	3
K -- Special Purpose Pins						
W1S	C	89	~40	316	50,000	Fuel center temperature no hot spot factors, to be 1450°C (2640°F)
W2S	C	89	~40	316	75,000	
W3S	C	89	~40	316	100,000	
W4S	(U,Pu)O ₂	~92	~18-20	316	50,000	Smeared density 90%(typical)
W5S	(U,Pu)O ₂	~92	~18-20	316	50,000	Smeared density 85%(typical)
W6S	(U,Pu)O ₂	~92	~18-20	316	50,000	Smeared density 80%(typical)

6. Specification of appropriate tests will be initiated during the next reporting period and actual testing started following the fabrication of the irradiation test pins.

SECTION 8

UPC-600 - TRANSIENT IRRADIATION STUDIES

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1. OBJECTIVES

The objectives of this task are to investigate the performance and failure mechanisms of sodium-bonded (U,Pu)C fuel pins under transient conditions. Irradiated and unirradiated pins will be tested to evaluate the safety margins of fuel pin designs proposed for LMFBR use.

2. PRIOR WORK

Preliminary nuclear and thermal analyses demonstrated the need for an efficient thermal neutron shield to give uniform heating in the fuel, typical temperature profiles, and reasonably low power generation with respect to the TREAT core. The fuel pin to be tested is identical to that being used for EBR-II irradiation studies. The use of fully enriched fuel makes the nuclear and thermal problems more severe.

3. CURRENT PROGRESS

3.1 Thermal Analysis

Pulse width $t_{1/2}$ in TREAT increases as the initial reactivity insertion, $k_{ex(0)}$, decreases. With a maximum $k_{ex(0)}$ of 2.42 percent, $t_{1/2} = 0.2$ seconds. The total energy released, Q , decreases with decreasing reactivity insertion. To keep a constant pulse width, $t_{1/2}$, Q may be varied only by adjusting the thickness of the neutron shield and hence the relative pin-to-core power generation.

The physical model assumed for the thermal analysis is shown in Figure 600-1. The thick steel cylinder enclosing the fuel pin serves as a heat sink to

Figure 600-1. TREAT Capsule Model Used in Transient Thermal Analysis

suppress the temperature rise in the sodium bond and to maintain capsule integrity after a pressure build-up due to fuel pin failure.

The results of the thermal analysis are summarized in Table 600-1. The fraction of the fuel cross sectional area which attains temperatures, in excess of 2640 and 4500°F is given in the table. These are, respectively, the maximum normal operating (in a 1000 MWe system) and melting temperatures of the fuel. The table also gives the time period during which the center stays above these temperatures. To prevent vaporization of the sodium bond, the pin must be internally pressurized as indicated.

Figure 600-2 shows a typical radial temperature distribution at 0.5 second and 2.0 second following the initiation of a pulse. The temporal distribution of the same pulse is given in Figure 600-3.

3.2 Nuclear Analysis

Steady state nuclear calculations were carried out to predict the power level in a test capsule containing a (U,Pu)C fuel pin contained in a stainless steel capsule and shielded by a B_4C neutron filter. The filter modified the highly thermalized neutron spectrum of TREAT to more nearly simulate a hard, fast-reactor spectrum and to reduce the power density in the test pin relative to that in the TREAT reactor. Table 600-2 lists the properties of the fuel pins considered in this study.

The cross section of the capsule is similar to the one shown in Figure 600-1 except that part of the stainless steel heat sink was replaced by a B_4C neutron absorber with a 1.2 cm inside diameter.

Criticality calculations were carried out for the TREAT core with the ANISN transport theory code with the objective of selecting a multi-group cross section library capable of handling both the fast and the thermal end of the neutron spectrum. Good agreement with published data was obtained by using

TABLE 600-1

MAIN RESULTS OF TRANSIENT THERMAL ANALYSIS

$k_{ex}(0)$ (%)	Q (j/cm)	$T_{c_{max}}$ (°F)	$T_{s_{max}}$ (°F)	Frac. ≥2640 (%)	Frac. ≥4500 (%)	Time $T_c \geq 2640$ (Sec)	Time $T_c \geq 4500$ (Sec)	Required Pressure (Psi)
2.42	2600	3300	860	38	0	0.4	0.0	14.7
2.42	4000	4950	2260	84	18	0.9	0.3	170
2.42	5000	6100	2800	100	50	1.2	0.5	614
1.20	5000	4070	2120	70	0	1.2	0.0	110
1.20	6000	4850	2470	84	14	1.5	0.4	300

Figure 600-2. Radial Distribution of the Temperature during a Typical Pulse, $k_{ex} = 2.42\%$, $Q = 5000 \text{ j/cm}$

Figure 600-3. Temporal Distribution of the Temperature during a Pulse, $k_{ex} = 2.42\%$, $Q = 5000 \text{ j/cm}$

