AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

1 of 1

DECLASSIFIED

HW-58205

AEC RESEARCH AND DEVELOPMENT REPORT

COPY No. 22

QUARTERLY PROGRESS REPORT FUELS DEVELOPMENT OPERATION

JULY, AUGUST, SEPTEMBER 1958

OCTOBER 15, 1958

HANFORD LABORATORIES

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

GENERAL ELECTRIC

THIS DOCUMENT CONTAINS INFORMATION OF A CONFIDENTIAL NATURE AND IS TO BE KEPT SECRET. IT IS TO BE DISTRIBUTED ONLY TO PERSONS AUTHORIZED BY THE AEC. IT IS TO BE DESTROYED WHEN NO LONGER REQUIRED. IT IS TO BE KEPT IN A SECURE PLACE AND NOT TO BE LOANED TO ANY OTHER PERSON. IT IS TO BE RETURNED TO THE AEC WHEN NO LONGER REQUIRED.

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DECLASSIFIED

DECLASSIFIED

HW-58205

This document classified by
S. H. Bush

C-25, Metallurgy and Ceramics
(M-3679, 22nd Ed.)

This document consists
of 103 pages. Copy No. 22
of 241 copies. Series

Classification Cancelled and Changed To

DECLASSIFIED

By Authority of A S Lewis QUARTERLY PROGRESS REPORT
12-16-93 C G-PR-2 FUELS DEVELOPMENT OPERATION
By Q E Sawley 1-4-94 (PR-24) JULY, AUGUST, SEPTEMBER, 1958
Verified By DK + Janson 1-4-94

By

The Staff of Fuels Development Operation

J. J. Cadwell

Manager

J. C. Tobin

Supervisor, Physical Metallurgy

J. E. Minor

Supervisor, Fuel Element Design

E. A. Evans

Supervisor, Ceramic Fuels
Development

S. H. Bush

Supervisor, Fuels Fabrication
Development

October 15, 1958

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

Work performed under Contract No. W-31-109-Eng-52 between
the Atomic Energy Commission and General Electric Company

This document contains Restricted Data as defined in the Atomic
Energy Act of 1954. Its transmittal or the disclosure of its
contents in any manner to an unauthorized person is prohibited.

WARNING - PRELIMINARY REPORT

This report contains information of a preliminary nature and is prepared
primarily for the use of Hanford Atomic Products Operation personnel.
It is subject to revision on further checking or collection of additional data.

Route To:	P. R. No.	Location	Route Date	Signature and Date
<u>L. A. Hartcorn</u>	<u>1824</u>	<u>326</u>	<u>SEP 24 1958</u>	<u>L. A. H</u>
<u>L. A. Hartcorn</u>	<u>1824</u>	<u>326</u>	<u>SEP 15 1958</u>	

Printed in USA. Charge \$2.20. Available from the U.S. Atomic Energy
Commission, Technical Information Extension, P.O. Box 1001,
Oak Ridge, Tennessee. Please direct to the same address inquiries
covering the procurement of other classified AEC reports.

DECLASSIFIED

Ex

UNCLASSIFIED

- 1.2 -

HW-58205

C-25, Metallurgy and Ceramics
(M-3679, 22nd Ed.)

INTERNAL DISTRIBUTION

Copy Number

1	F. W. Albaugh - H. M. Parker
2	A. G. Blasewitz
3	C. W. Botsford
4	L. P. Bupp
5 - 6	S. H. Bush
7	J. J. Cadwell
8	A. B. Carson
9	J. L. Carter
10	V. R. Cooper
11	R. L. Dickeman
12	E. A. Eschbach
13	E. A. Evans
14	J. E. Faulkner
15	R. M. Fryar
16	G. T. Geering - R. G. Wheeler
17	L. L. German - H. F. Zuhr
18	S. M. Gill
19	J. W. Goffard - T. T. Claudson
20	Q. H. Greager
21	R. E. Hall
22	L. A. Hartcorn
23	W. M. Harty
24	D. C. Kaulitz - F. E. Young
25	W. S. Kelly
26	W. M. Mathis
27	L. H. McEwen - S. Goldsmith
28	K. R. Merckx - J. W. Weber
29	J. E. Minor
30	J. F. Music
31	R. S. Paul
32	H. J. Pessl
33	W. H. Reas
34	J. W. Riches
35	O. C. Schroeder
36	H. P. Shaw
37	E. A. Smith
38	K. V. Stave
39	J. T. Stringer
40	J. W. Talbott

UNCLASSIFIED

UNCLASSIFIED

-1.4-

HW-58205

C-25, Metallurgy and Ceramics
(M-3679, 22nd Ed.)

EXTERNAL DISTRIBUTION (Contd.)

Copy Number

104	Bureau of Ordnance
105	Bureau of Ships
106	Canoga Park Area Office
107 - 109	Chicago Operations Office
110	Chicago Patent Group
111 - 112	Combustion Engineering, Inc.
113	Curtiss-Wright Corporation
114	Division of International Affairs (Pennington)
115	Dow Chemical Company (Rocky Flats)
116 - 118	duPont Company, Aiken
119	duPont Company, Wilmington
120	Engineer Research and Development Laboratories
121	Fluor Corporation
122	Franklin Institute of Pennsylvania
123	General Atomic Division
124 - 126	General Electric Company (ANPD)
127	GE Company, San Jose (AT(30-3)-502)
128	General Nuclear Engineering Corporation
129	Hanford Operations Office
130	Iowa State College
131 - 134	Knolls Atomic Power Laboratory
135	Lockland Aircraft Reactors Operations Office
136 - 138	Los Alamos Scientific Laboratory
139	M and C Nuclear, Inc.
140 - 145	Mallinckrodt Chemical Works
146	Marquardt Aircraft Company
147	Martin Company
148	Mound Laboratory
149	National Advisory Committee for Aeronautics, Cleveland
150	National Bureau of Standards
151 - 152	National Lead Company of Ohio
153	Naval Air Development Center
154	Naval Air Material Center
155	Naval Air Turbine Test Station
156	Naval Research Laboratory
157	New Brunswick Area Office
158	New York Operations Office
159	Nuclear Development Corporation of America
160	Nuclear Metals, Inc.

UNCLASSIFIED

DECLASSIFIED

-1.6-

HW-58205

The present Quarterly Report is the continuation of a series issued by the new Fuels Development Operation. Reports in this series combine portions of the quarterly reports by the former Metallurgy Research and Fuel Technology Sub-Sections. Work reported includes research conducted by the Physical Metallurgy Operation, and research and development conducted by Fuel Design, Fuels Fabrication Development, and Ceramic Fuels Development Operations. Studies formerly reported by the Radio-metallurgy, Metallography, and Welding and Corrosion Units, in addition to portions of the Fuels Technology work, are reported elsewhere.

Reports previously issued as Metallurgy Research and Fuel Technology Quarterlies are listed in the first report of the present series, HW-46091.

QUARTERLY PROGRESS REPORTS - FUELS DEVELOPMENT

Reports previously issued are as follows:

1. HW-46091 Quarterly Progress Report - Fuels Development Operation - July, August, September, 1956
2. HW-47850 Quarterly Progress Report - Fuels Development Operation - October, November, December, 1956
3. HW-49803 Quarterly Progress Report - Fuels Development Operation - January, February, March, 1957
4. HW-51704 Quarterly Progress Report - Fuels Development Operation - April, May, June, 1957
5. HW-53488 Quarterly Progress Report - Fuels Development Operation - July, August, September, 1957
6. HW-54702 Quarterly Progress Report - Fuels Development Operation - October, November, December, 1957
7. HW-56029 Quarterly Progress Report - Fuels Development Operation - January, February, March, 1958
8. HW-57146 Quarterly Progress Report - Fuels Development Operation - April, May, June, 1958

T

DECLASSIFIED

-1.8-

HW-58205

TABLE OF CONTENTS (contd.)

	<u>Page</u>
Restraint of Uranium Swelling by Zirconium Cladding -	
J. W. Weber	4.6
Coextruded Zr-2 Clad Uranium Fuel - J. W. Goffard	4.7
Uranium-Magnesium Matrix Fuel Material - J. W. Goffard	4.9
Aluminum Clad Fuel Elements for High Temperature Service -	
R. G. Wheeler and J. W. Goffard	4.10
FABRICATION DEVELOPMENT	4.10
Effect of Alpha Phase Treatment of Cold Closure Fuel Elements -	
R. S. Kemper	4.10
Zircaloy-2 Uranium Coextruded Rod Closures - T. T. Claudson	4.11
Closure of Coextruded Rod Fuel Element End Caps by Brazing -	
W. I. Steinkamp and W. L. Wyman	4.11
Self Supported Aluminum Jacketed Fuel Elements - W. L. Wyman	
and W. I. Steinkamp	4.12
Direct Casting of Uranium into Zircaloy Cans - J. C. Tverberg	
and R. S. Kemper	4.16
Evaluation of Cast I and E Elements - R. S. Kemper	4.17
Al-Si Bond Quality Studies - E. A. Smith	4.17
Swaging - F. B. Quinlan	4.24
FUEL EVALUATION	4.24
Seven-Rod Cluster Fuel Elements - T. T. Claudson	4.24
Four-Rod Cluster Fuel Elements - G. T. Geering and	
T. T. Claudson	4.25
Rod and Tube Fuel Element - G. T. Geering	4.27
Co-Axial Fuel Element - J. W. Goffard	4.27
Evaluation of Vacuum-Canned and Hot-Pressed Slugs -	
E. A. Smith	4.36
REFERENCES	4.37
CERAMIC FUELS DEVELOPMENT.	5.1
FUEL FABRICATION	5.1
UO ₂ Powder for Swaging - D. W. Brite and R. J. Anicetti,	5.1
UO ₂ Fuel Cores for SRE Irradiation - D. W. Brite and	
R. J. Anicetti	5.1
Zircaloy-2 Clad Swaged UO ₂ Fuel Rods for VBWR Irradiation -	
M. K. Millhollen and D. R. Stenquist	5.2
Swaging of UO ₂ - D. R. Stenquist and R. J. Anicetti	5.2
Internally Cooled Only Fuel Element - M. K. Millhollen	5.9

SUMMARYPHYSICAL METALLURGYMECHANICAL AND PHYSICAL PROPERTIES OF IRRADIATED MATERIALS

It has been found that at low irradiation temperatures appreciable hardness increase is observed in normal uranium after an irradiation equivalent to a single spike cycle. In these extremely low exposure ranges it is apparent that damage results from energy release and atomic displacements rather than fission product buildup. Annealing of the induced hardening appears to be complex and no single recovery process has yet been definitely identified.

