WARD-3791-18

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WARD-3791-18

MASTER

URANIUM PLUTONIUM CARBIDE DEVELOPMENT

QUARTERLY PROGRESS REPORT

FOR THE PERIOD

ENDING MARCH 31, 1968

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Prepared for the New York Operations Office
U. S. Atomic Energy Commission
Under AEC Contract AT(30-1)-3791

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SECTION 1

INTRODUCTION

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

1. SUMMARY OF OBJECTIVES

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 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,"

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

WARD-3791-12, "Uranium Plutonium Carbide Quarterly Progress Report for the Period Ending December 31, 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 Task UPC-100, the effects of a reduction in FY-1968 funding were analyzed, and the FY-1968 scope was adjusted accordingly. The revised scope and overall project accomplishments and the re-estimate of monthly charges through June were documented and submitted to the USAEC New York Operations Office. A project review meeting was later held with USAEC (DRD&T) personnel to review the current program and progress and to discuss future plans. In addition, day-to-day contract administration and project planning, coordination, and control continued during the quarter.

Under Task UPC-200, the production of fuel pellets for GETR Phase I irradiation test capsules was completed, and archive samples of each GETR fuel type were prepared. The fabrication line was then closed down and cleared of all remaining fuel material for an accountability check followed by an AEC audit. Fabrication of material for the EBR-II test pins was started. The buttons made in 1966 at Battelle Memorial Institute were found to have oxidized and were unsuitable as starting material for the fuel. New buttons had to be prepared from raw materials; by the end of the quarter, two batches of material had been fabricated.

Under Task UPC-300, nine fuel pins for the GETR Phase I irradiations were fabricated, and preparations were made for the fabrication of EBR-II Phase I

test capsules. Dummy fuel pellets were used to establish and verify an acceptable pin-bonding technique. A fabrication process was completed for the vanadium alloy tensile specimen container assembly to be inserted in the EBR-II Phase I capsules. Mock-up EBR-II Phase I capsules were then fabricated and characterized using dummy pins and dummy container assemblies. Closure welding process development was started on three different irradiation pin sizes.

Under Task UPC-310, experimental testing in water using flat plate and cylindrical geometries continued during the quarter. Work on an analytical prediction of the thermal and hydrodynamic behavior of the bond was performed.

Under Task UPC-330, fabrication of the sodium vapor deposition test apparatus was completed, and all components were checked-out. The test conditions for initial operation of the apparatus were defined.

Under Task UPC-400, the fabrication, assembly, and inspection of all GETR Phase I capsules were completed. The capsules were inserted into designated GETR pool positions by V-RAFT's and brought to power late in March. The instrumentation operating procedures were completed. Arrangements for post-irradiation shipping and examination were started.

Under Task UPC-500, the EBR-II Phase I design and hazards report was completed for submittal to Argonne National Laboratory for their approval. EBR-II Phase II design studies were conducted and show the need for a high performance sub-assembly, probably containing 12 capsules, to attain ratings above 30 kw/ft. A review of the irradiation program was started to assure its conformance with current Commission policy on mixed carbide fuel development.

SECTION 2

UPC-100 - PROJECT ADMINISTRATION

P. L. Wyzenbeek, W. B. Heubel, A. E. Duval

1. OBJECTIVES

The objectives of this task are as follows:

- a. To assure timely completion of the full scope of the project within the financial plan.
- b. To provide day-to-day project planning, coordination, and control.
- c. To satisfy USAEC contract administration requirements.
- d. 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:

- a. Finalization of the contract.
- b. Establishment of project controls.
- c. Preparation and submittal of FY-1967 and FY-1968 Work Programs, (WCAP-3791-1 and WCAP-3791-5).
- d. Preparation and submittal of Quarterly Reports, (WCAP-3791-2, WCAP-3791-4, WCAP-3791-8, and WCAP-3791-10).
- e. Preparation and submittal of Topical Report, WCAP-3791-3, "Compatibility of Stabilized Hypostoichiometric Uranium-Plutonium Monocarbide Fuels with Stainless Steel".
- f. Preparation and submittal of (U,Pu)C Development Irradiation Test Program, (WCAP-3791-6).

- g. Initiation and submittal of monthly letter progress reports.
- h. Initiation and submittal of monthly reports of actual costs.
- i. Application for and receipt of license to operate the plutonium fuels development laboratory.
- j. Order for and receipt of initial requirements of plutonium metal and natural uranium.
- k. Negotiation of thermal irradiation services at GETR.
- l. Negotiation and placement of subcontract with GETR for design and fabrication of three irradiation capsules for irradiation of fuel pins.
- m. 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.
- n. Receipt and acceptance of Contract Modifications 1, 2, 3, and 4.
- o. 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 January 1, 1968 to March 31, 1968 were as follows:

- a. Receipt and acknowledgement of Contract Modifications 5, 6, and 7.
- b. Preparation and submittal of WARD-3791-12, "Uranium Plutonium Carbide Development Quarterly Progress Report for the Period Ending December 31, 1967".
- c. Submittal of monthly progress letter reports.
- d. Submittal of monthly reports of actual costs incurred.

- e. Analysis of the effects of a reduction in FY-1968 funding. The FY-1968 work scope was adjusted, and the revised FY-1968 scope, the overall project accomplishments, and the re-estimated monthly charges through June were documented and submitted to the USAEC New York Operations Office.
- f. Review of Uranium Plutonium Carbide Development Project with USAEC (DRD&T) personnel. The current program and progress were reviewed, and future plans were discussed.
- g. Continuing effort to broaden the scope of the facility license for the Plutonium Fuels Development Laboratory to add operational flexibility and to allow larger amounts of special nuclear materials to be stored and used.
- h. Initiation of preparation of the Work Program for FY-1969.

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

- a. 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.
- b. 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 operating techniques and equipment trials in the Plutonium Laboratory were conducted with depleted uranium. After sufficient experience had been gained with use of the equipment and the major problems had been overcome, plutonium was introduced into the line and fabrication of GETR irradiation specimens started.

Initial difficulties were encountered, and each operation appeared to have its own characteristic problems. In retrospect, it now appears that many of these difficulties were either directly or indirectly associated with the high oxygen values that prevailed in the fabrication line during that period. Carbide powder was found to oxidize readily to form an oxycarbide and, in this form,

appeared difficult to press and sinter to the specified densities. Increases in the applied pressure produced slight increases in the fired densities, but also resulted in appreciable capping and other pellet defects.

The use of secondary containment vessels for powder handling and storage greatly reduced the oxidation rate to the stage where pellets of acceptable composition and dimensions could be prepared. Homogeneity of the arc-melted button, the powder, and the stabilized dispersion also improved during this period, and work was started on the effects of some of the process variables. Most of the pellets representing GETR fuel were produced during the last few weeks of the fourth quarter.

Difficulty was experienced in obtaining a good lasting polish on UC and (U,Pu)C metallographic samples, and it was found through analysis that the lubricating fluid contained >500 ppm water. Solvents also broke down in the argon purifier causing a massive increase of impurities in the argon atmosphere. The polishing problem was overcome by substituting a silicone oil-kerosene mixture for the faulty Metadi fluid.

Metallograph examination revealed that gross defects, e.g., capping, in some of the earlier samples were caused principally by faulty pressing techniques. Chemical etching with 85 percent orthophosphoric acid proved satisfactory for Cr_{23}C_6 -stabilized UC and (U,Pu)C and revealed uniform grain size and pore distribution.

Attempts were made to correlate milling time with average powder particle size (as determined by using the Fisher sub-sieve analyzer). However, tests showed that a crust of oxide forms on each powder particle size.

A limited number of immersion density determinations were carried out; during these determinations, dried xylene was substituted for ethylene dibromide as the immersion vehicle because halogen ions were detected in the argon atmosphere.

2.2 Related Work Not Charged to the Contract

The vacuum cathodic etcher was received from NUMEC and used outside the box line. Satisfactory results were obtained with stainless steel samples.

Water in the Fisher sub-sieve analyzer was replaced with silicone oil before installation into the fabrication line. The apparatus was then calibrated in the argon atmosphere and used to determine the average particle size of zirconia, chromium carbide, and iron samples.

3. CURRENT PROGRESS

Work performed during this period is reported below in four parts: GETR Phase I fuel fabrication, investigation of process variables, EBR-II Phase I fuel fabrication, and characterization data.

3.1 GETR Phase I Fuel Fabrication

The initial fuel requirements for the nine GETR fuel rods were completed in January. During this work, the ability to recycle both stabilized and unstabilized fuel pellets was demonstrated. Dimensionally out-of-specification material was reprocessed from the crushing/grinding stage, and fuel pellets prepared by this method have been included in the irradiation program.

Also during January, archive samples of each fuel type were prepared, and other samples were encapsulated for future compatibility work. These samples provide the physical record of fuel material characteristics at this stage of the program and will be retained for future comparison with the irradiated pins.

Upon completion of this work, the fabrication line was closed down and cleared of all remaining GETR fuel material. A preliminary accountability check of the line indicated acceptable levels of plutonium, and this was verified by the complete inventory, followed by an AEC audit.

The material flow diagram for the fabrication of the GETR Phase I fuel is shown in Figure 200-1. The process variables and characteristics of the GETR Phase I fuel specimens are listed in Table 200-1.