TABLE 600-2
 PROPERTIES OF FUEL USED IN NUCLEAR
 ANALYSIS OF TREAT EXPERIMENT

Pellet diameter (cm)	0.62 $\pm$ 0.002
Pellet density (g/cm ³)	11.7 $\pm$ 0.2
(U,Pu) in pellet (w/o)	91.5 $\pm$ 0.9
Pu in (U,Pu) (w/o)	20.0 $\pm$ 0.4
U-235 in U (w/o)	93.1 $\pm$ 0.5
Pu-239 and Pu-241 in Pu (w/o)	91.7 $\pm$ 0.5

the 26-group ABN cross section library⁽¹⁾ and an S_8 transport approximation. The TREAT core description used is that given in ANL-6174.⁽²⁾

To attain the lineal energy of 2600 j/cm indicated in Table 600-1, the ratio of the maximum power in the test pin to the maximum power in TREAT, i.e. $(P_{\max})_{\text{pin}} / (P_{\max})_{\text{TREAT}}$, must equal 15. A parametric study of the B_4C filter thickness versus $(P_{\max})_{\text{pin}} / (P_{\max})_{\text{TREAT}}$ was performed and the results are listed in the Table 600-3.

A filter of 1.6 cm thickness is required to attain the required power ratio of 15 and this filter has a very significant reactivity worth. This is undesirable, however, as indicated in Figure 600-4 the filter serves to flatten fission density distribution and a comparison of the spectra shown in Figure 600-5 indicates that the filter effectively modifies the neutron spectrum in the test fuel to a distribution resembling the neutron spectrum in a fast reactor.

3.3 Experimental Program

A three-phase experimental program has been outlined which replaces earlier plans. Its principle objectives are to define the limiting transient conditions which will cause failure and to study the mechanisms of pin failure.

The mechanisms of pin failure and the limiting transient conditions will be established on unirradiated pins, confirmed on unirradiated pins designed to simulate irradiated pins, and finally confirmed on a few irradiated pins.

The most probable causes of transient failure are boiling of the sodium bond and carbide fuel melting. Both are independent of fuel composition and density. The cladding material may influence the limiting conditions for failure.

(1) I. I. Bondarenko, ed., Group Constants for Nuclear Reactor Calculations, Consultants Bureau, New York (1964).

(2) D. Okrent, et al, The Reactor Kinetics of the Transient Reactor Test Facility (TREAT), ANL-6174.

TABLE 600-3
RESULTS OF ANISN CALCULATIONS

B ₄ C Filter Thickness (Cm)	Treat Multiplication Factor k	Reactivity Worth Of Capsule With Filter, k/k%	$\frac{P_{max}}{P_{center}}$ In Test Pin	$\frac{P_{max\ pin}}{P_{max\ TREAT}}$
0	1.0567	(Test Specimen Removed)		
0.276	1.0048	4.92	1.143	46.72
0.500	0.9986	5.51	1.093	33.09
0.800	0.9928	6.06	1.055	24.30
1.600	0.9745	7.79	1.018	15.03

Figure 600-4. Fission Density in the Fuel Sample for Various B_4C Filter Thicknesses

Figure 600-5. Comparison of Neutron Flux Spectrum in the TREAT Reactor with the Spectrum in a Fuel Pin

Phase I. Establish Limiting Conditions for Pin Failure.

All pins will contain 89 percent dense Cr-modified fuel clad in Type 316 stainless steel. The limiting conditions are as follows:

Pin 1	No bond boiling	$T_s \approx 800^\circ\text{C}$	No fuel melting	$T_c \approx 1500^\circ\text{C}$
Pin 2	No bond boiling	$T_s \approx 850^\circ\text{C}$	No fuel melting	$T_c \approx 1800^\circ\text{C}$
Pin 3	Bond boiling	$T_s \approx 905^\circ\text{C}$	No fuel melting	
Pin 4	Bond boiling		20% fuel melting	
Pin 5	Bond boiling		50% fuel melting	
Pin 6	Bond boiling		90% fuel melting	

The above six pins will bracket the conditions at which failure occurs.

Pins 7 and 8 will repeat the pins which bracket the failure point and attempt to improve the definition of failure conditions.

Phase II. Simulated Irradiated Pins

These simulated pins will contain Cr-modified fuel with density and diameter chosen to simulate burnups of 50,000 MWD/T. The pins will be pressurized to simulate fission gas release.

These pins are tentatively suggested:

Pin 9	50,000 MWD/T simulation, clad in Type 316 SS
Pin 10	50,000 MWD/T simulation, clad in Type 347 SS
Pin 11	100,000 MWD/T simulation, clad in Type 316 SS

TREAT conditions will be defined from Phase I experience.

Phase III. Five Irradiated Pins

All pins will contain 89 percent dense Cr-modified fuel and will have been irradiated under similar conditions (say 35 Kw/ft).

Pin 12	50,000 MWD/T, clad in Type 316 SS
Pin 13	50,000 MWD/T, clad in Type 347 SS
Pin 14	100,000 MWD/T, clad in Type 316 SS
Pin 15	100,000 MWD/T, clad in Type 316 SS
Pin 16	100,000 MWD/T, clad in Type 347 SS

Phase II and III are tentative suggestions and will be revised as the program develops.