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SOLID STATE DIVISION

ANNUAL PROGRESS REPORT

FOR PERIOD ENDING AUGUST 31, 1957

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C-84 – Reactors–Special Features of Aircraft Reactors

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SOLID STATE DIVISION

ANNUAL PROGRESS REPORT

For Period Ending August 31, 1957

D. S. Billington, Director
J. H. Crawford, Jr., Assistant Director

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CONTENTS

RADIATION METALLURGY 1

LITR Creep Tests 1

MTR Creep Tests 1

Stress Corrosion 2

Resistivity of Inconel 2

ENGINEERING PROPERTIES 2

Reactor Shield Materials 2

Boron Nitride 3

Boron Carbide 4

Cermet Shielding Materials 4

REACTOR EXPERIMENTS 13

Effect of Radiation on Static Corrosion of Structural Materials by Fused Salt Fuels 13

ANP LITR Vertical Loop 13

Effect of Radiation on Chemical Analysis of Chromium, Iron, and Nickel in ANP Fluoride Fuel 14

ETR Irradiation of High-Temperature Moderator Materials 14

HOT CELL OPERATIONS 14

[redacted]

[redacted]

SOLID STATE DIVISION ANNUAL PROGRESS REPORT

RADIATION METALLURGY

J. C. Wilson

Tube-burst tests of Inconel in a helium atmosphere in the LITR showed no shortening of the rupture life attributable to irradiation when high-quality tubing was used for specimen stock; earlier data, obtained with lower quality tubing, indicated that LITR irradiation shortened the rupture life by a factor of 3.

An eight-specimen tube-burst test was operated in the MTR. The rupture times were considerably shorter than was expected on the basis of out-of-pile data, but interpretation must await the completion of bench tests that duplicate the in-pile stress-time-temperature cycle.

LITR CREEP TESTS

W. W. Davis J. C. Zukas

A reduction in the rupture life of Inconel tubes stressed by internal pressure in a helium atmosphere was reported earlier.[1] In these tests, drawn tubing with a nominal 0.01-in. wall was used. Nondestructive testing of sections of the tubing stock had indicated that detectable defects existed in the tubing, although very few of the indicated defects could be found under the microscope. Another series of tests was run with the use of a carefully inspected lot of tubing which gave no indications of defects. The wall thickness of the test section was ground down to 0.010 in. from the as-drawn value of 0.025 in. Tests were run in hole HB-3 of the LITR at 2000 psi and 1500°F in helium. Rupture times of 1050 and 1650 hr were recorded (a third specimen had not ruptured at 1675 hr), in good agreement with data obtained in the Metallurgy Division.[2] The corresponding bench tests had rupture times of over 2000 hr. The previously reported rupture times for the tubing that showed defects in nondestructive testing were 260 hr for the irradiated and 760 hr for the unirradiated specimen. From the limited data, it appears that irradiation may be effective in reducing the rupture life of defective tubing (defective in the sense that numerous indications of defects showed up in nondestructive testing); but, in tubing of known integrity, there was no substantial effect of irradiation at the flux level of the tests.

MTR CREEP TESTS

W. E. Brundage W. W. Davis
N. E. Hinkle

A tube-burst creep test on Inconel was operated in the MTR.[3] Eight specimens, at three stress levels and two different wall thicknesses, were tested simultaneously. The rupture times observed[4] were almost a factor of 12 shorter than had been reported in bench tests by the Metallurgy Division.[2] A bench test in the same type of apparatus as that used for the in-pile tests is now in progress. The drastic shortening of the rupture time in the MTR was not expected in view of the LITR results reported above and an earlier MTR creep test (in tension) at a lower stress.[5] Electrical resistivity results from the in-pile apparatus indicated that leakage of thermocouple signals through irradiated insulation could not have caused a temperature error large enough to account for the short rupture life. Other insulation tests in the LITR substantiate this conclusion. Overheating under partial stress at the start of irradiation (because of a high level of gamma heating at startup of the reactor) does not seem to have had a strong influence, inasmuch as Douglas and East of the Metallurgy Division ran tests in their apparatus (in which the specimen is isolated in a metallic chamber rather than being in the chamber with the furnace, as in the in-pile rig) which indicated that the time-stress-temperature cycle of the in-pile test did not shorten the rupture life appreciably.