3.2 Investigation of Process Variables

3.2.1 The Effect of Oxygen

It is now evident that the minimum levels of oxygen in the final sintered

--- OUT OF SPECIFICATION RECYCLE ROUTE

Figure 200-1. Material Flow Diagram for GETR Phase I Fuel Preparation

TABLE 200-1

PROCESSING VARIABLES AND CHARACTERIZATION OF GETR PHASE I IRRADIATION SPECIMENS

Material Type	Pin/Batch No.	Mill Time (Hours)	Addition ^a (%)	Sieve Size (Mesh)	Pressure (tsi)	Fire Temp. (°C)	Fire Time (Hours)	% Theoretical Density (Green)	% Theoretical Density (Fired)	Weight Loss (%)	Length Shrinkage (%)	Diameter Shrinkage (%)	Avg. Grain Size (Metallography) (μ)	Avg. Pore Size (μ)	Immersion Density (% Theoretical)	Open Porosity (%)	Closed Porosity (%)	Pu/U Ratio	Pu (%)	U (%)	Cr (%)	Fe (%)	Total U + Pu (%)	Carbon (w/o)	Oxygen (w/o)	Nitrogen (w/o)	Total Metal Impurities ^b (w/o)	Carbon Equivalent
(U,Pu)C Fuel	1H/72	2.5	5.01 Cr ₂₃ C ₆	325	55	1550	2	74-75	84-89	1.50	5.1	6.9	15	20	No Sample Available	No Sample Available	No Sample Available	0.2498	18.14	72.64	4.18	--	90.78	4.34	0.1179	0.004	0.077	4.43
	1L/126	1.5	--	325	30	1550	2+1+2	71-72	87.5-88	0.70	4.2	6.9	No Sample Available	No Sample Available	"	"	"	0.2420	18.46	76.29	--	--	94.75	4.42	0.1887	0.016	0.121	4.57
	1M/71	2.5	3.85 Cr ₂₃ C ₆	325	55	1550	2	74-76	88-90	0.64	4.8	6.5	"	"	"	"	"	0.2425	17.85	73.63	3.67	--	91.48	4.53	0.1336	0.009	0.073	4.64
	2H/130	1.5	--	325	30	1550	1	70-71	86-88	0.60	5.6	7.4	"	"	86.6 - 87.6	11.3 - 11.6	1.8 - 2.1	0.2521	19.14	75.91	--	--	95.05	4.54	0.1883	0.006	0.112	4.69
	2L/154	2.5	1.51 Fe	200	30	1450	0.5	73-74	88-90	0.30	4.1	6.6	10	12	91.9	4.2	5.9	0.2475	18.47	74.65	--	1.49	93.12	4.57	0.2411	0.033	0.041	4.78
	2M/145 ^c	2.5	4.00 Cr ₂₃ C ₆	200	30	1550	1+1	72-73	89-91	0.32	5.5	7.9	15	10	88.1 - 88.5	0.6 - 1.2	10.7 - 10.9	0.2487	18.42	74.06	1.92	--	92.48	4.71	0.2919	0.003	0.096	4.93
	3H/148 ^c	2.5	2.50 Fe	200	30	1500	1	72-74	87.5-90.5	0.39	6.5	7.0	14	32	89.9	5.8	4.3	0.2490	18.34	73.65	--	2.70	91.99	4.28	0.3830	0.068	0.075	4.63
	3L/135	2.5	3.75 Cr ₂₃ C ₆	200	30	1550	1+2	72-73	89-90	0.44	6.2	7.6	No Sample Available	No Sample Available	No Sample Available	No Sample Available	No Sample Available	0.2442	17.90	73.29	3.52	--	91.19	4.45	0.2831	0.010	No Sample Available	4.67
	3M/147 ^c	2.5	3.00 Cr ₂₃ C ₆	200	30	1550	1	70.5-71.5	87.5-88.5	0.38	5.8	8.3	10	15	84.9 - 87.8	0.0 - 1.5	12.2 - 13.6	0.2396	17.71	73.91	2.85	--	91.62	4.60	0.2887	0.006	0.136	4.83
UC Insulators	1H , 1M/68	2.5	3.82 Cr ₂₃ C ₆	325	50	1550	2	70-72	89-92	0.38	6.8	8.4	No Sample Available	No Sample Available	88.3 - 89.1	9.8 - 10.6	1.1 - 1.6	--	--	91.53	3.60 ^d	--	--	4.50	0.1466	0.003	No Sample Available	4.61
	1L , 2H/113	16.0	--	325	30	1550	1	68-73	88-89	0.52	7.3	8.3	"	"	88.0 - 89.0	10.0 - 10.6	1.4 - 2.0	--	--	95.32	--	--	4.47	0.2039	0.014	"	4.63	
	2M, 3L/123 3M	2.5	3.75 Cr ₂₃ C ₆	325	30	1550	2	70-71	87-89	0.10	7.3	9.5	"	"	No Sample Available	No Sample Available	No Sample Available	--	--	91.41	3.54 ^d	--	--	4.55	0.2862	0.014	"	4.78
	2L, 3H/152	2.5	2.71 Fe	200	30	1550	1	74-75	91-93	0.53	7.6	6.2	18	12	93.0 - 93.8	3.9 - 5.0	2.3 - 3.1	--	--	92.71	2.71 ^d	--	--	4.36	0.1947	0.022	"	4.53

^a0.5 Carbowax added in every case

^bTungsten not included, taken as 0.50 w/o

^cThese pins were reprocessed material - for route see Figure 200-1

^dBased on figures in column four

pellets that can be obtained with the present method of fabrication is between 2000 and 3000 ppm.

As the minimum level obtained in the prepared powder prior to pelletizing is 200 to 500 ppm, and as the same handling precautions are observed throughout the pelletizing process, the increase in oxygen is believed to be due to a relatively constant source. With the present processing techniques, four possible sources should be considered:

- a. Alcohol (used as a solvent)
- b. Carbowax (used as a binder)
- c. Cr_{23}C_6 or Fe (used as stabilizers)
- d. Sintering atmosphere (the argon used contains 100 ppm O_2 , but is "purified" by passing over red-hot graphite in the furnace)

Of these, carbowax is believed to be the most likely source. The Cr_{23}C_6 , Fe, and carbowax were analyzed for oxygen, and the results are given in Table 200-2. From these, it will be appreciated that the Cr_{23}C_6 and Fe cannot make significant contributions to the oxygen level, while the carbowax may readily provide 2,500 ppm. The alcohol and argon sintering atmosphere are believed to be noncontributing for the following reasons:

- a. The alcohol is evaporated at the comparatively low temperature of 50°C during processing and no weight gain of the powder has been observed.
- b. The substitution of other solvents, e.g., trichlorethylene, for the alcohol has resulted in pellets with similar oxygen contents.
- c. Alterations in sintering times have had no appreciable effect on the oxygen contents from pellets of similar batches.
- d. The substitution of high-purity argon for bottle argon did not lower the oxygen levels.

At present, the use of reduced oxygen content binders is being considered; e.g., Cranco, paraffin wax, or naphthalene.

TABLE 200-2

OXYGEN CONTENTS OF VARIOUS ADDITIVES

Material	Typical w/o Added to Carbide Powder	Oxygen Content of the Material (%)	
		By Composition	By Analysis
Carbowax	0.5	50	Not Available
Cr ₂₃ C ₆	1.00 to 5.44	0	0.51
Fe	1.07 to 3.12	0	1.00

3.2.2 The Effect of Carbon

Attempts to study the effect of carbon on powder oxidation rates and on pressing and sintering characteristics of pellets were marred by non-reproducible results. Chemical analyses had to be used as the major source of data, but scatter on duplicate results indicated that basic problems first had to be solved. Three sources of discrepancy were indicated, as follows:

a. Inherent errors in the analytical method

A statistical study of the carbon results indicates the following confidence levels for material containing approximately 4.6 weight percent carbon.

2σ (95 percent confidence) for standards, ± 0.082 weight percent

2σ (95 percent confidence) for samples, ± 0.110 weight percent

A value of ± 0.15 weight percent has been assumed to cover all sampling and analytical errors and is the presently accepted tolerance level. Work is continuing to reduce this figure.

b. Carbon variations due to binder additions

While other laboratories have reported carbon contamination from binders, no pickup attributable to carbowax has been observed in the present work. Two factors could, however, mask any such increase. These are: (1) the tolerance on the results, ± 0.15 weight percent (see above), and (2) removal of carbon as CO during sintering.

c. Carbon variations due to Cr_{23}C_6 additions

Some discrepancies have been noted between the analyses of carbide powder and those for subsequent chromium-stabilized fuel. An unexpected decrease in the carbon value of the final pellet has been observed, as can be seen in Table 200-3. (Cr_{23}C_6 contains 5.68 percent carbon and increased additions should result in higher carbon figures.) Proposed reasons for discrepancy are: (1) variations in the chemical analyses - paragraph (a) above; (2) burnout of carbon during sintering - as in paragraph (b) above, (3) passification of

TABLE 200-3

SUMMARY OF CHARACTERISTICS OF FUEL FABRICATES FOR EBR-II TEST PINS

Batch No.	Chemical Analysis (w/o)						Mill Time (Hrs)	% Additions	Sieve Size (Mesh)	Pressure (tsi)	% Theoretical Density (Green State)	Fire Temp. (°C)	Fire Time (Hr)	% Theoretical Density (Fired)
	Powder			Pellet										
	C	O ₂	N ₂	C	O ₂	N ₂								
501	4.71	0.20	0.06	4.67	0.34	0.05	2.5	0.5 Carbowax	200	20	72	1550	2	87
								2.00 Cr ₂₃ C ₆						
502	4.63	0.04	0.04	4.59	0.29	0.002	2.5	0.5 Carbowax	200	20	72	1550	2	89
								2.00 Cr ₂₃ C ₆						

Cr_{23}C_6 by the formation of an oxide film during analysis.