In the irradiation of thorium tensile specimens in the MTR for longer than ten operation cycles at flux densities greater than 1×10^{14} nv, the atom per cent burnup is about 20 per cent higher than predicted on the basis of reactor operating time due to continuous U-233 buildup. Since uncertainties in flux values and resonance capture could also be in error by this amount, the adjustment for this difference in burnup does not appear to be necessary in preparation of thorium tensile specimen irradiations.

An investigation of the effect of strain rate on the yield strengths of irradiated Zircaloy-2 and Zircaloy-3 indicates that the irradiated materials are less sensitive to strain rate than the unirradiated. Irradiated Zircaloy-2 showed a considerable increase in yield strength and a marked drop in load yield point after irradiation. Zircaloy-3 yield strength was increased by about the same amount after irradiation but no yield point effect was observed. The reasons for the difference in radiation effects in these two alloys is not immediately obvious.

Zircaloy-2 tensile specimens are prepared and ready for charging into the ETR loop facility. These will be irradiated in contact with hot water in the loop.

Scope and design studies for an in-reactor testing instrument facility have been completed. A capsule designed for in-reactor creep measurements was bench tested successfully for over 100 hours. Several other capsule designs are in the initial stages of development.

MECHANICAL AND PHYSICAL PROPERTIES OF UNIRRADIATED MATERIALS

No new information has been obtained this quarter.

DECLASSIFIED

-2. 3-

HW-58205

A study is progressing of the effect on bond strength and defect behavior of the heat treatment of coextruded Zircaloy-2 clad uranium rods. Initial results indicate that the "as-extruded" Zr-U bond is tenacious, and is embrittled by improper beta heat treatment of the uranium core. Uniformly side-defected "as-extruded" fuel rods behave consistently during the initial stages of failure in 300 C water in both dynamic loop and static autoclave tests. Beta heat treatment of the uranium followed by air cooling raises the shear strength of the bond about 30 per cent, while beta heat treatment followed by an isothermal treatment near 600 C restores the shear bond strength to near that of the "as-extruded" condition.

A matrix fuel consisting of particulate uranium in a matrix of magnesium has been under investigation as a possible high burnup fuel. The latest data are taken from a 1.00-inch diameter by 4-inch long element clad in 0.050-inch Zircaloy-2. This element, containing 65 v/o uranium, underwent an estimated volume increase of 0.5 per cent while being irradiated to 9900 MWD/T. Volume increase was estimated from post-irradiation diameter measurements.

Solid state reaction of aluminum with uranium is a possible service limit in the use of aluminum jackets on metallic uranium elements in high temperature coolants. Temperature limits on some diffusion barriers are being studied. The results of a 3500-hour test in 370 C recirculating organic coolant has given some measure of the effectiveness of AlSi, hot pressed nickel, and colloidal graphite as diffusion barriers in this system.

FABRICATION DEVELOPMENT

The fuel core of cold closure elements is slightly deformed during the closure operation. Subsequent bonding methods require the element to be heated to 500-600 C and it is necessary to know if residual strain and heating cause adverse changes in grain structure and properties. Sections from such elements were heated for 10, 20, and 60 minutes at 500 and 600 C. Hardness of the uranium decreased by five points Rockwell G; no excessive grain growth occurred. No adverse changes were noted.

The possibility of the use of a braze in forming closures on coextruded rod fuel elements is being explored. Zr-Be-Cu alloys studied have not produced a satisfactory braze. Brazing with fine silver is now under study.

Self-supported fuel elements designed to permit accurate positioning in smooth-bore process tubes are now under study for use in Hanford reactors. A production test IP-84-A is now being irradiated in B reactor, and an expansion of the test to include irradiation of 13 tubes of I and E elements up to 900 MWD/T is proposed. The equipment and process for attaching the supports to such elements is described.

DECLASSIFIED

- 2. 5 -

HW-58205

[REDACTED]

exposure, and remained steady thereafter. Apparently cyclic operation from low to high power did not affect the operating characteristics of the element. A companion element without provision for temperature monitoring was also irradiated during this period, and remains at the MTR for further irradiation.

Post-irradiation evaluation of vacuum-canned, hot-press canned, and lead-dip canned I and E fuel elements has been completed. Their behavior was studied comparatively on the basis of quality loss suffered by the elements as they were irradiated to progressively increasing exposures, up to 800 MWD/T. No failures were incurred in any of the elements, and none of the three groups appeared outstandingly superior or inferior with respect to ability to withstand reactor exposure.

CERAMIC FUELS DEVELOPMENT

FUEL FABRICATION

Enriched (8 per cent U-235) UO_2 fuel cores were fabricated for the first of two six-foot tubular fuel elements to be irradiated in the SRE as part of a joint HAPO-Atoms International program. The UO_2 fuel core for the second element is being modified so that it can be instrumented with six thermocouples extending axially three feet into the fuel core.

Nine fuel rods consisting of UO_2 compacted in Zircaloy-2 cladding by swaging were fabricated for irradiation in the VBWR. The rods were approximately 30 inches long and 0.57-inch OD. The UO_2 core, containing 2.43 w/o U-235, was compacted to 87.5 ± 0.5 per cent of the theoretical density.

Swage compaction studies indicate that UO_2 densities can be increased in most cases by controlling particle size distribution. The effect of various particle size distributions on swaged density was determined. Highest densities were obtained with UO_2 consisting of a single size fraction of relatively large particles. The UO_2 density also was increased by using cotton, rather than rubber plugs, to contain the UO_2 during swaging.

The feasibility of fabricating a proposed internally-cooled-only fuel element was indicated by the production of a section of an element cooled by six tubes internally spaced in a sintered UO_2 core.

FUEL TESTING

Four irradiated fuel elements consisting of loose-packed, non-sintered, enriched UO_2 powder are being examined. Preliminary data indicate that the elements operated with molten UO_2 cores which, in each case, redistributed to one end of the element during irradiation.

[REDACTED]

PHYSICAL METALLURGYMECHANICAL AND PHYSICAL PROPERTIES OF IRRADIATED MATERIALS

Numerous experiments have been reported indicating that the structure sensitive properties of metals are significantly altered by reactor irradiation. In order to design advanced fuel elements and select materials for future reactors two broad programs have been established. The first program is to determine the effects of irradiation on the mechanical and physical properties of uranium. To aid in interpreting the damage to fuel systems, irradiations of thorium and thorium-uranium alloys enriched in U-235 are also being conducted. The second program involves the study of neutron damage to non-fissionable metals. Determinations of mechanical and physical properties of irradiated metals and alloys are being made as part of this program.

Effect of Irradiation on Uranium - A. L. Bement, R. L. Hales, J. M. Tobin, and J. C. Tobin

Hanford Irradiation - PT-3NA. The purpose of this test is to determine the mechanical and physical properties of beta-heat treated uranium irradiated to 0.018, 0.031, 0.075, and 0.010 a/o burnup. Tensile tests have been performed at room temperature and elevated temperatures on as-irradiated specimens, and at room temperature on specimens annealed after irradiation. The final phase of this program, which involves cyclic annealing, has been started. From this series of tests the effects of phase transformations on the removal of irradiation-induced damage in uranium will be determined. Furthermore, the effects of cyclic annealing in the alpha phase through the temperature range where self-diffusion occurs will be evaluated.

The alpha-beta temperature cycling is between 400 and 700 C, the alpha-gamma temperature cycling is between 400 and 800 C, and the alpha-phase temperature cycling is between 100 and 625 C. The specimens are being held two hours at each terminal temperature, and the rate of heating and cooling is 10 C per hour.

Thermal Conductivity of Irradiated Uranium (BMI Assistance to HAPO). There is some concern that an appreciable decrease in uranium thermal conductivity might occur under reactor operating conditions. Such a decrease might increase uranium core temperatures to an unsafe level. The two available information sources concerning the effect of neutron irradiation on the thermal conductivity of uranium are at best accurate to ± 20 per cent. In an attempt to gain additional information that is more precise than previously reported work, a production test was designed to measure thermal conductivity of uranium before and after irradiation. Nine NaK filled capsules were assembled at Hanford and shipped to Battelle Memorial Institute

TABLE 3.1

AVERAGE HARDNESS DATA* OF URANIUM SPECIMENS
IRRADIATED AT LOW TEMPERATURES TO LOW EXPOSURES

<u>Spec. No.</u>	<u>Time, Irrad. and Exposure</u>	<u>As-Irrad. Hardness (R_A)</u>	<u>% Increase</u>	<u>Anneal</u>	<u>As-Anneal Hardness (R_A)</u>	<u>% Recovery</u>
Control (5 Spec.)	0	48.7				
4-1	22M	54.3				
4-2	or	53.4				
5-1	3 x 10 ¹⁵ nvt	54.1				
6-1		50.7		37d Room Temp.	51.2	--
5-2		--		10h - 200 C	52.0	25**
6-2		--		10h - 300 C	50.9	48**
	Avg.	53.1	9.0			
1-1	6h	55.4				
1-3	or	56.4				
4-4	5 x 10 ¹⁶ nvt	55.6				
1-4		--		1h - 300 C	55.3	7**
4-3		--		2h - 300 C	52.0	37**
1-2		--		10h - 300	50.8	60**
	Avg.	55.9	14.7			
1-5	30h	56.5				
5-3	or	57.1				
6-4	2.5 x 10 ¹⁷ nvt	56.6				
1-6		55.8		10h - 300 C	51.4	50
5-4		55.3		10h - 400 C	50.4	55
6-3		57.2		25h - 400 C	51.9	58.
	Avg.	56.4	15.9			
4-5	84h	57.3				
5-5	or	57.2				
6-6	7 x 10 ¹⁷ nvt	58.3				
4-6		57.9		10h - 300 C	54.7	30
5-6		57.7		10h - 400 C	50.4	69
6-5		57.7		25h - 400 C	52.3	51
	Avg.	57.7	18.4			

* Six readings were taken on each shoulder or twelve per specimen. Each value listed represents the average of these twelve readings.