Work on all aspects is continuing.

3.2.3 The Effect of Powder Preparation Techniques

The powder preparation procedure has been modified to achieve the following objectives:

- a. a further reduction in the oxygen content of the final sintered pellet;
- b. a greater uniformity of material from batch to batch; and
- c. the production of a powder that is more amenable to pressing.

The present powder preparation procedure is shown in Figure 200-2. The -20# precrushed material is used to achieve a uniform mill feed material. The first blending stage is in the formation of a uniform slurry, thus coating each particle with a layer of carbowax. Additional blending stages are included to insure batch homogeneity, and the final stage of sieving through 35# mesh is used to achieve appropriately sized granules for pressing. Previous work using a 100# sieve at this stage resulted in a powder that caused capping problems during pressing.

Additional work is contemplated on the effects of milling time, sieve size, type, and percentage of binder.

3.3 EBR-II Phase I Fuel Fabrication

3.3.1 Production of (U,Pu)C Fuel Pellets

Work on the EBR-II Phase I fuel fabrication started with an examination of the buttons made in 1966 at Battelle Memorial Institute (BMI). The analyses are given in Table 200-4, with the original results quoted by BMI. The oxygen contents are very high, from 0.22 to 1.25 weight percent, and this is attributed to a slow, continuous oxidation of the material during storage. Because of these high oxygen figures, the material was unsuitable as starting material for the fuel, and new buttons have been prepared from the raw materials.

Figure 200-2. Powder Preparation Flow Diagram

TABLE 200-4

CHEMICAL ANALYSIS OF BATTELLE (U,Pu)C ARC-MELTED BUTTONS

Button Number	WARD Analysis (w/o)			Batteille Analysis (w/o)		
	C	O2	N2	C	O2	N2
G-6	4.29	0.222	0.002	4.54	0.018	0.014
G-7	4.27	0.613	0.021			
G-11	4.38	0.403	0.013	4.47		
G-12	3.72	0.689	0.007			
G-13	4.39	0.475	0.090	4.41		
G-14	4.78	0.271	0.003			
G-15	4.58	0.363	0.004			
G-16	4.54	0.496	0.004	4.52		
G-18	4.32			4.53		
G-19	4.60					
G-21	3.23	0.893	0.172	3.84	0.023	0.015
G-22	4.44	0.315	0.097			
G-23	3.99			4.42		
G-29	4.36	0.412	0.005			
G-30	4.08					
G-31	4.13					
G-32	4.16	0.541		4.77	0.038	0.017
G-33	3.15	1.248				
G-34	4.06			5.78		
G-35	4.09					
G-36	3.83	1.041	0.015	4.21		
G-37	3.89					
G-38	4.19	0.768	0.007	4.45		
G-39	3.95					
G-40	4.23			4.19		
G-43	4.50			4.19		
G-45	4.51			4.27		
G-46	2.32					
G-47	4.26					
G-48	3.81			4.09		
G-49	4.02					
G-50	3.67					
G-51	4.28			4.68		

Modifications to the powder preparation procedure have been noted above (3.2.3 of this section) while the pressing and sintering procedures remain similar. Pressures of 20 tsi with a prior 15-second dwell at tsi have yielded green pellets with densities of 72 percent of theoretical. Sintering of these pellets at 1550°C for two hours produced fired pellets with densities approaching 89 percent of theoretical. Two batches of material had been fabricated by the end of the quarter, and the results are given in Table 200-3.

3.3.2 Production of UC Insulator Pellets

The uranium carbide insulator pellets, which are situated at either end of an irradiation fuel pin column, were manufactured under "non-plutonium" conditions at the Westinghouse Astronuclear Laboratory. These pellets have now been manufactured and delivered to the Plutonium Laboratory to be included in the EBR-II irradiation pins.

The type, physical size, and number of these pellets are shown in Table 200-5. The uranium carbide was synthesized by a process of carbothermic reduction of the dioxide at approximately 1550°C under a vacuum of $\sim 10^{-5}$ mm mercury. After reduction, the carbide was converted to powder ($< 5\mu$) in a tungsten carbide mill and subdivided into three parts; the first part was converted to a stoichiometric fuel by the addition of carbon; the second was stabilized by the addition of Cr_{23}C_6 ; and the third was stabilized by addition of Fe. These materials were pressed and sintered to give pellets of the required density. The pellets were then ground to size according to the specifications.

Samples of carbide at the powder and sintered stage were taken for chemical analysis of C, O, and N, and selected pellets was taken for ceramography. Oxygen analysis of the pellet fell between 1500 and 2000 ppm.

The program demonstrated that a hypostoichiometric carbide can be produced by carbothermic reduction of the dioxide. The reduction was carried out under vacuum, leading to a low overpressure of CO gas which obviously assisted in the completion of the reaction. However, the carbothermic reduction of mixed $(\text{U,Pu})\text{O}_2$ may lead to preferential problems. The program demonstrated that kilogram batches of oxide + carbon mixes can be reduced and converted to the hypostoichiometric carbide.

TABLE 200-5

DISCRIPTION OF PELLETS OF UC BLANKET/INSULATOR FUEL

Type ^a	Dia(in.)	Length(in.) ^b	Density (g/cm ³)	No. of Pellets	c(w/o)	C _E (w/o) ^c	Cr(w/o)	Fe(w/o)
C	0.245-0.243	0.378-0.372	11.7±0.2	16	4.66±0.07	< 4.80	1.88-5.44	-
U	0.204-0.202	0.378-0.372	12.1±0.2	4	4.6 ±0.1	< 4.80	-	-
U	0.245-0.243	0.378-0.372	12.1±0.2	8	4.6 ±0.1	< 4.80	-	-
F	0.245-0.243	0.378-0.372	11.6±0.2	4	4.5 ±0.15	< 4.75	-	1.08-3.12
C	0.204-0.202	0.378-0.372	11.7±0.2	4	4.66±0.07	< 4.80	1.88-5.44	-
C	0.261-0.259	0.378-0.372	11.7±0.2	8	4.66±0.07	< 4.80	1.88-5.44	-

^aC = Cr₂₃C₆ Modified Hypostoichiometric Uranium Carbide

U = Unmodified Hypostoichiometric Uranium Carbide

Fe = Iron Modified Hypostoichiometric Uranium Carbide

^bInsulator Corners 0.003-0.010 Radius Chamfer

^cC_E w/o = w/o C + 0.75 w/o O ± 0.857 w/o N

w/o O < 0.2 w/o N < 0.05

3.4 Evaluation and Characterization

3.4.1 Contract Work

A variety of carbide samples having a wide range of compositions have been metallographically examined. Mounting procedures have been revised so that samples can be vacuum cathodically etched. Two principal methods have been adopted; the more familiar one is the use of "pre-molds" which, as their name implies, involve hot mounting the sample with a prefabricated copper impregnated plug to provide "back" electrical contact. The other technique is based on the same principle, but consists of fabricating a mount using non-conducting bakelite at the polishing face and copper-impregnated bakelite for the conducting base of the mount. Both methods proved successful in that electrical contact was established, and the carbide samples were satisfactorily vacuum cathodically etched. This mounting technique also provided a means for applying back contact during electrolytic etching.

The grinding and polishing procedure was streamlined since it was felt that excessive grinding and polishing was being done. Experiments showed that some stages of grinding might be omitted without deterioration of the final polish. Grinding papers of 120, 180, 240, and 400 mesh are now used with dried kerosene as the lubricant. Final polishing on metcloth using 6 and 1/4 μ diamond pastes is now preferred. The diamond paste is spread with 4 to 5 drops of silicone oil, and dried kerosene is once again used as the lubricant. Since this abbreviated polishing technique was adopted, the "orange peel effect" has been eliminated.

Most of the samples were examined longitudinally rather than transversely. The extra information thus obtained compensates for the extra material removed. Longitudinal examination was further justified when some Fe-stabilized pellers were found to have a skin of approximately 500 μ depth, an effect not apparent from a light transverse polish.

While the electrolytic etching techniques were not discarded, emphasis was placed on vacuum cathodic etching and chemical swab etching with an 85 percent orthophosphoric acid solution. Although excellent results were obtained with the vacuum cathodic etching technique, its limitations, discovered during operation, periodically required the use of alternative techniques. In

particular, a sample having a porosity exceeding 14 percent or one containing cracks became badly stained during the etching discharge. This was attributed to breakdown of foreign matter entrapped in the pellet during the grinding and polishing operation. Efforts to overcome this effect included prolonged ultrasonic cleaning, outgassing, and impregnation of the polished face with an epoxy resin. This last technique excluded the presence of dirt particles but allowed possible epoxy breakdown during vacuum cathodic etching. Mixed success with this technique points to other factors as yet not fully evaluated. For instance, it is known that a prolonged discharge causes breakdown of unprotected bakelite mount material; consequently, a teflon shielding disc, which exposes only the pellet area, is often placed over the mount. Nevertheless, the use of low potentials for short durations, with lengthy cooling intervals, has resulted in good etches even when no shielding was used.