** The per cent recovery is based on the average as-irradiated hardness for the series.

FIGURE 3.1

The Hardness of Uranium as a Function of Exposure Time at about 20 C

FIGURE 3.2
Incremental Increase in Hardness as a Function of Spike Cycle

- (2) Significance can be placed on the increase in hardness at 25 hours compared to 10 hours at 400 C for the 84-hour exposures.
- (3) The effect of fission products is responsible for the decrease in recovery with exposure time following a 10-hour anneal at 300 C.

The residual hardness remaining after annealing can be attributed to either incomplete recovery of the lattice damage caused by thermal spikes or to the intrinsic effect of foreign atoms in the lattice. However, considering the movement and conglomeration of defects to relatively stable forms, it appears doubtful that an arbitrary temperature limit below the recrystallization range will be found above which lattice damage due to thermal spikes will not be a major factor.

Asymmetrical Flux Effects. A turbo-mechanism has been developed that will rotate an irradiation capsule during exposure in the MTR. Rotation of the capsule will give the samples a radially uniform exposure to the neutron flux. Comparative irradiation tests using both rotating and stationary capsules will be performed to measure the effect of a thermal neutron flux asymmetry on the warping of flat uranium tensile specimens. It should be possible to determine the ability of a rotating capsule to reduce warping attributed to non-uniform exposure of the specimens.

The capsules and turbine components for the test have been fabricated, and tensile specimens have been machined out of normal and enriched uranium stock. The arrangement of the specimens and the capsule design were presented in the preceding report.

Samples from the specimen stock were examined by X-ray diffraction for the presence of residual rolling textures or any preferred orientations resulting from heat treating. The results of this examination indicate that from a structural standpoint the specimens should be dimensionally stable during exposure to a neutron flux.

The specific power generated and, hence, the maximum temperature developed in the normal and enriched uranium specimens during irradiation will be equal. This will be accomplished by equilibrating the reaction rates according to the following relationship:

$$\Sigma_n \Phi_n = \Sigma_e \Phi_e;$$

where Σ = effective, macroscopic, fission cross section,
 Φ = thermal neutron flux density, and subscripts "e" and "n" refer to
 and "enriched" and "normal" uranium respectively.

Assuming that the differences in density and atomic number between uranium containing 0.7 and 1.732 per cent U-235 can be neglected, the ratio of selected flux densities can be approximated as follows:

$$\frac{\Phi_e}{\Phi_n} = \frac{\Sigma_n}{\Sigma_e} = \frac{F_n}{F_e} = \frac{.007}{.01732} = 0.404$$

where F = fraction of U-235 atoms in the specimen.

A squarewave operating cycle at 40 megawatts power was assumed. The operation time per cycle was averaged over five MTR cycles to give an average full-power period of 1.27×10^6 seconds and an average no-power period of 0.46×10^6 seconds. The atom-per cent burnup that occurs in thorium during several cycles of operation is plotted as a function of flux density in Figure 3.3. The solid curve represents the condition where the production of U-233 during "down time" is accounted for. The dotted curve represents the condition where only the power period is included in the burnup calculation.

For irradiations longer than 10 cycles and flux densities greater than 1×10^{14} nv, the atom-per cent burnup is about 20 per cent higher when the down-time production of U-233 is taken into account. Since the uncertainties in the determination of flux densities or resonance capture could also be in error by this amount, the adjustment for this difference in burnup is unnecessary.

Effect of Irradiation on Thorium-Uranium Alloys - A. L. Bement

Alloys of thorium containing 1 w/o or alloy (93 per cent U-235), 4 w/o or alloy, and 5.4 w/o or alloy have been obtained for irradiation tests to determine changes in mechanical properties and some physical properties due to burnup. Irradiations to burnups overlapping and exceeding those normally attained in uranium are being scheduled. Tensile specimens and samples to be used for density, hardness, and metallography will be irradiated.

Test specimens have been machined from the 4.0 and 1.0 per cent or alloy alloys. Flow adaptors to permit individual charging of the under-sized irradiation capsules in an MTR X-basket have been designed and fabricated. Spacers to internally align the test specimens coaxial with the capsule have been fabricated. Capsule assembly and NaK filling for heat transfer will proceed following the heat treating of the specimens.

Effects of Irradiation on Non-Fissionable Materials - A. L. Bement,

R. L. Hales, and J. C. Tobin

Neutron Damage to Metals. The purpose of this program is to advance the theory of irradiation damage to metals by neutrons. A number of metals consisting of copper, nickel, titanium, zirconium, iron, molybdenum, and type 347 stainless steel were irradiated at Brookhaven, Hanford, and the MTR under various exposure and temperature conditions. Post-irradiation measurements of mechanical and physical properties which were initiated at KAPL will be completed at HAPO.

FIGURE 3.3

Per Cent Burnup of Thorium as a Function
of Neutron Flux Density
(Thermal Plus Resonance)

Gradual Taper from End of
Reduced Section to Middle

FIGURE 3.4

Dimensions for Zr-2 and Zr-3 Tensile Specimen

A comparison of the effects of strain rate on the yield and tensile strengths of unirradiated Zircaloy-2 and Zircaloy-3 is shown in Table 3.2.

TABLE 3.2

**THE EFFECTS OF STRAIN RATE ON THE YIELD AND
TENSILE STRENGTHS OF UNIRRADIATED ZIRCALOY-2 AND ZIRCALOY-3**

	Strain Rate (inch per minute)					
	.005 Yield (X10 ³ psi)	(1)** Tensile	.05 Yield (X10 ³ psi)	(1) Tensile	.005/.05* (3) Yield (X10 ³ psi)	(3) Tensile
Zircaloy-2	39.0	51.6	46.0	58.6	41.4 - 2.4	56.5 - 2.1
Zircaloy-3	33.0	49.3	36.5	55.3	35.2 - 2.2	57.0 - 4.9

* The strain rate was .005 inch per minute beyond initial yielding and .05 inch per minute throughout the remainder of the test.

** Number of specimens represented.

UNCLASSIFIED

Increased strain rate affected an increase in yield and tensile strength for both Zircaloy-2 and Zircaloy-3. These increases in yield and tensile strength for Zircaloy-2 are about 18 and 14 per cent respectively, and for Zircaloy-3 about 11 and 12 per cent respectively.

From the data in Table 3.2, the small dependence on prior strain rate or "path independence" can be observed for both Zircaloy-2 and Zircaloy-3. It is seen that increasing the strain rate after the specimens have been strained beyond the yield point gives nearly the same ultimate strength as if the specimen had been strained only at the higher rate. To show this effect conclusively, additional test data are to be obtained.

The average tensile properties for both the irradiated and unirradiated Zircaloy-2 and Zircaloy-3 specimens are listed in Table 3.3. Typical engineering stress-strain curves for Zircaloy-2 and Zircaloy-3 are illustrated in Figures 3.5 and 3.6. After irradiation, the yield and tensile strengths of both materials increased about 45 and 15 per cent respectively which is analogous to about three to five per cent cold work in unirradiated material. The per cent total elongation for Zircaloy-2 and Zircaloy-3 decreased about 20 per cent, but the uniform elongation for Zircaloy-2 decreased about 55 per cent and for Zircaloy-3 about 45 per cent.

A drop-in-load yield point in the stress-elongation curves for irradiated Zircaloy-2 was consistently observed. The increment of stress between the upper and lower yield points averaged 360 psi which indicates the degree of interstitial "pinning" of dislocations induced by neutron bombardment. Irradiated Zircaloy-3 did not exhibit this behavior. A slight increase in the elastic modulus of the irradiated specimens was observed. This may be fortuitous due to the difficulty in locating the proportional elastic limit with the degree of accuracy required for these comparisons.

It may be observed in Table 3.3 that the yield strengths of both irradiated Zircaloy-2 and Zircaloy-3 were sensitive to strain rate. By increasing the strain rate by a factor of 10 (from 0.005 to 0.05 inch per minute) an increase of eight and nine per cent resulted in the yield strengths for Zircaloy-2 and Zircaloy-3 respectively. This increase is somewhat lower than that indicated in Table 3.2 for the unirradiated specimens.

One criterion for the rupture resistance of a material is the difference between the tensile and the yield strength. This difference represents the rupture stress tolerance available once yielding has begun. Another criterion is the ratio of uniform to total elongation. Here, uniform elongation refers to the extension within the gage length that occurs up to the point of horizontal tangency to the stress-elongation curve. Beyond this point, discontinuous deformation or "necking" occurs. A sizeable reduction in the uniform to total elongation ratios was perhaps the most marked effect of irradiation. This ratio decreased from 0.56 to 0.31 for Zircaloy-2 and from 0.53 to 0.37 for Zircaloy-3.

FIGURE 3.5

Schematic Stress-Strain Curves for Zr-2

FIGURE 3.6

Schematic Stress-Strain Curves for Zr-3

The stress tolerance for Zircaloy-2 after irradiation is about 5,000 psi compared to 15,000 psi before irradiation. Similar values for Zircaloy-3 are 14,000 and 22,000 psi respectively.