Another disadvantage of the vacuum cathodic etching technique is the gross variation in the degree of etching which occurs across the face of some samples, the central area being etched more deeply. Teflon shielding discs with slit apertures have been used in an attempt to overcome this etching gradient, but the effect has not been completely eliminated with certain samples, particularly the Fe-stabilized samples.

Chemical etching has been improved by swab rather than by immersion techniques using "aged" solutions. This has been especially successful in improving grain boundary definition in Fe-stabilized UC and (U,Pu)C samples, e.g., where a slightly "aged" 85 percent orthophosphoric acid solution was used. The structures thus revealed provide an explanation of the very low volume of open porosity detected by immersion density techniques in some Fe-stabilized pellets. Such pellets had a density of approximately 90 percent theoretical, with 9 percent closed and 1 percent open porosity. It is felt that an excessive firing temperature had, in this instance, caused melting of the Fe-stabilizer, which had subsequently been drawn to the surface of the pellet by capillary action during the early stages of sintering. This effectively sealed the pores, giving a resultant low open porosity value (Figure 200-3). To some extent this is conjectural since direct identification of the phase

Figure 200-3. Photomicrograph (X600) of Zoned Fe-Stabilized (U,Pu) Pellet, Showing Dense Skin Area.
(3.14% Fe, 4.22% C, 147.4 ppm O₂, 449 ppm N₂
0.25 Pu/U)

has not as yet been made. However, it is hoped to examine such pellets on an electron microprobe and an x-ray diffractometer to confirm this.

Improvements in electrolytic etching have resulted indirectly from the use of copper impregnated mounts in conjunction with back application of the electrode, e.g., superior etch control and less pronounced etching gradients across the sample have been effected (See Figure 200-4, the structure of a Cr_{23}C_6 -stabilized (U,Pu)C sample.)

Further examination of some GETR Phase I type Cr_{23}C_6 -stabilized (U,Pu)C pellets by chemical, electrolytic, and cathodic etching techniques has revealed a large amount of light-colored second phase material. Theoretically, this phase should be a solid solution of $\text{Cr}_{23}\text{C}_6 + \text{Cr}$, which is left following the stabilization of hitherto hypostoichiometric (U,Pu)C. In this particular case, however, the amount of second phase was considered too great in conjunction with the analytical chemistry results to be accounted for in this way, raising the possibility of its being $(\text{U,Pu})_2\text{C}_3$ in solid solution with a carbide of chromium (Figures 200-4, 200-5, 200-6). In these photographs, different aspects of the etching techniques are quite apparent.

The earlier Fe-modified samples produced all showed the "skin" effect referred to above. Figures 200-3 and 200-7 show the dense skin area of a zoned pellet compared with the structure of homogeneous specimens which were obtained later. Until the areas are fully evaluated, composition of the zones areas is conjectural. However, comparison with work carried out elsewhere, e.g. at Atomic Energy Research Establishment, Harwell, indicates that the major light phase is (U,Pu)C containing small areas of $(\text{U,Pu})_2\text{C}_3$, which precipitates out. The dark acicular phase is thought to be an Fe-rich phase, which crystallized out on cooling.

During this period, a transition was made from the use of depleted to enriched uranium following the completion of the GETR Phase I program and the initiation of the EBR-II Phase I program. Consequently, time was necessarily devoted to rigorous accountability checks and removal of all U-238 that was not required prior to the introduction of material containing U-235.

Routine analytical chemistry determinations have been made to determine C, O, N, Pu/U, and Pu contents for the various fuels. Figures for total metallic

Figure 200-4. Photomicrograph (X600) of Cr_{23}C_6 -Stabilized (U,Pu)C Pellet; Etched Electrolytically with 1:1:1 Sulphuric:Acetic:Lactic Acids, for 5 Seconds at 5 v.

(4.00% Cr_{23}C_6 , 4.71% C, 2919 ppm O_2 , 27 ppm N_2 , 0.25 Pu/U)

Figure 200-5. Photomicrograph (X600) of Pellet Shown in Figure 200-4; Swab Etched for 20 Seconds "Aged" 85% Orthophosphoric Acid.

Figure 200-6. Photomicrograph (X600) of Pellet Shown in Figure 200-4; Vacuum Cathodically Etched at 1.5 kv under 150 μ Argon for 4 Minutes.

Figure 200-7. Photomicrograph (X300) of an Homogeneous Fe-Stabilized (U,Pu)C Pellet. (2.1% Fe, 4.58% C, 3374 ppm O₂, 1611 ppm N₂, 0.25 Pu/U).

impurities, tungsten content, and gas content of pellets have been obtained at NUMEC. No major difficulties have been encountered with WARD equipment during this period. Indeed, an improvement has been effected by placing the absorption bottles of the "Carbotrane" outside the glovebox, such that they can now be weighed outside.

Estimates of the total error in the results (analytical plus sampling) at a 95 percent confidence level are as follows:

C	- $\pm$ 0.15 w/o
N ₂	- $\pm$ 30%
O ₂	- $\pm$ 25%
Pu	- $\pm$ 0.2%
U	- $\pm$ 0.8%

Alternative analytical techniques are currently being considered to improve these values.

3.4.2 Related Work Not Charged to the Contract

Equipment for the compatibility program was designed and requisitioned during the quarter ending December 31, 1967. Most of it has now been received, including the furnaces which have been checked against the pin holder length for a constant temperature zone.

SECTION 4

UPC-300 - FUEL ELEMENT FABRICATION DEVELOPMENT

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1. OBJECTIVES

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

2. PRIOR WORK

Equipment was commissioned for fabricating, testing, and characterizing irradiation test fuel pins. Development progressed in pellet loading, in sodium bonding, in end-plug welding, and in characterization tests such as: eddy-current trace analysis of bonded pins, radiographic interpretation of internal component positioning, and visual observation and correlation with other test results after removal of the cladding.

3. CURRENT PROGRESS

3.1 Contract Work

3.1.1 GETR Irradiation Pins Fabrication

The nine fuel pins required for the GETR Phase I irradiation program were fabricated and inserted in the reactor during this quarter. No pins were rejected, and all fabricated pins were acceptable in accordance with tests listed below:

- a. Eddy-current traces for voids and unbonded areas of a prescribed maximum size (0.004-square inch frontal area)

- b. Radiographs of end plug welds for penetration and quality
- c. Radiographs of full length for internals location
- d. Dye penetrant test for surface cracks or imperfections
- e. Helium leak test to ascertain absence of voids in fabricated pins
- f. Dimensions and straightness survey
- g. Weight of various components

3.1.2 EBR-II Irradiation Test Components Fabrication

3.1.2.1 Fuel Pin

Twelve mock-up fuel pins have been fabricated, and additional pins will be fabricated during the next quarter to establish and verify an acceptable process. Stainless steel or alumina pellets were used to simulate carbide fuel pellets. All other components were identical to the final drawing requirements. The combination of 0.260-inch diameter pellets in 0.280-inch I.D. tubing was used exclusively in this initial work to study the smallest radial gap. Sodium was melted to cover the pellets at 500°F and the bonding temperature was 1000°F for two hours.

A brief description of observations during characterization of the pins follows. The first two pins had numerous voids which were detected by the eddy-current test. These voids were correlated with the eddy-current trace by subsequent destructive examination. The voids were attributed to incomplete cleaning of components. Of the next three pins, one had no voids, another had one void, and the third had one 0.06-inch frontal diameter void. The next five pins were used for insertion in capsules. While the last two pins had no voids, they showed severe lateral shifting of pellets which is detectable by the eddy-current test and destructive examination.

During the next quarter, investigation will be concentrated on the changes that occur within the bonded fuel pin after it is subjected to the capsule bonding cycle. The procedure will be to characterize the capsule bond destructively and non-destructively and then to recharacterize the fuel pin itself.

3.1.2.2 Container for Vanadium Tensile Specimens

A fabrication process is complete for the tensile specimen container assembly. The container has a 0.062-inch diameter hole drilled in one end plug. A pool of sodium is melted in a vacuum chamber, the container immerses itself in the sodium, and the chamber is then slowly pressurized to two atmospheres inert gas pressure. With this procedure no gas entrapment is possible.

Twelve containers, sodium-loaded by this vacuum technique, have been characterized by eddy-current testing and destructive examination. Only minor shrinkage cavities have been observed. Various operation temperatures from 500°F to 970°F were tried, with the high temperatures giving superior wetting and reproducibility of results.

3.1.2.3 Capsule Assembly

Capsules are about 40-inches long by 0.312-inch I.D. and enclose a 28-inch fuel pin and a 2.735-inch-long tensile specimen container. These two components must be sodium bonded to the capsule.

Characterization of the first few assemblies showed that unbonded void regions occurred which could be attributed to inadequate cleaning. Good bonding was achieved with extreme cleanliness. Hardware components were cleaned in acetone, followed by hot Alconox solution and hot water rinse, and finally swabbing with alcohol.

A weighed quantity of extruded sodium stick is placed in the capsule and melted under vacuum. Inert gas is then introduced and the pin and container inserted. The reverse procedure, in which the sodium was melted in vacuum with the pin and container in place, caused ejection of sodium particles because of the restricted annular gap for gas evolved from the melting sodium.

3.1.3 Closure Welding Development

Welding process development was started on the three different irradiation pin sizes: 0.300-inch O.D. x 0.010-inch wall thickness, 0.300-inch O.D. x 0.012-inch wall thickness, and 0.250-inch O.D. x 0.010-inch wall thickness. Both Type 316 and Type 304 stainless steel tubing were used. The procedures

and techniques that were qualified for GETR Phase I tubing with 0.012-inch wall thickness were modified for the 0.010-inch wall tubing. As expected, more wall thinning occurred during welding of the 0.010-inch tubing than of the 0.012-inch wall tubing. This is attributed to the greater difficulty of guiding the end plug into the thinner wall tubing without distorting the tube. If the two parts are not in intimate contact, the gas gap acts as an insulator and causes excessive melting of the tubing. A double pass technique has been developed which appears to produce more consistent welds.

With the present design for EBR-II irradiation pins, the hold-down spring extends about one inch beyond the end of the tube. Using available equipment, the end-plug could not be inserted in perfect axial alignment without distorting the end of the tube. A new end-plug insertion device has been designed, constructed, and is operational. In the twelve mock-up fuel pins fabricated, the three-to-five pound force exerted by the spring on the end plug has presented no problems. Final verification of welding parameters suitable for the EBR-II irradiation pins will be completed early in the next period.

The design of the end-plugs for the EBR-II irradiation capsule is different than that for the fuel pins. Qualification of the welding and inspection procedures for these capsule welds has been completed. The same TIG welding equipment and weld box have been used for this work as are used for other components. The type 304 stainless steel tubes have dimensions of 0.375-inch O.D. by 0.020-inch wall, and are of sufficient mass that tube wall thinning is eliminated.

Figure 300-1 presents a schematic illustration of the critical dimensions and a typical resulting weld nugget shape. Figure 300-2 shows a sectional end-plug closure and illustrates how the root of the groove machined in the end plug becomes isolated after welding. After establishing the optimum welding parameters, a qualification program was initiated to insure a high confidence level for the welding procedure as determined by radiographic and metallographic results. The average weld penetration was 150 percent of the tube wall thickness, but a part of the root groove remained in all sections. Dimension X, which is considered to be a limiting wall thickness, was measured

(a)

(b)

a. SCHEMATIC FIT-UP SHOWING CRITICAL DIMENSION "X" OF WELD NUGGET.

b. PHOTOMICROGRAPH (50X) OF TYPICAL WELD NUGGET.

Figure 300-1. End Plug Closure Weld on EBR-II Capsule

Figure 300-2. Micrograph (9X) of Longitudinally Sectioned EBR-II End Closure Mock-up.

(Dark grey area is etched weld nugget isolating remnant of machined groove)

and averaged 0.030 inch ± 0.005 inch. The maximum increase in the outside diameter after welding was less than 0.003-inch, which is acceptable for insertion into EBR-II shroud dimples. The results of the weld qualification indicated sound weld quality with no evidence of internal or external cracking, porosity, or tube wall thinning. These results were derived from radiography, helium leak testing, dye penetrant examination, and metallographic sectioning. Detailed procedures have been written based on the qualification results obtained.

3.1.4 Atmosphere Control of Sodium Handling Glove Box

Studies were started to learn more about the behavior and capabilities of the sodium-handling glove box equipment located at Cheswick. The fabrication procedure for GETR and EBR-II irradiation test pins includes the handling of sodium in a purified, recirculating, inert atmosphere glove box system. This system consists of a 78 cubic foot evacuable chamber equipped with six glove ports, a 1 x 1 x 2-foot transfer chamber, a large full box cross-section access door, two 1 x 4-foot viewing windows, and four bottom access ports. The glove material is a low sulfur grade neoprene rubber. Two of the four bottom access ports have recently been adapted for low temperature (approximately 400°F) sodium meltdown operations with the heating elements located outside the box.

The vacuum system consists of a CVC Type PMC-6B six-inch oil diffusion pump backed by a Kinney Type KS-47 (47 cfm) mechanical pump. The argon or helium atmosphere is purified by a Vacuum Atmospheres Model HE-373-B-I "Dri-Train" recently rebuilt at the factory. Briefly, this system uses a copper charged furnace operating at 600°C to react with oxygen and a dryer containing approximately 25 pounds of Linde 13X molecular sieve for moisture removal. The atmosphere is circulated by a positive displacement type blower at a flow rate of 40 CFM. The atmosphere is continuously monitored for oxygen by a Westinghouse oxygen analyzer and for moisture impurities by a CEC moisture monitor.

During the developmental phase of GETR and EBR-II pin fabrication procedures, problems associated with atmosphere control continued. The diffusion pump was found to cause backstreaming of hydrocarbons into the box. These

hydrocarbons adversely affect the Westinghouse oxygen analyzer, causing it to give erroneously low readings. For this study the diffusion pump was not used, and overnight evacuation with only the mechanical pump produced about 35 microns residual pressure. Helium backfilling resulted in an atmosphere with oxygen and moisture contents of 3 and 1 ppm, respectively. These values indicated that the Dri-Train can ultimately remove oxygen and moisture to less than 1.0 and 0.5 ppm, respectively.

The first experiment was designed to determine the oxygen removal rate of the purifier. The procedure involved releasing 1,000 cubic centimeters of air in the box with the aid of a small vibrating diaphragm pump to effect nearly instantaneous release. The oxygen and moisture concentrations were then continuously monitored. A freshly cut surface of sodium was also exposed in the box at selected oxygen concentrations to visually observe the sodium oxide film formation as a function of oxygen concentration. These data are presented graphically in Figure 300-3. The moisture content remained constant at 0.6 ppm for the duration of the test.

The sodium formed a bluish-black oxide in five seconds at the 160 ppm and 110 ppm oxygen levels; while at 50 ppm and 20 ppm, a gray film was formed in 5 minutes and 6.5 minutes, respectively. It is also interesting to note the excellent response time of the Westinghouse oxygen analyzer to an oxygen excursion. Future tests will provide data at lower oxygen levels.

A second experiment was devised to study the effect of oxygen diffusion through the gloves on the quality of the box atmosphere. To accomplish this, the box was isolated from the purifier, and oxygen concentration readings were taken approximately every hour. The first part of the experiment consisted of taking readings with air on the operator side of the gloves and these data are presented in curve 1 of Figure 300-4. The curve shows an increase in the oxygen level within the box, but at a low enough rate that the purifier should be able to continuously remove it.

The second part of the experiment consisted of placing the gloves inside the chamber, sealing the glove ports, and alternately evacuating and backfilling the gloves several times. In this manner, a relatively pure atmosphere was maintained on both sides of the gloves. Curve 2 of Figure 300-4 shows no

Figure 300-3. Oxygen Removal by Vacuum Atmospheres Dri-Train

Figure 300-4. Oxygen Contamination Due to Permeability of Gloves

detectable oxygen increase for this condition and illustrates the leak tightness of the box. Slightly positive box pressure was maintained for both tests. Future tests will employ a small pump to draw a sample; these tests will study the effect of a slightly negative box pressure.

SECTION 5

UPC-310 - SODIUM BOND HEAT TRANSFER BEHAVIOR

A. A. Bishop, E. H. Novendstern, S. E. Majka

1. OBJECTIVE

To determine the thermal and hydrodynamic steady state and transient behavior of a sodium bond in fast reactor carbide fuel elements.

Tests will be initiated to study the hydrodynamic effects of fission gas release in narrow annuli. One of the tests will include "simulated pellets" that are fabricated from solid and/or porous materials. The narrow annuli will be filled with various fluids in order to vary the important fluid properties. 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

Two different test sections were designed and assembled. One test section consisted of cylindrical pellets inserted in a glass tube. Gas was injected through a hole in one of the pellets into the sodium bond. Figure 310-1 shows the location of bubbles for a sodium bond having a width of 0.063-inch. The second test section consisted of a fluid contained between two flat plates. The separation of the plates was varied from 0.010-inch to 0.025-inch. Gas was injected through a small hole in one plate into the fluid, and the motion of the bubbles was observed. Analysis was also performed to determine temperature distribution behind a stationary bubble, the effect of fission products in the bond on the effective bond thermal conductivity, and the rate of fission gas release on displacement of the sodium bond.

Figure 310-1. Gas Bubbles in a Simulated Fuel Rod Having Eccentrically Located Fuel

3. CURRENT PROGRESS

Work through January was performed under Commission sponsorship. However, because of FY 1968 funding reductions, this work has been funded by Westinghouse since February. Work reported below covers the entire quarter, but no further reporting of this work will be made.

3.1 "Simulated Pellets" Tests

Tests have been initiated on this apparatus using solid bronze pellets. Pellets, 0.875 and 1.000-inch in diameter, were inserted in a glass tube whose I.D. was slightly larger than 1.000-inch. The test fluids were water, acetone, ethanol, and glycerin. Both concentric, eccentric, and randomly stacked pellets were used. Figure 310-1 shows that stationary bubbles can exist under "ledges" of fuel pellets which were randomly stacked. Other data are being reduced, and the effect of vibration on bubble motion will be determined.

3.2 Flat Plate Tests

Experimental work continued on the flat plate experiments. Bubble motion and size were photographically studied through the front glass plate. The spacing between the glass and stainless steel plate was accurately controlled. Tests have been performed with (a) a 0.020-inch hole for a gas inlet between smooth plates, (b) a 0.020-inch gas inlet hole between plates which were roughened by sand blasting and (c) a 0.030-inch gas inlet hole between smooth plates. Fluids used were water, NaK, ethanol, methanol, acetone, and ethanol-water solutions. Much of the data is still being reduced.