ETR Irradiations of Structural Materials. Information is currently needed concerning property changes in high strength materials which may occur under combined high temperature and high flux environments. Irradiations of selected materials will be conducted in the ETR 3 x 3 loop where these combined conditions will be available. The materials of interest include the age-hardening alloys Hastelloy-X, Monel-X, and K-Monel; the quench hardening alloy 410 stainless steel; and the work hardening alloys 347 stainless steel and Zircaloy-2. Tensile specimens of these materials will be irradiated under different conditions of cold work, and changes in mechanical and structural properties will be determined.

Zircaloy-2 specimens have been prepared and will be charged in the ETR 6 x 9 loop facility with the first fuel element irradiation.

In-Reactor Testing Facility - A. L. Bement and J. C. Tobin

The need for information concerning the changes in properties of materials used in the design of reactors, reactor fuels, and fuel element cladding is increasing rapidly. There are no factual data, or facilities for obtaining data, concerning the instantaneous effects of neutron radiation on the mechanical and physical properties of metals. To establish accurate criteria, in-reactor testing as contrasted to pre-post irradiation testing is required.

Research and development work to provide remotely controlled creep capsules for an in-reactor testing facility is being performed. This facility will be installed in the X-O level at 105-KW reactor. Both in-reactor and ex-reactor capsules are being developed in which the test conditions will be practically identical except for the neutron flux.

Efforts during the quarter were focused on the project proposal for the instrument facility. Cost estimates based on the scope design were completed, and the construction and progress schedules were prepared. The proposal is scheduled to be submitted to the AEC early in the last quarter of CY 1958.

A creep capsule which maintains a constant load by differential thermal expansion and indicates creep by temperature measurements was bench tested during September. The capsule operated with a Zircaloy-2 specimen at 300 C for 120 hours when it was unloaded. Although the operation followed the predicted pattern, the creep measurement had a random error of 4 per cent and the stress varied by 10 per cent. Tests will resume after these variations have been reduced.

MECHANICAL AND PHYSICAL PROPERTIES OF UNIRRADIATED MATERIALSThermal Conductivity of Uranium - R. L. Hales

Thermal conductivity and thermal expansion vary as a function of crystal orientation. An experiment to correlate these properties for uranium metal is in progress. Materials under study include as-worked, annealed, cold rolled production materials and dingot uranium specimens.

Results from thermal conductivity measurements at Battelle Memorial Institute have been received for a specimen of Hanford production material. A control specimen was retained and has been used to calibrate existing conductivity apparatus. Preliminary results show a deviation of ± 5 per cent from the data received. Apparatus modifications will be necessary to extend the present maximum temperature of 350 C to the 800 C achieved at Battelle.

FUNDAMENTAL STUDIESFractographic Studies - T. K. Bierlein and B. Mastel

No new developments or data can be reported this quarter.

Optical and Electron Microscopy - T. K. Bierlein and B. Mastel

Metallography. Metallographic studies of reactor materials, both irradiated and non-irradiated, are being conducted by specialized techniques. The prime objective of these studies is to supply basic information on the effect of neutron irradiation on the microstructure of reactor materials.

A knowledge of the behavior of the bonds in standard fuel elements during irradiation is important if the thermal conductivity changes markedly. Since the appearance and uniformity of such bonds can be determined by microscopic techniques, replicas of an etched specimen from an internally cooled uranium fuel element irradiated to a burnup of 0.2 a/o at a maximum fuel temperature of 300 C have been examined. The micrographs in Figure 3.7 show two regions of the intermetallic zone in a transverse section of the fuel element, as well as the high magnification appearance of the microstructure of the uranium. The integral compound layer formed during the reaction of uranium and AlSi has a normal appearance in (a), but in (b) evidently additional solid state diffusion has occurred and large voids have formed. Such diffusion in the absence of an irradiation environment does not occur at such a low temperature. The absence of grain boundary structure in the uranium, a high magnification view in

a. 250X

Alsi

b. 250X

U

c. 10,000X

FIGURE 3.7

**Replica Showing Microstructure of an Internally Cooled Uranium Fuel Element
Irradiated to a Burnup of 0.2 a/o at a Temperature Less Than 300 C.**

- a. A region of the compound layer which has undergone no apparent change.**
- b. Different region of the compound layer which has undergone change.**
- c. Fine structure in the uranium.**

An example of the application of this autoradiographic technique is shown in Figure 3.8. Molten eutectic AlSi reacts with uranium metal to form solid intermetallic compounds which under certain conditions break away and float considerable distances from the AlSi-Uranium interface prior to solidification of the AlSi. The alpha tracks associated with these regions positively show the presence of uranium.

The application of the above technique is currently being applied to cellulose acetate replicas of specimens which have become highly radioactive as a consequence of reactor irradiation. In such cases an extraction by the replica of submicroscopic particulates from the highly radioactive specimen occurs. The alpha tracks from the submicroscopic particulates on the replica can be correlated with the microstructural features shown by the replica. This type of pseudo-autoradiography appears to be a useful metallographic approach in the field of nuclear metallurgy. Figure 3.8 C shows the appearance of a 1.65 w/o Pu, 12 w/o Si aluminum base alloy which was replicated after irradiation. The star shaped tracks originate from regions in the replica which correspond to the silicon rich network interconnecting the spheroidal aluminum rich phase.

Micro-Deformation Studies. The tensile deformation of uranium at a temperature of 400 C has been studied in the hot stage microscope facility. Prior to the test, the specimen was etched and scribed with a Bierbaum Microcharacter. Figure 3.9 shows the appearance of the uranium before and after deformation. The fact that slip is an important deformation mode in uranium at this temperature is illustrated by the shift in the scribe lines within a grain; it is quite evident that considerable deformation occurs near grain boundaries. Additional microspecimens for study at higher temperatures are being prepared. Fracture in notched specimens will be followed with a cine camera.

A Study of Irradiated Zirconium and Zirconium Alloys with Electron Microscopy and X-Ray Diffraction - B. Mastel, R. L. Hales, and D. L. Gray

There is no new data to present this quarter.

Basic Swelling Studies - J. M. Tobin, J. C. Tobin, J. H. Sako, D. C. Kaulitz, T. K. Bierlein, B. Mastel, and K. R. Merckx

The diffusion of the inert gases, krypton and xenon, and subsequent void formation results in swelling of irradiated fuel materials. This swelling appears to be dependent upon (1) irradiation temperature, (2) temperature cycling, (3) restraining pressure, (4) burnup, (5) burnup rate, (6) fuel composition, and (7) material parameters such as residual work and grain size.

Al AlSi U+AlSi
a intermetallics

FIGURE 3.8

A New Application of Microautoradiography

- Appearance of Saran replica and adhering NTB emulsion stripped from a slug wafer with Scotch tape - Bright Field Illumination.
- Same as a, but photographed with polarized light illumination.
- Appearance of a replica of irradiated Al-Si-Pu alloy. Radioactive particulates present in the replica are responsible for the tracks in the thin NTB emulsion placed on the replica.

Magnification: 250X

Prior to deformation

After deformation

FIGURE 3.9

Tensile Deformation of Uranium at 400 C
Grain boundary sliding, slip, and twinning are revealed. 200X

Physics calculations pertinent to neutron absorption by the stainless steel capsule walls indicate that considerably less than 20 per cent of the thermal flux will be absorbed. Design modifications will be made on the basis of the foregoing evaluations.

A second type of instrumented irradiation capsule is being planned. In addition to the temperature monitoring and controlling functions of the above capsule, the second will be designed to make continuous in-reactor volume change measurements on the swelling specimens. The latter function is desirable since it would considerably reduce the total number of capsule irradiations necessary to correlate swelling with burnup. However, it will be a more complex capsule design and it will require much more time to develop than the above-mentioned capsule. If possible, the second type of capsule will be made by modifying the first capsule design in order to benefit from the experience gained from irradiation of the first capsule.

Metallographic Studies. The kinetics of fission gas bubble nucleation and growth in uranium must be ascertained as a function of irradiation exposure and temperature, since the swelling at high temperatures causes extensive volume changes. Optical electron microscopy affords a direct physical means for studying the number and sizes of bubbles in the metal.

Specimens irradiated at temperatures less than 300 C to burnups up to 0.2 a/o have been examined by optical and electron microscope techniques to reveal microstructural changes. Negative cellulose acetate replicas of the etched specimens reveal the presence of very small pores in the uranium. These pores, 200-400 Å in diameter, appear in uranium with burnups as low as 0.01 a/o and become more numerous at higher burnups. Figure 3.10 shows the appearance of a control, non-irradiated uranium specimen, (a); a specimen with a burnup of 0.02 a/o, (b); and a specimen with a burnup of 0.2 a/o, (c). At the high magnification obtainable with the electron microscope, pores can be readily detected by virtue of their characteristic shadows. At the low temperatures of irradiation, less than 300 C, the holes are randomly distributed throughout the uranium. A study of pores in uranium with a burnup of 0.2 a/o after a one-hour anneal at 600 C is in progress. Successive anneals should show how the gas bubbles grow and whether they form at preferential sites.

Since the possibility exists that the small holes are artifacts which arise during one of the several experimental steps, non-irradiated uranium of various fabrication histories has been polished, etched, and replicated for subsequent electron microscope examination. In addition, fracture surfaces of both non-irradiated and irradiated uranium specimens have been replicated. Although this phase of study has not as yet been completed the results obtained support the conclusion that the 200-400 Å diameter holes in uranium occur as a consequence of reactor irradiation.

FIGURE 3.10

Electron Micrographs of Uranium - 40,000X
The arrows point to typical holes, 200-400Å in diameter.
a. Non-irradiated control specimen.
b. Irradiated to a burnup of 0.02 a/o.
c. Irradiated to a burnup of 0.2 a/o.

alloys. However, the effectiveness of nickel as a diffusion barrier during extremely long anneals will require further study because of the complexity of ternary diffusion in the U/Ni/X-8001 system.

Analysis of U/X-8001 and U/A-2 aluminum alloy diffusion couples, annealed five days at 300 C, indicated that the rates of penetration of uranium into A-2 and into X-8001 aluminum alloy are both comparable to the rate of penetration of uranium into AlSi at this temperature. The same was found in U/X-8001 couples annealed seven days at 250 C. No appreciable diffusion occurred in U/A-2 couples annealed seven days at 250 C. Several additional U/A-2 couples will be annealed at 250 C to check the validity of this data.

An estimation of the maximum times and temperatures at which slugs could be run during tests in the KER pressurized water loops without failure due to cladding penetrations will be required in the near future. For this reason, penetration of uranium into the X-8001 cladding in AlSi bonded depleted uranium slugs is being studied. Fifty specially canned slugs have been prepared for this study. Ten of these were selected at random and examined metallographically to determine the initial condition of the bond. Defective bonds were found in sections approximately one inch from the cap end of each of these ten slugs. Other areas of the slugs seem to be normally bonded. Two slugs have been annealed 14 days at 360 C and one has been annealed 14 days at 300 C in water autoclaves. These slugs are now being examined.

Irradiation Study of Thermocouples - J. H. Sako and D. R. Green

This investigation has been initiated to determine the effect of reactor irradiation on the stability of thermocouples in contact with gas atmospheres. Its purpose is to determine the extent to which reactions between the atmospheres and common metal couples are affected by irradiation. The thermal emf's generated by chromel-alumel, copper-constantan, and iron-constantan couples are to be measured during extended in-reactor exposures to determine their stability at temperatures between 150 C and 800 C.

The quartz insulated and asbestos insulated thermocouple lead wire assemblies have received a total of 1872 hours of exposure in KW Reactor. After 1104 hours of exposure, both assemblies started to show a decrease in resistance between the pairs of thermocouple lead wires. At 1632 hours, the asbestos insulated wires read 5.3×10^5 ohms (Cr-Al), 5.0×10^6 ohms (Cu-Constantan) and 5.3×10^6 ohms (Fe-Constantan), and the quartz insulated wires read 3.3×10^6 ohms (Cr-Al), 8.2×10^6 ohms (Cu-Constantan) and 5.5×10^6 ohms (Fe-Constantan). Upon further exposure, all the resistance readings started to increase, and with the exception of the asbestos insulated chromel-alumel pair, have reached the

minutes) have been reported to 224 days exposure. To date, no marked change in corrosion rate (breakaway) has been noted. The corrosion rate is defined as

$$CR = \frac{\log W_2 - \log W_1}{\log t_2 - \log t_1},$$

where W_2 is the weight gain (mg/dm^2) corresponding to a test time of t_2 and W_1 is the weight gain corresponding to a test time of t_1 (days). For the Zircaloy-2, (50 per cent cold work, heat treated in vacuum at temperatures from 300 to 700 C, 1,000 minutes and 800 C, 100 minutes), $CR = 0.41$. For both the vacuum melted and argon melted Zircaloy-3 (same history as above) $CR = 0.48$. After 224 days in test, the weight gains of the Zircaloy-2 varied from 34 to 46 mg/dm^2 . This may be compared to total weight gains of 82 to 98 mg/dm^2 obtained after the same test time for Zircaloy-2 samples heat treated in air at 500 to 700 C for 1,000 minutes and 800 C, 100 minutes. The Zircaloy-2 specimens heat treated at temperatures below 500 C in air were withdrawn from test at times less than 224 days when the weight gains exceeded 100 mg/dm^2 . As mentioned previously, (3) the helium treatments produced results very similar to those obtained using air as an atmosphere. To date it appears that vacuum heat treatments in general result in improved corrosion resistance of Zircaloy-2 and Zircaloy-3 as compared to the previous results from heat treating in air or helium.

Recovery of Cold Work Measured by X-Ray Diffraction. Recovery of 50 per cent cold worked zirconium heat treated in helium at six temperatures from 300 to 800 C for 10 and 100 minutes was investigated by measuring X-ray line broadening. Two quantities, $B_{\frac{1}{2}}$ and $\frac{H}{I_{\text{max}}}$ were used as an index of cold work recovery. $B_{\frac{1}{2}}$ is the half width of the alpha 1 peak at half of the maximum intensity (see Figure 3.11) and $\frac{H}{I_{\text{max}}}$ is a measure of alpha 1-alpha 2 resolution (see Figure 3.11). Cobalt $K\alpha$ radiation was used and the measured peak was (105). The two sets of recovery data are shown in Figures 3.12 and 3.13. Both $B_{\frac{1}{2}}$ and $\frac{H}{I_{\text{max}}}$ vs. temperature clearly delineate the recrystallization temperature for 50 per cent cold worked zirconium as 600 and 640 to 650 C respectively for the 100 minute and 10 minute heat treatments. Measurements of line width indicated a gradual recovery below 400 C. This reflects a change in the structure which does not affect the hardness noticeably. Figure 3.14 summarizes hardness data obtained from these samples. It should be noted that a small increase in hardness occurs at 600 and 700 C for the 100 and 10 minute helium treatments. This increase is not observed by the X-ray work. It is suggested that a second reaction, most likely gas pickup from the helium, causes the small increase in hardness.

FIGURE 3.11

Distribution of Intensity as a Function of Diffraction Angle
for Completely Re-crystallized Zirconium

FIGURE 3.12

Recovery of Line Width for 50 Per Cent Cold Worked Zirconium,
Heat Treated in Helium

FIGURE 3.13

Recovery of $\alpha_1 - \alpha_2$ Resolution for 50 Per Cent Cold Worked Zirconium,
Heat Treated in Helium

FIGURE 3.14
Recovery of Hardness,
50 Per Cent Cold Worked Zirconium,
Heat Treated in Helium

METALLIC FUELS DEVELOPMENTNEW FUELS DEVELOPMENTThermal Contact Conductance of Fuel Element Materials - R. G. Wheeler

The thermal contact conductance of joints between materials covering a wide range of thermal and mechanical properties was measured. Changes in contact conductance caused by changing experimental conditions or materials are generally in accord with the theoretical treatment of Cetinkale and Fishenden.⁽¹⁾ The actual conductance of a joint is so sensitive to test conditions that measured values obtained in tests closely approaching service conditions are necessary if the core temperature of an unbonded fuel element is to be predicted with accuracy.

Aluminum is used as a fuel element jacket for cores that may vary from refractory metals to ceramics. Figure 4.1 shows the effect of pressure on the conductance of joints between aluminum and various materials. With one exception, the surfaces in contact were lapped to a 10 microinch rms surface. The Zircaloy-2 specimen had a machined (35 rms) surface. All tests were made in a vacuum (<3 microns of Hg) except where the symbol "He" indicates that $1/2$ atmosphere of helium was used. Aluminum in contact with aluminum illustrates the effect of temperature. With 230 psi on the joint, the conductance was 2300 BTU/hr/ft²/°F at a mean joint temperature of 110 C and 6800 BTU/hr/ft²/°F at a mean joint temperature of 200 C. A joint between aluminum and reactor grade graphite compares with a joint between aluminum and other metals at joint pressures less than 100 psi, but above 100 psi the conductance approaches that of bonded joints. The conductance of the joint between aluminum and sintered uranium dioxide increased rapidly with joint pressures above 200 psi. The poor conductance of oxidized Zircaloy-2 in contact with aluminum may be attributable in part to the machined instead of lapped zirconium surface.

The conductance of sintered uranium dioxide, 95 per cent of the theoretical density of UO₂, in contact with various materials is shown in Figure 4.2. All the values obtained in a vacuum are less than 500 BTU/hr/ft²/°F except for the uranium dioxide in contact with aluminum. The conductance of these joints becomes essentially infinite above 400 psi and 250 C joint temperature.

Miscellaneous contact conductance values obtained recently are listed in Table 4.1.

One prototype of a test specimen designed to measure the pressure dependance of thermal contact conductance between uranium and aluminum in reactor was fabricated. The unbonded test element has ellipsoidal end closures to prevent thermal ratcheting of the jacket. A thermocouple on the axis of the cylindrical element will measure the core temperature change caused by changing the pressure of the cooling water.

FIGURE 4.1

Thermal Contact Conductance of Joints
Between Aluminum and Various Materials

FIGURE 4.2

Thermal Contact Conductance of Joints Between
Sintered Uranium Dioxide and Various Materials

DECLASSIFIED

HW-58205

TABLE 4.1

THERMAL CONTACT CONDUCTANCE OF VARIOUS JOINTS

Run Number	Contact Conductance BTU/hr/ft ² /F	Joint Pressure psi	Mean Joint Temperature °C	Direction of Heat Flow	Thermal Flux BTU/hr/ft ²	Atmosphere
Aluminum (10 microinches surface roughness) and Graphite (as machined surface)						
5	30,000	230	210	Al to Graph	160,000	vacuum
5	30,000	230	180	Graph to Al	160,000	vacuum
6	1,700	33	210	Al to Graph	85,000	vacuum
6	950	33	150	Graph to Al	85,000	vacuum
7	1,600	63	210	Al to Graph	85,000	vacuum
7	1,300	63	150	Graph to Al	85,000	vacuum
10	30,000	230	110	Graph to Al	75,000	vacuum
12	6,400	63	110	Graph to Al	66,000	vacuum
Zircaloy-2 and Graphite (both with as machined surfaces)						
10	1,600	230	150	Zr-2 to Graph	75,000	vacuum
12	700	63	150	Zr-2 to Graph	65,000	vacuum
Zircaloy-2 (0.0002 g/cm ² of oxide scale) and Graphite (as machined)						
25	790	230	410	Graph to Zr-2	56,000	vacuum
26	2,300	115	350	Graph to Zr-2	100,000	1/2 Atm. He
27	300	115	370	Graph to Zr-2	51,000	vacuum
28	150	63	400	Graph to Zr-2	36,000	vacuum
Sintered Uranium dioxide and sintered uranium dioxide (surface roughness 10 microinches rms)						
97	468	435	220	UO ₂ to UO ₂	73,000	vacuum
98	157	230	220	UO ₂ to UO ₂	56,000	vacuum
99	120	230	120	UO ₂ to UO ₂	21,000	vacuum
1	146	60	220	UO ₂ to UO ₂	50,000	vacuum
Aluminum (10 microinches surface roughness) and Zircaloy-2 (machined)						
10	1,700	230	280	Al to Zr-2	75,000	vacuum
12	710	63	300	Al to Zr-2	66,000	vacuum
38	845	230	88	Zr-2 to Al	73,000	vacuum
40	743	115	80	Zr-2 to Al	68,000	vacuum
41	598	64	65	Zr-2 to Al	55,000	vacuum
42	803	115	52	Zr-2 to Al	50,000	vacuum
43	5,400	115	67	Zr-2 to Al	65,000	1/2 Atm. He

DECLASSIFIED

-4.6-

HW-58205

TABLE 4.1 (contd.)