An attempt to obtain a correlation between the motion of gas bubbles and the size of the annular gap has been started. Work has also been performed to determine the maximum size of stationary bubbles. It has been found that for the 0.020-inch diameter hole with smooth plates, the data appear to be fairly consistent. When the dimensionless group, $\sqrt{A/d}$, the ratio of the bubble frontal area to the gap spacing, is plotted against the bond number,

$\rho d^2/\sigma$, the data are correlated by the following equation

$$\sqrt{A/d} = 1.49 / \left(\frac{\rho d^2}{\sigma} \right)^{0.66} \quad (1)$$

with a 95 percent confidence level of ± 15 percent. Figure 310-2 shows the experimental results. The correlation is based on limited data and will be revised as more data from other tests become available. The maximum size stationary bubble in the photograph was used to obtain this correlation.

3.3 Effect of Fission Gas Bubbles on Fuel Rod Heat Transfer

Fission gas bubbles in the sodium bond will result in a poor local gap conductance, thus causing the design temperature limits of the fuel or sodium to be exceeded. A parametric study was made to determine the effects of various sized stationary fission gas bubbles.

As shown in Figure 310-3, the fuel surface temperature is very sensitive to bubble thickness when it is small compared to the gap thickness. As the bubble thickness increases, a point is reached where the insulating effect of the bubble is complete, and a further increase in thickness does not appreciably affect the temperature distribution. The surface temperature increase caused by the bubble is found to be very localized; temperatures quickly drop close to "no bubble" values just outside of the area covered by the bubble. The maximum fuel pellet temperature will not necessarily be at the pellet centerline. As the amount of surface covered increases, the fuel surface temperature at the center of the bubble gradually approaches and eventually exceeds the centerline temperature at the same axial coordinate. For a thick bubble (> 5 mils) the surface temperature equals the centerline temperature at approximately 60° of blockage.

The analysis is presently being extended to study the fuel surface temperature transient in the region vacated by a bubble once it begins to move.

3.4 Effects of Axial Gaps

Axial gaps filled with sodium could develop between fuel pellets. An analysis was performed to determine the steady state temperature distribution in the

Figure 310-2. Bubble Area versus Bond Number

Figure 310-3. Surface Temperature at Bubble Center versus Bubble Thickness.

fuel pellet at various gap widths. The two dimensional model used assumed symmetry about the pellet axial centerline. The system consisted of cladding, sodium bond, and a fuel pellet bounded on one axial face by a sodium filled gap of uniform thickness and extending for the axial centerline to the cladding.

Figure 310-4 shows the temperatures at the pellet-gap interface and at the axial centerline (both at radius = 0) at various gap widths. The fuel pellet temperatures at the fuel sodium gap interface are lowered because of the higher thermal conductivity of sodium relative to the fuel. As the gap width is increased, more heat, which would normally flow radially through the pellet, is routed instead axially to the sodium gap and then radially outward. At the low values of gap width which would most probably occur, the depression of pellet-gap interface temperature is not significant, and temperatures greater than the sodium boiling point will occur. The temperature at the pellet axial and radial centerlines is not significantly reduced even at large gap widths.

3.5 Boiling in the Sodium Bond

The axial gap analysis discussed above showed the possibility of sodium boiling occurring in the axial gap between two fuel pellets. If such boiling did occur, the large volume of vapor created would create a bubble in the sodium bond possibly more severe than the fission gas bubbles discussed in 3.3 of this section.

A literature search obtained data on nucleate boiling in sodium. Holtz^[1] calculated a superheat of 474°F as that which would be required to initiate nucleate boiling in sodium at 1 atmosphere. Krakoviak^[2] extrapolated the 90°F of superheat required for water to sodium and obtained a value of 774°F. The value of 90°F corresponds to distilled, deaerated water, which should be similar to very high purity sodium. Krakoviak pointed out that because of the high thermal conductivities of liquid metals, the volume of superheated liquid would be large and "... when a bubble nucleates, the energy of superheat from the entire volume is released in a matter of microseconds in a sudden 'explosion' of liquid into vapor". This unstable boiling of sodium was confirmed experimentally by Marto and Rohsenow^[3]. If unstable boiling then did

Figure 310-4. Fuel Surface and Centerline Temperature Rise versus Sodium Gap Width.

occur, the resulting sodium vapor bubble would push out from the axial gap into the highly subcooled region in the sodium bond. Florscheetz and Chao^[4] presented curves of bubble radius versus time for inertia controlled and heat transfer controlled bubble collapse. They defined a parameter, B_{EFF}

$$B_{EFF} = \psi^2 \left(\frac{\rho c \Delta T}{\rho_v L} \right) \frac{R_o^\alpha}{R_o} \left(\frac{\rho}{\Delta P} \right)^{1/2} \quad (2)$$

where:

$$\psi = \frac{2}{(\Delta P) (\Delta T)} \int_{T_\infty}^{T_{SAT}} [P_\infty^* - P_v(T)] dT \quad (3)$$

and

$$\frac{1}{\rho_v} = \frac{1}{\Delta T} \int_{T_\infty}^{T_{SAT}} P_v(T) dT \quad (4)$$

where:

ρ = density of liquid

R_o = initial bubble radius

c = specific heat of liquid

ρ = liquid density

$\Delta T = T_{SAT} - T_\infty$

$\Delta P = P_\infty^* - P_v$

T_{SAT} = saturation temperature at pressure P_∞^*

P_v = vapor pressure corresponding to T_∞

T_∞ = liquid temperature at a large distance from the bubble.

$P_v(T)$ = vapor pressure at temperature T

α = liquid diffusivity

The significance of B_{EFF} is that for $B_{EFF} < 0.05$, heat transfer to the bubble controls the rate of collapse while for $B_{EFF} > 10$, liquid inertia controls the collapse. When $0.05 < B_{EFF} < 10$, either effect controls. For the case of a sodium vapor bubble in the highly subcooled regions of the bond, inertia

controls the collapse and the time required for collapse would be given by

$$t = \frac{0.75 R_o}{\left(\frac{2}{3} \frac{\Delta P}{\rho}\right)^{1/2}} \quad (5)$$

The collapse time would be approximately 1×10^{-3} sec.

If the rapid "explosive" boiling does not occur, the bubble collapse would be heat transfer controlled and the collapse times would be much longer than for inertia controlled collapse.

The sodium boiling investigation will continue in an attempt to determine whether or not unstable boiling will occur.

SECTION 6

UPC-330 - SODIUM VAPOR DEPOSITION STUDIES

R. A. Markley, J. L. McKinney, J. D. Sutherland

1. OBJECTIVE

In several of the processes proposed for sodium bonding of carbide fuel to its cladding, the possibility exists of exposing liquid sodium to glove box atmospheres. As a result, some evaporation of sodium from the molten pool will take place with subsequent deposition on cooler surfaces within the glove box or associated systems. Potential consequences of excessive sodium deposition are loss of visibility through windows, jamming of mechanical parts, short circuiting of electrical apparatus, and plugging of gas systems.

The objective of this task is to determine evaporation rates from an open sodium pool into an argon filled enclosure and subsequent distributions of deposited sodium within the enclosure and argon system. Included is the determination of effects of sodium temperature, sodium pool diameter, and argon flow rate.

2. PRIOR WORK

Analysis was performed to predict sodium evaporation rates from open sodium pools. The results were plotted as curves of evaporation rate versus sodium temperature for various pool diameters. The effects of using nitrogen or helium atmospheres instead of argon were considered, and no significant differences were predicted for the three gases.

To experimentally verify the analysis, a test rig was designed which permits an open pool of sodium to be exposed to an argon atmosphere. Collectors placed in the atmosphere can be withdrawn periodically and examined for sodium buildup. During prior quarters all major components for the test apparatus were received from suppliers and final assembly was started.

3. CURRENT PROGRESS

3.1 Test Apparatus Fabrication

Fabrication of test apparatus was completed. Thermocouples, heaters, and insulation were installed on the sodium piping. Plumbing for the gas system was installed, and all wiring was completed for the electrical heaters and control panel. The test apparatus, just prior to attachment of the piping insulation, is shown in Figure 330-1.

3.2 Pre-operational Tests

Several pre-operational tests were performed before filling the system with sodium. These tests were designed to check-out all systems and components and also to determine the upper limits of sodium temperature which could be used in the experiments.

Before applying heat to the system, the entire system was purged with argon and the gas purification system started. Initial difficulties in obtaining required argon purity levels were experienced as the result of leakage at several fittings in the gas system. These leaks were repaired, and purity levels of 1 to 2 ppm H_2O and less than 1 ppm O_2 were obtained and maintained through subsequent operations.

After shakedown of the argon system, the temperature of the sodium system was progressively raised. In the process, a heater failure was experienced on the dump tank. The failure was caused by a thermocouple head which had cut through the heater insulation and caused a short circuit. Shields were applied to all thermocouples on the dump tank, and a new heater installed. No further problems were experienced while the dump tank was being taken up to its design temperature of 1200°F.

The sodium test-pot temperature was raised to 700°F. At this temperature, the temperature of the flange which joins the pot to the bell jar was determined to be approximately 300°F under static conditions and 250°F when 12 cfm of argon were circulated through the bell jar. These temperatures are considerably below the maximum flange temperature of 400°F which is dictated by the elastomer seals used. As a result, an 800°F sodium temperature during the

Figure 330-1. Sodium Vapor Desposition Test Apparatus

deposition tests seems to be attainable. During these tests a defective sodium line heater was found and subsequently replaced.