THERMAL CONTACT CONDUCTANCE OF VARIOUS JOINTS

Run Number	Contact Conductance BTU/hr/ft ² /F	Joint Pressure psi	Mean Joint Temperature °C	Direction of Heat Flow	Thermal Flux BTU/hr/ft ²	Atmosphere
Aluminum and Uranium (10 microinch rms surface roughness)						
46	642	460	97	U to Al	81,000	vacuum
47	523	230	88	U to Al	72,000	vacuum
48	462	115	88	U to Al	67,000	vacuum
49	3,150	115	111	U to Al	96,000	1/2 Atm. He
57	5,972	460	91	U to Al	65,000	vacuum
58	2,550	230	92	U to Al	64,000	vacuum
59	2,030	115	91	U to Al	62,000	vacuum
60	1,780	63	90	U to Al	62,000	vacuum
61	*1,468	115	77	U to Al	61,000	1/2 Atm. He

* Introduction of He probably caused additional oxide to form on the uranium surface.

Restraint of Uranium Swelling by Zirconium Cladding - J. W. Weber

Swelling rates of unrestrained uranium irradiated in the 400-800 C temperature range have been measured, but to date no swelling data are available for unalloyed Zr-2 clad uranium with a 250-350 C cladding surface temperature and a 450-600 C maximum fuel temperature. The swelling behavior of clad uranium operating in the above temperature ranges is of importance for Hanford's fuel element development work.

The initial fuel rod irradiation GEH 3-31, which operated at a measured average maximum uranium temperature of 400 C and has an estimated exposure of 2100 MWD/T (11.6×10^{19} fissions/cm³) has been returned to Hanford for examination. A graph of the maximum core temperature versus exposure time is given in Figure 4.3. An increase of 0.4 per cent has been found on the outside diameter of the fuel rod. Measurements over two gauge length at different axial locations on the fuel rod cladding show no significant changes in length. There was an increase in one gauge length of 0.4 per cent and a decrease of 0.6 per cent in the other. A longitudinal section cut into the bottom of the fuel rod shows that the uranium has expanded into the 0.025-inch gap between the end cap and the original free uranium surface (Figure 4.4). If the uranium fills both end gaps the length increase is 1.1 per cent. With the above length and diameter increases the corresponding volume increase is 1.9 per cent. Density measurements will be made to confirm this value

FIGURE 4. 3

Measured Maximum Uranium Core Temperature
of GEH 3-31 versus Exposure Time in Reactor

FIGURE 4. 4

Ends of Fuel Rod Showing Uranium Expansion into 0.025-inch
End Gaps Between Free Uranium Surface and End Caps. Top
of Fuel Rod at Right.

DECLASSIFIED

-4.9-

HW-58205

Uranium-Magnesium Matrix Fuel Material - J. W. Goffard

Uranium-magnesium matrix material is being investigated as a possible high burnup fuel. Initial irradiation testing of this candidate fuel material was accomplished at the MTR and has been reported in HW-43973. (2) Post-irradiation examination of 12 specimens after various burnups ranging from 1000 to 20,000 MWD/T, indicated that the material was dimensionally stable. Subsequent irradiations of powder metallurgy U-Mg specimens and a HAPO sized U-Mg fuel element have revealed that the fuel material may not be as dimensionally stable as first indicated. Post-irradiation examination of the latest U-Mg fuel element after a calculated exposure of 9900 MWD/T, corroborates the initial observation as to the dimensional stability of the material. The U-Mg specimens which have displayed dimensional stability have been clad in 0.050 inch thick Zr-2, whereas the unstable specimens have been clad in 2S aluminum. The degree of restraint the fuel material meets during its irradiation as offered by the two different cladding materials may be a clue to its dimensional behavior. A companion piece to the above mentioned 9900 MWD/T exposure fuel element is still being irradiated at the MTR (current exposure about 9000 MWD/T) and will be examined following attainment of its goal of 10,000 MWD/T.

A resume of U-Mg irradiations and pertinent remarks follows:

<u>Test</u>	<u>Description</u>	<u>Exposure</u>	<u>Increase^{v/o}</u>	<u>Remarks</u>
1	12 specimens, 1-1/2 in x 0.40 in dia., clad in 0.050 in Zr-2, 50 v/o U machine chips	1,000 - 20,000 MWD/T	None detected	Fuel readily removed from capsule
2	5 specimens, 1-1/2 in x 0.35 in dia., clad in 0.136 in Al, powder met. prepared, 62-67 v/o U powder	1,000 MWD/T 1,000 3,000 3,000 3,000	2.63 4.41 8.56 9.74 10.2	Swelling noted on capsules as well as fuel
3	1 HAPO size fuel element, core 3.6 in x 1.35 in dia., clad in 0.050 in Al, U machine chips, 1.75% U-235, 35 v/o U	800 MWD/T	2.7	Swelling of both fuel element and core
4	1 fuel element, core 4.0 in x 1.00 in dia., clad in 0.050 in Zr-2, 65 v/o U shot	9,900	About 0.5	Based upon 0.007 in dia., increases and 0.004 in decrease in length of element

DECLASSIFIED

-4.11-

HW-58205

occurred and the only microstructure change observed was a sharpening of the sub-grain boundaries and development of equiaxed recrystallized grains. The results indicate that no adverse changes are to be expected in subsequent bonding of elements produced by the cold closure process.

Zircaloy-2-Uranium Coextruded Rod Closures - T. T. Clauson

To eliminate the costly process of boring and etching uranium from coextruded rod for welded end closures, the development of welding end caps which are slipped over the end of such rod has achieved some progress. To make the closure, a Zircaloy-2 end cap is slipped over a fuel rod and a circumferential lap weld is made between the cladding and end cap by the electron beam welding process. After welding the end cap is swaged down to the original rod diameter. First attempts resulted in weld contamination by uranium diffusion. A severely cracked weld zone appeared during the swaging operation. By cleaning the weld area by a brite pickle and controlling the weld penetration more accurately, rods were closed satisfactorily. These rods are being autoclaved for 120 hours at 350 C and subsequent metallography will follow.

Closure of Coextruded Rod Fuel Element End Caps by Brazing - W. I. Steinkamp and W. L. Wyman

The possibility of using brazed end caps closures on coextruded rod fuel elements is being explored. If it is possible to produce a satisfactory brazed end cap closure, the fabrication costs of these elements could be reduced markedly because it is necessary to machine and etch out the uranium core to permit insertion of the present plug type caps.

To facilitate studies on brazing an electron gun which produces a diffuse beam of electrons has been developed. This is substituted for the welding unit which produces a highly concentrated electron beam that is not particularly well suited for brazing. A travelling jig has also been designed and put into operation. This permits making linear brazed joints so that the physical characteristics of the braze may be more fully evaluated.

Braze alloys studied to date have melting points near that of the uranium core and as a result very close temperature control is required. Other alloys are under consideration which will give a greater temperature spread and thus simplify the brazing operation. An alloy of Zr + 5% Be has shown the best results to date. The Zr-Be-Cu alloys studied have not produced a satisfactory braze.

Very limited observation indicates a possibility that a diffusion weld might be formed by brazing with fine silver and holding the parts at temperature until the silver had diffused into the Zr-2. The physical and chemical properties of such a bond remain to be evaluated.

FIGURE 4.5

Resistance Spot Weld Attachment System for Fabrication of Self-Supported Fuel Elements

50 Diameters. Etch. 0.5% HF
(Attachment Area)

250 Diameters. Etch. Electrolytic Phosphoric 0.5% HF
(Compound Area Beneath Weld Nugget)

FIGURE 4.6

Fuel Element Support Attachment on M-388 Jacketed I & E
Element-Resistance Spot Weld System

FIGURE 4.7
Finished Self-Supported I and E Fuel Elements
(M-388 Jacket)

DECLASSIFIED

-4.16-

HW-58205

Direct Casting of Uranium into Zircaloy Cans - J. C. Tverberg and R. S. Kemper

Economic savings which would result from the direct casting of the uranium fuel into Zircaloy-2 jackets would be significant. Closure on the cast pieces could be completed by welding an end cap in place with an electron beam welder. Heat treating the completed element would result in a finished fuel element with proper grain size and orientation.

Initial work was done by casting into unheated Zircaloy tubes in which the wall thickness to diameter ratio was very high. The lower section of these tubes was completely unbonded, but the upper section, where the Zircaloy had been heated into the beta phase (980 C) by the molten uranium did exhibit some bonding. Bonds at this point were quite strong, as determined by the peel test, especially in some small areas where spots of Zircaloy were torn from the jacket. Examination of Figure 4.8 indicated a dark grey bond which touches the uranium in one spot. As the grey bond appears to be the stronger bonding layer, this spot could be one of higher strength. Figure 4.9 illustrates the typical bond obtained at the heated end of the can.

A second series of test utilized a heated mold in which the Zircaloy can had a lower wall thickness to diameter ratio. Examination of the casting at 500 diameters indicated that a very narrow alloy zone had formed on the zirconium side of the interface the full length of the element as well as the bottom. Sections were salt bath beta heat treated as many as three times without fracturing the bond. The resulting bond is shown in Figure 4.10. Samples were also heated in a vacuum to 750 C and held for four hours, then slowly cooled, also without fracturing the bond. Figure 4.11 illustrates the bond after this heat treatment.

The next effort was to cast into I and E Zircaloy-2 cans. The graphite mold, Figure 4.12, was filled with three thermocouples and then inserted into the mold heater to preheat the can prior to casting. Mold preheat temperatures ranging from 625 C to 825 C have been used; however, complete evaluation of the cast pieces at the higher temperature is not complete. Uranium was cast, in all instances, at 1475 C and at a pressure of 2×10^{-3} mm Hg. Mold temperature readings were made every minute until the mold cooled to the uranium beta transition temperature. A typical curve is shown in Figure 4.13 which shows the maximum can temperature reached, and the cooling rates involved.

These castings did not have the extensive bonding indicated in the earlier experiments, but the Zircaloy did show the effects of uranium migration. Excessive uranium and zirconium carbides were found along the Zircaloy-uranium interface which could account for the poorer bond. This is illustrated in Figure 4.14. In each casting the internal tube bowed

FIGURE 4.8

Bond formed by direct casting of uranium into Zircaloy-2 containers. Note the numerous carbides precipitated along the interface. The dark gray bonding layer appears to be the strongest. 500X

FIGURE 4.9

Bond formed by direct casting of uranium into Zircaloy-2 containers. Note the presence of three distinct bonding layers. 500X

FIGURE 4.10

Direct cast into thin wall Zircaloy-2 can, beta
heat treated at 730 C and water quenched, 500X

FIGURE 4.11

Direct cast into thin wall Zircaloy-2 can, heat
treated at 750 C for two hours, 500X

FIGURE 4. 12

**Graphite Mold for the Direct Casting
of Uranium Metal into Zircaloy Cans**

UNCLASSIFIED

-4.21-

HW-58205

FIGURE 4.13

Typical Cooling Curves for Direct Cast I and E Fuel Element

ARC-08 RICHARD, WASH.

UNCLASSIFIED

FIGURE 4.14

Bonding zone in direct cast Zircaloy-2 clad fuel element. Section near bottom on I.D. tube. Note the excessive carbides along the Zircaloy-uranium interface. 500X

FIGURE 4. 15
Profiles of Cast I and E Fuel Element

DECLASSIFIED

-4. 24-

HW-58205

those susceptible to fracture under thermally-induced stress; (b) immersion of refrigerated fuel elements in hot water to induce expansion of the jacket away from the core in a manner similar to that described for the Frost Test. The fracturing bond should produce an audible sound useful in detecting affected fuel elements; (c) use of a microcharacter hardness test to scratch a groove across a polished specimen of the compound layer to be tested. The load necessary to cause spalling along the scratch groove would be a measure of the frangibility of the bond. Although completely destructive, this test should give a revealing analysis of the bond structure to aid in relating causes and effects.

Other tests that must be developed to permit a thorough study of the factors affecting bond quality and permanency are: (a) a test to measure heat transfer rates from core to external jacket surface; (b) quantitative measurement of the effects on a canned fuel element of long time holding at somewhat elevated temperatures (~ 250 C); (c) a method of analyzing the composition of the various strata of the compound layer and of determining the area covered by each zone.

Swaging - F. B. Quinlan

Swaging dies, Figure 4. 16, for producing external ribs on tubing, have been used successfully on aluminum, Figure 4. 17. Some progress has been made with Zircaloy-2 tubing, Figure 4. 18, but this cannot be properly evaluated until Zircaloy tubing with sufficient wall thickness is available.

The forming of internal ribbing has been very successful with aluminum and is now being attempted on Zircaloy tubing. If this proves successful, the simultaneous forming of both internal and external ribbing will be tried.

FUEL EVALUATION

Seven-Rod Cluster Fuel Elements - T. T. Claudson

The cluster fuel geometry meets many of the requirements for the proposed NPR to be constructed at Hanford. As part of a program to develop an extended surface fuel element capable of operating in a high temperature recirculating facility, four seven-rod cluster fuel elements were charged for irradiation. These fuel elements were fabricated from 0. 500 inch uranium clad in 30 mil wall 304 stainless steel tubing. Thermocouples were placed in the fuel to monitor fuel core temperature and also fore and aft of the fuel basket to obtain data making it possible to calculate the operating specific power. The fuel is now operating at 103 kw/ft with a core temperature of 310 C. The fuel has reached a burnup of about 2000 MWD/T. Goal exposure for this irradiation will be increased to 3500 MWD/T.

DECLASSIFIED

-4. 26-

TABLE 4. 2

FOUR-ROD CLUSTER FUEL TEST

Fuel Material	Natural Uranium
Clad Material	AlSi Type 304L Stainless Steel
Spacer Ring Material	17-4 PH Precipitation Hardening Stainless Steel, aged at 975 F to R _c 42
Support Clips	AlSi Type 304L Stainless Steel, Cold Rolled
Clad Outer Diameter	0. 630 ± . 001
Clad Inner Diameter	0. 570 + . 002 - . 000
Core Diameter	0. 5695 ± . 0005
Core Length	8. 000 ± . 010
Fuel Element Length (Over-all)	9. 350 Inch
No. of Elements in Charge	20
Process Tube Diameter	2. 125
Coolant Flow Rate	60 gpm
Maximum Coolant Temperature	240 C
Maximum Core Temperature	500 C (Calculated)
Average Exposure	2300 MWD/T
Maximum Exposure	3000 MWD/T
Post Irradiation Clad Outer Diam.	0. 630 ± . 001
Post Irradiation Core Diam.	0. 574 ± . 001
Post Irradiation Core Length	8. 030 ± . 003
Core Volume Increase	2. 3 ± . 3%
$\frac{\% \Delta V}{V}$	7 (Calculated)
% Burnup

FIGURE 4.16
Swaging Die

FIGURE 4.17
Ribbed Aluminum Tube

UNCLASSIFIED

-4.30-

HW-58205

FIGURE 4.18
Ribbed Zircaloy-2 Tube

FIGURE 4. 19
Four-Rod Cluster Fuel Element

FIGURE 4.20

Stainless Steel Clad Uranium Rod Exposed
to 3000 MWD/T in 240 C Water Coolant

FIGURE 4.21

Micrograph of uranium-stainless steel interface
after 3000 MWD/T at 350 C. Cathodic bombard-
ment etch. 250X

UNCLASSIFIED

-4.34-

HW-58205

FIGURE 4.22
Zircaloy-2 Clad Rod and Tube Fuel Elements

UNCLASSIFIED

UNCLASSIFIED

-4.35-

HW-58205

FIGURE 4.23

Details of Fuel Element Supports and End Seals

UNCLASSIFIED

DECLASSIFIED

-4.36-

HW-58205

During the co-axial irradiation, the elements experienced a period of reduced power operation when the MTR was fueled with an experimental Pu fuel loading. At the conclusion of the Pu fuel experiment, the co-axial elements were returned to their previous power of 88 kw/ft and the attained temperatures indicated that operation was normal. Apparently, the cyclic operation from high to low and return to high power did not effect the operating characteristics or behavior of the fuel element. As yet, the same cannot be stated for cyclic operation in which the power is raised to some higher power and then reduced back to normal. The temperature monitored element has been discharged and will be shipped to the HAPO radio-metallurgy facility for examination. The last phase of the irradiation will be completed after the second element has attained a goal exposure of 1500 MWD/T.

Evaluation of Vacuum-Canned and Hot-Pressed Slugs - E. A. Smith

Post-irradiation evaluation of the experimental and control fuel elements covered by PT's IP-44-A and IP-45-A was completed. In these tests the irradiation behavior of vacuum canned, hot-press canned, and lead-dip canned I and E fuel elements was studied comparatively, on the basis of quality loss suffered by the fuel elements as they were irradiated to progressively increasing exposures, up to 800 MWD/T. No failures were incurred in any of the fuel elements, and none of the three groups appeared outstandingly superior or inferior with respect to ability to withstand the rigors of reactor exposure.

The vacuum-canning process was an attempt to apply the Al-Si bonding principle by a technique which would reduce the number of critical operating variables in comparison with the standard hot-dip canning process. Thereby the attainment of high quality in a greater-than-normal percentage of the output would be expected.

The hot-press canning process was an attempt to provide high quality in the product by supplanting the Al-Si bond with a diffusion bond between the uranium core and a nickel sweater coat, and between the nickel coat and the aluminum jacket. Thereby many of the weaknesses of the typical Al-Si bond were expected to be eliminated.

The results of these production tests showed that all three types of fuel elements suffer to essentially the same extent under similar reactor operating conditions, and that if the process used in canning provides non-leaking jackets and uniform heat-transfer, fuel element failure during irradiation is promoted by factors not associated with the canning process. Far more important in this respect are:

1. The nature of the jacket material and the pre-treatment it has undergone.
2. The nature of the core material and the heat-treatment it has undergone.

CERAMIC FUELS DEVELOPMENT

FUEL FABRICATION

UO₂ Powder for Swaging - D. W. Brite and R. J. Anicetti

Enriched UO₂ powder (2.42 per cent U-235) was prepared for swaging fuel elements for irradiation in the Vallecitos Boiling Water Reactor (VBWR). The ADU source UO₂ was supplied to CFDO as a pyrophoric powder having high surface area (7 m²/g). Preparational steps included helium purge, isostatic pressing (50,000 psig), sintering 10 hours in hydrogen at 1700 C, crushing, pulverizing, and screening. The minus 20 mesh fraction was heated four hours in helium at 1050 C to displace adsorbed hydrogen, cooled in helium, and degassed at 700 C for three days in vacuum. The UO₂ was immediately loaded into Zircaloy-2 tubes for swaging.

The density of the sintered rods was 95 per cent of theoretical. Examination of a polished surface of the sintered UO₂ revealed both porous and non-porous areas, as illustrated in Figure 5.1. The effect of lumps (approximately 1/4 inch in diameter) on the texture of the non-sintered, pressed rods is shown in Figure 5.2, a photograph of a fractured rod before sintering. Photomicrographs, Figures 5.3 through 5.5, of a polished, sintered specimen illustrate the contrast in structure of the porous and non-porous areas. Segregation of the UO₂ into porous and nonporous regions is attributed to the lumpy nature of the powder used for the rods.