3.3 Test Program

During the quarter, test conditions for initial operations were defined. Initial operations will consist of at least five series of tests. The variation of parameters of these tests include surface area of the sodium pool temperature of the sodium pool ranging from 300°F to 800°F, flow rates of cover gas, and length of specimen exposure time.

The first series of test parameters is as follows:

Argon Flow Rate	Static
Sodium Pool Surface Area	28.3 sq. in. (6-in. dia)
Specimen Exposure Time	100 hours at each temperature
Temperature of Pool	(a) 300°F (b) 400°F (c) 600°F (d) 800°F

The second series of test parameters is as follows:

Argon Flow Rate	10 cfm
Sodium Pool Surface Area	28.3 sq. in. (6-in. dia)
Specimen Exposure Time	Determined by visual observation
Temperature of Pool	(a) 400°F (b) 600°F (c) 800°F

The third series of test parameters is as follows:

Argon Flow Rate	1 cfm
Sodium Pool Surface Area	28.3 sq. in. (6-in. dia)
Specimen Exposure Time	Determined by visual observation
Temperature of Pool	(a) 400°F (b) 600°F (c) 800°F

The fourth series of test parameters is as follows:

Argon Flow Rate	Static
Sodium Pool Surface Area	12.5 sq. in. (4-in. dia)
Specimen Exposure Time	100 hrs. at each temperature
Temperature of Pool	(a) 400°F (b) 600°F (c) 800°F

The fifth series of test parameters is as follows:

Argon Flow Rate	Static
Sodium Pool Surface Area	3.1 sq. in. (2-in. dia)
Specimen Exposure Time	100 hrs. at each temperature
Temperature of Pool	(a) 400°F (b) 600°F (c) 800°F

During testing, six sodium vapor collectors will be placed within the bell jar as shown in Figure 330-2. These will be removed at the end of each test run and analyzed for sodium build up.

Figure 330-2. Sodium Collector Locations

SECTION 7

UPC-400 - THERMAL FLUX IRRADIATION STUDIES

A. Biancheria, J. R. F. Hayes, P. J. Levine, E. H. Novendstern,
W. E. Pennell, D. J. Rop, J. A. Rylatt

1. OBJECTIVE

The objective of this task is to evaluate the in-pile performance of sodium-bonded (U,Pu)C fuel elements. In-pile tests will be performed in currently available thermal flux facilities. The post-irradiation test results are intended to provide:

- a. A preliminary performance test of the sodium-bond concept in carbide fuel pins,
- b. a comparison of fuel performance characteristics, such as fission gas release rates and fuel swelling rates, for various modifications of (U,Pu)C fuel,
- c. a comparison of fuel-cladding compatibility at fuel surface temperatures in the range of expected large plant operating conditions,
- d. a substantial volume of data to supplement the fast flux test results, and
- e. early performance information for reassessing the direction of future fast flux tests.

2. PRIOR WORK

Specific test objectives and irradiation exposure parameters for individual pins were defined in conformance with the objectives of the overall irradiation program^[5]. Design and analysis of Phase I fuel pins and capsules were completed^[5,6,7] and the subsequent fabrication and assembly were partially completed. The Phase I instrumentation console was designed, fabricated, and shipped to the reactor site. Purchase orders were approved by the AEC and

placed with General Electric for the Phase I irradiation services, special operating services, and V-RAFTs; fabrication of V-RAFTs was completed. Phase II design and analysis of fuel pins were initiated.

3. CURRENT PROGRESS

3.1 Summary

The fabrication, assembly, and inspection of all Phase I fuel pins and capsules were completed. The capsules were inserted into designated GETR pool positions by the V-RAFTs and brought to power in March. The instrumentation operating procedures were completed. Arrangements for post-irradiation shipping and examination were initiated.

3.2 Fuel Pins

The fabrication, assembly, and inspection of the nine Phase I fuel pins^[8] were completed. A description of the fabrication procedures was prepared and will be incorporated into an overall design and fabrication report on Phase I fuel pins and capsules. This report will contain the detailed inspection and characterization results of each fuel pin. These pins were sent to the GETR in three separate shipments of three each.

3.3 Irradiation Capsules

Assembly and inspection of the three Phase I capsules were completed with relative ease and only minor difficulties. Photographs of the W3G capsule taken during assembly are shown in Figures 400-1 and 400-2. The pins designated to position the thermocouple tips, as shown in Figure 400-1 of WCAP-3791-10, "Uranium Plutonium Carbide Quarterly Progress Report for the Period Ending September 30, 1967", were replaced by 0.005-inch diameter wires tied around the thermocouples to fix their location adjacent to the thermal sleeves (See Figure 400-2). This change necessitated a re-calculation of expected thermocouple readings reported in the Data Package.

Post-assembly inspection of the first capsule indicated a lack of concentricity of the thermal sleeve with the containment tubes at the joint between the Zircaloy and aluminum sleeves. To correct this, a small spacer similar to that

Figure 400-1. Components of Typical Phase I Irradiation Capsule

Figure 400-2. Enlarged View of Typical Capsule Components

used on the lower sleeve was inserted. In order to use a bellows assembly one inch longer than planned in the secondary containment, the outer containment tube was lengthened. All of these changes are being incorporated in the as-built drawings, which will be supplied to WARD in the next quarter.

3.4 Instrumentation

The wiring diagrams and operating procedures for the Phase I instrumentation console were completed and sent to GETR. An additional contact closure was added to the circuit by GE to provide a signal at the GETR supervisory panel in the event of a main power failure to the console. After a final checkout and calibration, the instrumentation was moved to the third floor of the GETR facility where it was installed for the irradiation.

3.5 Design

Stress analyses were initiated for three types of potential failure modes; thermal ratchet, low cycle fatigue, and creep rupture. The first two modes were evaluated and reported in the previous quarterly report. A preliminary analysis of creep rupture was conducted during this quarter using the same ground rules as previously reported.

The preliminary creep rupture analysis was insufficient to determine if this is the controlling failure mechanism. However, if this is the case, one way to avoid the possibility of failure is to increase the fission gas space in the fuel pin, which would reduce the internal gas pressure. This can be accomplished either by decreasing the fuel column height or by increasing the fuel pin length. The former would reduce the L/D ratio of the column to less than ten, which is undesirable from heat transfer considerations. Increasing the bottom fuel pin lengths would raise the top pins and result in low undesirable power ratings and subsequent elimination of these pins. However, initial results from Phase I indicate that the top pins in Capsules W1G, W2G, and W3G are operating at powers below their design targets, making the elimination of top pins from future capsules imminent.

An alternate method of preventing potential failure of the Phase I fuel pin design at 100,000 MWD/T burnup is to reduce the operating temperature of the cladding. This would produce undesirably low conditions for the study of in-pile fuel-clad compatibility. Resolution of a design for Phase II fuel pins was postponed early in the quarter as a result of reduced funding of this program.

3.6 Test Site Activities

After initial positioning and calibration of the Phase I V-RAFTs, capsules W1G and W2G were inserted into GETR pool positions X2 and X3, respectively, and initially brought to power midway in Cycle 94 (March 10). Capsule W3G was inserted in GETR pool position X11 and brought to power at the start of Cycle 95 (March 31). Initial power ratings, determined from the instrumentation readings, are slightly lower than the anticipated design ratings. The characterization of this deviation is in progress.

3.7 Fuel Pin Shipment

Preparation for post-irradiation shipment of the Phase I capsules was initiated. Cask and licensing requirements are being investigated so that the necessary arrangements for shipping can be made and the available equipment ready on time.

3.8 Post-Irradiation Examination

Expanded hot cell capabilities for handling high-purity sodium and plutonium at the Westinghouse Waltz Mill Site will not be available for Phase I post-irradiation examination. Therefore, a detailed plan of the proposed examination was started to provide a basis for a request for a proposal to perform the examination. Preliminary cost estimates for neutron radiography and post-irradiation examination have been received for incorporation into the Fiscal Year 1969 Work Program.

Three designs of a diametral-measuring and recording instrument for fuel pins have been reviewed. Literature describing the components of these designs has been obtained for evaluation. This effort is temporarily suspended until a subcontract for post-irradiation work is prepared.

SECTION 8

UPC-500 - FAST FLUX IRRADIATION STUDIES

A. Biancheria, B. L. Harbourn, R. G. Cockrell

1. OBJECTIVE

The objective of this task is to evaluate the performance of sodium-bonded (U,Pu)C fuel pins in fast flux. Various types of modified and unmodified fuel will be tested to study the effects of composition variations with the objective of defining satisfactory carbide fuel for LMFBR application.

2. PRIOR WORK

Design of eleven Phase I capsules to operate to 50,000 MWD/T at 25 to 30 kw/ft was completed, and a first draft of the design and hazards report was prepared. Pin and capsule fabrication development was begun.

The objectives of Phase II tests were defined, and a parametric study to support the design was initiated.

3. CURRENT PROGRESS

3.1 Summary

The Phase I design and hazards report for the first series of test capsules was completed for submittal to Argonne National Laboratory (ANL) for their approval.

Phase II design studies have shown the need for a high performance sub-assembly, probably containing 12 capsules, to attain ratings above 30 kw/ft. To keep the WARD program consistent with current national policy on carbide fuel development, all irradiation programs are being revised to provide base technology data rather than engineering design data. The review will be completed next quarter.