UO₂ Fuel Cores for SRE Irradiation - D. W. Brite and R. J. Anicetti

Enriched (8 per cent U-235) uranium dioxide rings, 2.310 in. OD x 1.786 in. ID x 1/2 in. long, and pellets 0.505 in. in diameter were fabricated for a six-foot tubular fuel element to be irradiated in the Sodium Reactor Experiment as part of a joint HAPO-Atoms International program. The fuel cores were fabricated by the conventional techniques of ball milling, cold pressing, sintering and grinding. The density of the tubular fuel cores varied from 92 to 96 per cent of the theoretical UO₂ density, and for the pellets the average density was 93.9 per cent of theoretical UO₂ density. A group of tubular UO₂ fuel cores are shown in Figure 5.6, and the pellets in Figure 5.7.

The inner surfaces of the sintered UO₂ tubes were ground to size with a standard internal grinder, using a resin bonded diamond wheel. The internally ground tubes were placed on a steel mandrel and the outer surface ground concentric with the inner surface. Ends were ground flat and perpendicular to the tube axes. A 37 C 100-H7Y (Norton Co.) grinding wheel was used to grind the ends and the outer surface. Pellets were ground to size with a belt centerless grinder.

FIGURE 5.1
Sintered, Isostatically Pressed UO₂
7X

FIGURE 5.2
Fractured Cross-section of Isostatically Pressed UO₂ Rod
3X

FIGURE 5.3

Non-porous and Porous Areas of Sintered UO_2
100X

FIGURE 5.4

Non-porous Area of Sintered UO_2
250X

FIGURE 5.5

Porous Area of Sintered UO_2
250X

UNCLASSIFIED

-5.5-

HW-58205

FIGURE 5.6
Eight Per Cent Enriched UO_2 Fuel Cores

ASG-08 RICHARD, WMH.

UNCLASSIFIED

FIGURE 5.7
Eight Per Cent Enriched UO_2 Fuel Pellets

UNCLASSIFIED

-5.7-

HW-58205

FIGURE 5.8

Fuel Core for Instrumented SRE Test Element

UNCLASSIFIED

FIGURE 5.9
Swaged UO₂ Fuel Rods

TABLE 5.1EFFECT OF UO₂ PARTICLE SIZE ON SWAGED DENSITY*

<u>Powder Description</u>	<u>Particle Size Distribution, Tyler Screen Mesh Sizes</u>	<u>Packed Density</u>	<u>Swaged Density</u>
Arc-Fused UO ₂	-20	69	89
"	-20 + 35 (60%)	72	89.5
"	-200 (40%)	58	91.0
"	-20 + 35 (100%)		
Sintered UO ₂	-20	65	84.0
"	-20 + 35 (60%)	68	84.6
"	-100 (40%)	65	85.3
"	-20 + 35 (60%)	53	85.5
"	-65 + 100 (40%)		
"	-20 + 35 (100%)		

*UO₂ swaged in 3/4 inch OD x 0.035 inch wall Zircaloy-2 tubing

UO₂ density has been increased approximately one per cent by allowing entrapped air to escape during swaging. This is accomplished either by using a porous material such as cotton for end plugs, or by evacuating the air from the fuel rod before swaging. A high-fired UO₂ powder prepared by Union Carbide Nuclear Company was investigated. The swaged densities obtained with this powder were one to two per cent below densities previously obtained with sintered UO₂ powders prepared at Hanford.

Internally Cooled Only Fuel Element - M. K. Millhollen

Feasibility of fabricating a proposed internally cooled only fuel element was demonstrated by the assembly of a section of such an element (See Figure 5.10). The core of sintered UO₂, 2-1/2 inches in diameter, is cooled by six 3/8 inch OD x 0.020 inch wall 304 stainless steel tubes spaced 60° apart on a 1-7/16 inch diameter circle. The enclosing tube of ceramic insulation material is 3.0 inch OD and 2-1/4 inch ID and is, in turn, clad in a 3.0 inch ID x 3.060 inch OD Zircaloy-2 tube.

UNCLASSIFIED

-5.11-

HW-58205

FIGURE 5.10
Internally Cooled Only Fuel Element

UNCLASSIFIED

Swaged UO₂ Capsule Irradiation - J. L. Bates

Swaged capsules of natural UO₂ and 2.47 w/o U-235 enriched UO₂ are being irradiated in the MTR and ETR. The capsules^(2,3) are to be irradiated to exposures greater than 10,000 MWD/T with maximum core temperatures near the melting point of UO₂. Approximately 10 per cent of this exposure has been obtained.

Irradiation of Plutonium Enriched, Non-sintered, UO₂ Fuel Element - J. L. Bates and W. E. Roake

A plutonium enriched fuel assembly has successfully completed one irradiation cycle in the MTR, generating approximately 30 KW/ft. The Zircaloy-clad cluster is 11.4 inches long and is made up of three 0.030 inch wall, 0.565 inch OD rods, as shown in Figure 5.11. One rod contained UO₂ 2.47 w/o U-235 enriched UO₂ and two rods contained UO₂ with 1.00 w/o plutonium. The plutonium-bearing rods were prepared by diluting PuO₂-UO₂ mixed crystals with natural UO₂.

FACILITIESCFDO Facilities - R. J. Anicetti

Progress in expanding the Ceramic Fuels facility for future development work and for the PRTR fuel loading is summarized below.

The Sciaky Decatron seam welder for welding the fuel element end closures was installed, checked for operability and accepted. The large hydrogen-atmosphere molybdenum wound furnace for sintering the UO₂ compacts was completed and operated at low temperatures.

The contract was awarded for installation of the 300-ton automatic press and construction of the press enclosure. Excavation started at the end of the quarter.

The 12 KW arc image furnace, tungsten filament furnace, high temperature microscope, and cinemicrographic equipment were installed in a new high temperature research laboratory in the 325 Building. Cinemicrographic studies were initiated of the high temperature behavior of stoichiometric and non-stoichiometric UO₂, UC, and UN in helium, steam, and nitrogen atmospheres.

2. Ibid., HW-57146, p. 5.8.

3. Neidner, R., GEH-3-45, 46, 47, 48; Swaged UO₂ Capsule Irradiation, HW-55622, June 12, 1958 (Unclassified)

UNCLASSIFIED

-5.13-

HW-56205

FIGURE 5.11
Plutonium Enriched Fuel Element

ASC-68 RICHLAND, WASH.

UNCLASSIFIED

FACILITIES AND EQUIPMENTAnalysis of Pressure Piping with Temperature Gradients - K. R. Merckx

An elastic-plastic analysis has been made for a thick-walled tube internally cooled which undergoes cyclic pressure loadings and gamma heating. Improved estimates of the plastic strain per cycle and distribution of residual stresses can be made with this analysis since the analysis allows for the residual stress systems arising from subsequent plastic yielding. A program has been written for the 650 IBM digital computer and numerical calculations were made for conditions which will be encountered in the ETR 6 x 9 core facility. These numerical calculations showed that plastic yielding per cycle and maximum tensile residual stresses increased as pipe thickness increased. The report, HW-57195, "In-Reactor Pressure Tube for the ETR 6 x 9 Core Facility", by K. R. Merckx, describes the method of analysis and results of the numerical calculations. This analysis is applicable for tubing which is not undergoing creep.

In order to determine the stress distribution and strain rates in tubes undergoing secondary creep, equations were derived for these quantities in pressure tubing with logarithmic temperature distributions. Numerical evaluations can be made for creep rates which have either a power or exponential dependence upon stress. The stress and temperature dependence of the above material models for 15 per cent cold-worked Zircaloy-2 were determined with creep data from BMI and example calculations made for NPR operating conditions.

Since neither of the above theoretical approaches can account for cyclic application of temperature gradients on the total creep of pressure tubing, model tests were proposed to simulate actual operating conditions. The models should have the material, temperature drop, time scale and ratios of inner and outer tube diameters the same as the actual tubes. Actual testing of models operating under NPR conditions is being proposed, and design of a suitable testing apparatus has been started by the design section of the Irradiation Processing Department.

ETR Fuel Element Testing Facilities - D. C. Kaulitz

The Reactor Development Division has issued final approval to proceed with installation and operation with experimental samples for the 3 x 3 loop in the ETR. Present planning calls for completion of the facility by October 10th, 1958. The first loading of fissionable material will be charged by October 17th, 1958.

Placement and operation of the x-reactor components for the 6 x 9 facility is complete. The in-reactor portion of the facility had to be redesigned when Hanford obtained full time use of the 6 x 9-C-69 location to replace the half time use of the C-99 location. The redesign is now complete and the in-reactor tube is out on bid. Beneficial use of the facility by January, 1959 is still being attempted.

Drawbench - J. P. Pilger

A drawbench assembly is now near completion in the 306 Building. The unit is capable of a seven ton pull over an effective length of 20 feet. Power is furnished by a hydraulic pump driving a reversible hydraulic motor. The hydraulic motor is coupled to the bench through a 450 to 1 reduction gear. The drawing grip and carriage is manually engaged and disengaged into an endless chain. Figure 6.1 is an illustration of the assembled unit. Some of the outstanding features of the bench are as follows:

1. Versatility. The working length may be extended with only minor modifications to major components.
2. The capacity may be doubled by addition of a few stiffening type cross members to prevent bowing of the main load columns.
3. Very high torque drive at low speeds.
4. Infinitely variable drawing speed from zero to approximately 20 feet per minute.
5. Applicability of drawing dies, sizing dies, expanding plugs, turks head operations, "rotary swaging", and "uniskanning" operations with a rotary die attachment head.

UNCLASSIFIED

-6.4-

HW-58205

FIGURE 6.1
Seven-Ton Drawbench

ACC-82 RICHARD, WASH.

UNCLASSIFIED

DATE

FILMED

8/17/94

END