3.2 Phase II Pin Design

The target conditions established for Phase II tests were fuel center temperatures in the range of 1100-1550°C (2000-2800°F) and burnups to 100,000 MWD/T. Pins are to be sodium-bonded with the fuel-clad gap sized to accommodate fuel swelling without any mechanical interaction.

A parametric study was undertaken to study the maximum attainable pin performance in EBR-II. The study aimed to establish the maximum performance capability of Cr-modified fuel pins when placed in a standard 19-capsule subassembly in EBR-II. An evaluation was made of the maximum fuel center temperature versus burnup for a subassembly placed in grid position 1N1, which is the peak flux position. The results, plotted in Figure 500-1, are based on the following set of assumptions:

- a. 89 percent T.D. Cr-modified (U,Pu)C fuel
- b. 0.012-inch cladding wall thickness
- c. EBR-II power of 55 MWt*
- d. coolant flow rate through the subassembly of 65 gpm
- e. fuel swelling based on reported values of (U,Pu)C⁽⁹⁾

On the basis of these results it was clear that even in the hottest position in EBR-II, the desired performance could not be attained with a standard 19-pin subassembly.

Target irradiation conditions for this pin concept could be attained by increasing EBR-II power or by increasing the fuel diameter; however, the required power increase cannot be attained, and the required fuel diameter cannot be accommodated in a 19-capsule subassembly. Therefore, a parametric study of pin performance in alternate subassembly configurations was initiated.

A preliminary evaluation of the maximum burnup and the maximum centerline temperature versus linear power rating was made for subassembly configurations having 7-, 12-, 13-, and 19-capsules. The results, plotted in Figures 500-2

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* Discussions with ANL Idaho personnel suggest that 50 MWt may be a more realistic assumption for experiments during 1968 and early 1969.

Figure 500-1. Performance Parameters for a Cr-Modified Fuel Pin in a 19-Pin Subassembly Located in the Peak Flux Position in EBR-II). (EBR-II Power = 55 MWt)

Figure 500-2. Power Rating (q') versus Maximum Burnup for Several EBR-II Subassembly Configurations

and 500-3, are based on the following assumptions:

- a. EBR-II power of 50 MWt
- b. EBR-II grid position is 3N2
- c. fuel specifications:
  - 1) 89 percent T.D. Cr-modified (U,Pu)C
  - 2) composition given in the design and hazards report
  - 3) swelling data as previously described
  - 4) thermal conductivity of 9.25 Btu/hr-ft-°F
- d. unrestrained swelling with 0.001-inch diametral gap at end-of-life
- e. rod O.D. and cladding wall thickness as given in Table 500-1
- f. thermal conductivity:
  - 1) sodium bond: 34 Btu/hr-ft-°F
  - 2) cladding: 12.2 Btu/hr-ft-°F
- g. cladding surface temperature of 1025°F
- h. thermal expansion of fuel and cladding neglected
- i. nominal values, i.e. no uncertainty factors

Assumptions f and g are based on the results from the thermal analysis of several 19-capsule configurations. The pin diameters given in Table 500-1 were determined by requiring that each of the configurations fit within a hexagonal can that has the same across-flats dimension as a 19-capsule subassembly. The values for the cladding wall thickness were estimated.

Figures 500-2 and 500-3 indicate that the 7-, 12-, and 13-capsule subassemblies permit linear power ratings above 40 kw/ft with a burnup of at least 100,000 MWD/T; however, the capability of the 13-capsule subassembly to give this performance is marginal, and the maximum centerline temperature at 40 kw/ft is only approximately 2400°F. On the other hand, the 7-capsule subassembly is capable of attaining fuel temperatures and burnups much higher than the desired values. Therefore, it is unlikely that the full potential of a 7-rod subassembly would ever be needed for this program. The 12-rod subassembly, therefore, appears to be the logical choice since it permits the desired performance with a minimum reduction in rod positions per assembly.

Figure 500-3. Power Rating (q') versus Fuel Centerline Temperature for Several EBR-II Subassembly Configurations

TABLE 500-1		
Rod O.D. and Cladding Wall Thickness		
Subassembly Configuration	Pin Diameter (in)	Cladding Wall Thickness (in)
7-pin	0.610	0.024
12-pin	0.396	0.016
13-pin	0.356	0.014
19-pin	0.300	0.012

A preliminary layout of a 12-capsule subassembly is shown in Figure 500-4. The dimensions shown for the shroud tube and capsule are extrapolated directly from the dimensions given for a 19-pin subassembly described in the EBR-II user's guide and are subject to change following detailed thermal, fluid, and mechanical analyses. Note that the 12-pin subassembly has three open spaces that could be used for the irradiation of structural materials samples.

3.3 Phase I Fabrication

Techniques for the fabrication of Mk 1B capsules (for Phase I tests in a 19-capsule subassembly) have been successfully developed. Fuel for the first test pins is being manufactured.

3.4 Design and Hazards Analysis

A design and hazards report for the first series of test capsules has been completed for submittal to ANL. The Phase I test objectives are currently being revised, and the report will require some detailed modification which will be presented as an addendum.

3.5 Review of Test Objectives

The original primary objective of this program was to define engineering data on a selected pin concept required for an early LMFBR design. National LMFBR policy dictates a shift towards a base technology program on carbide fuels with

Figure 500-4. 12-Pin Subassembly for EBR-II

the objective of a critical evaluation of carbide technology in 1971. Changes in the programs under this contract to meet this new objective are being studied. Modified programs will be proposed in the FY 1969 Work Program early in the next period.

SECTION 9

UPC 600 - TRANSIENT IRRADIATION STUDIES

A. Biancheria, R. G. Cockrell, D. Doiron, B. L. Harbourne
J. Jedruch, P. Persson, J. Rylatt

1. OBJECTIVE

The objective of this task is to evaluate the inherent safety characteristics of sodium-bonded (U,Pu)C fuel pins under reactor transient conditions. To define these characteristics it is tentatively planned to irradiate 16 fuel pins in the TREAT reactor at the National Reactor Testing Site in Idaho. Varying transient irradiation conditions will be used to determine the limiting conditions, mechanisms, and consequences of fuel pin failure.

2. PRIOR WORK

A preliminary plan was developed with three phases of work consisting of unirradiated and irradiated fuel pins. Preliminary nuclear and thermal analyses were started on a model capsule to determine the effect of reactor pulse size on the performance of shielded and unshielded fuel pins during irradiation in TREAT. The results of these analyses are summarized below:

- a. Fully enriched, unshielded fuel pins have a significant flux design depression from fuel surface to fuel center resulting in a fuel pin power density and temperature profile significantly different from what would be expected in a fast reactor.
- b. Shielding the fuel pin from the TREAT thermal neutrons flux will give the desired fuel pin power density and temperature profile. A study on the type, thickness, and performance of materials that will effectively shield the fuel pin was begun. Results indicate that a B_4C shield 1.6 centimeters thick would be satisfactory.

3. CURRENT PROGRESS

Work on this task was stopped after January of this quarter, being postponed until FY 1969. Summarized in this section is the progress made during the third quarter of FY 1968 and the status of the task to date.

3.1 Status of the Experimental Program

The preliminary plan summarized in the last quarterly report will be re-evaluated. This review will include a thorough investigation of each test, its objective, and the methods of achieving these objectives.

The review and final plan of the program will be defined in the FY 1969 Work Program.

3.2 Selection of Pulse Height and Width

Prior analysis on a model capsule has shown that a shielded fuel pin will have the desired temperature profile if it is transient tested using a narrow pulse. As discussed in the last quarterly report, the pulse height and total energy released vary inversely with pulse width. Therefore, if a narrow pulse is used, the magnitude of the total energy would produce undesirably high peak temperatures in the fuel pin. A very effective method of reducing the total energy of a transient excursion is to clip the pulse. In a clipped pulse transient, the reactor is scrammed at some intermediate time after initiation of the pulse. A thermal and nuclear analysis will be performed during FY 1969 to evaluate the effects of a clipped pulse transient on the performance of the fuel pin.

3.3 Comparison of Thermal Neutron Absorber Materials

A nuclear analysis was made comparing B_4C and B_6Si as thermal neutron absorbers in the fuel pin shield. Cadmium was not considered because of its comparatively low melting point. The nuclear analysis was conducted employing transport theory calculations to determine the power distribution in the fuel pin and in the TREAT core. In these calculations, an absorber thickness of 1.079-centimeters was assumed. The B_6Si absorber was assumed to be laminated of successive layers of B_6Si and stainless steel, conforming with the present methods of construction being studied by ANL for neutron absorber shields.

The thickness of the successive layers assumed were:

<u>Material</u>	<u>Thickness (cm)</u>
B_6Si	0.125
B_6Si + Stainless Steel	0.008
Stainless Steel	0.008

The B_4C absorber was assumed to be a compacted powder, with a density of 70 percent of the theoretical density. The following table illustrates the nuclear absorption capability of each material based on the above assumptions:

	B_4C	B_6Si
$\frac{\text{POWER GENERATED AT PELLET SURFACE}}{\text{POWER GENERATED AT PELLET CENTER}} =$	1.06	1.07
$\frac{\text{PEAK POWER IN TEST SPECIMEN}}{\text{PEAK POWER IN TREAT}} =$	25	29

These values indicate that B_4C and B_6Si are comparable in their ability to flatten the power distribution in the pellet and in reducing the power generated in the fuel pin. Therefore, the final selection of the neutron absorber material to be used in the fuel pin shield will be based on the fuel pin capsule design and the relative ease of shield fabrication.

SECTION 10

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