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[1] W. W. Davis *et al.*, *Solid State Semiann. Prog. Rep. Aug. 30, 1956*, ORNL-2189, p 26.

[2] J. W. Weir, *Met. Semiann. Prog. Rep. Oct. 10, 1955*, ORNL-1988, p 36.

[3] J. C. Wilson *et al.*, *ANP Quar. Prog. Rep. March 31, 1957*, ORNL-2274, p 248.

[4] J. C. Wilson *et al.*, *ANP Quar. Prog. Rep. June 30, 1957*, ORNL-2340, p 268.

[5] J. C. Wilson *et al.*, *ANP Quar. Prog. Rep. Sept. 10, 1956*, ORNL-2157, p 241.

[redacted]

It is possible that contaminants in the helium (such as sulfur) could shorten the rupture life (air leaks would tend to lengthen it). The specimens have been removed from the irradiation apparatus for determination of the location of rupture and the state of the surface of the specimens.

A modified MTR apparatus is being bench tested; in it the specimens are oriented in such a way as to minimize the gamma heating gradients that have made temperature control so difficult.

STRESS CORROSION

J. C. Zukas W. W. Davis

The apparatus for carrying out in-pile tube-burst tests on Inconel in an environment of molten fuel salts is still not satisfactory. The last three tests have had to be discontinued because of leaks after less than 100 hr of irradiation. The source of the leaks is being sought with the aid of more complete instrumentation of the next apparatus and examination in the hot cells of the apparatus which showed failure. Thermal cycling of connectors or of the brazed joints is regarded as the most likely cause of leaks. During the tests which failed it has been possible to improve the temperature control to ±5°F, although during reactor operation the power density in the fuel in contact with the specimen may be as high as 420 watts per cubic centimeter.

RESISTIVITY OF INCONEL

T. C. Price J. C. Wilson

The electrical resistivity of several lots of Inconel pipe was determined in order to aid in the design and calibration of electromagnetic flow meters for the Aircraft Reactor Engineering Division.[6] The resistivity was measured from room temperature up to about 1600°F, and it was necessary to measure the resistivity as a function of time in the region between 600 and 1200°F because ordering of the alloy in this range can cause a resistivity change of several per cent.

An apparatus is under construction to test the resistivity changes that may be caused by irradiation. Such measurements are necessary because an ART level gage fabricated from Inconel will change calibration if the resistivity of the alloy is affected by irradiation.

[6] J. C. Wilson *et al.*, *ANP Quar. Prog. Rep. Dec. 31, 1956*, ORNL-2221, p 299.

ENGINEERING PROPERTIES

O. Sisman

Boron-containing thermal-neutron shield materials were studied under irradiation at various temperatures and neutron dosages. Stainless steel–clad cermets were tested, as well as hot-pressed boron nitride. At the higher B^{10} burnups, the Cu–B_4C cermets suffered cracking and separation of cladding from core. A cermet of BN–Ni clad with stainless steel was found to have cracked to a greater extent after 38% B^{10} burnup than CaB_6–Fe.

REACTOR SHIELD MATERIALS

J. G. Morgan M. T. Morgan
P. E. Reagan

Neutron shield materials containing boron have a high potential for radiation damage because of the (n,α) reaction. Effectively, this amounts to a fission of the boron atom into an atom of helium and an atom of lithium. Lithium-6 similarly produces helium and tritium. In such metals as cadmium, where the (n,α) reaction is not present, the rate of damage is much less and originates from energetic gamma recoils.

The most important advantage of using a boron-containing material is the low secondary gamma-ray intensity at the outside of the reactor shield, an advantage which is especially important in mobile reactors. Boron is easily fabricated in the form of metal dispersions, alloys, or ceramics and has a neutron cross section that extends into the epithermal region. Two types of boron-containing shielding materials were studied under irradiation.

Boron Nitride

Boron nitride is similar to graphite in many respects, but it is white in color and much more resistant to oxidation at high temperatures. Favorable results have been obtained from radiation damage tests conducted on boron nitride as a neutron shielding material. The tests were designed primarily to explore the general integrity of this relatively new ceramic, and specifically to determine changes in weight, density, and dimensions.

Boron nitride was obtained from two manufacturers (the Norton Company and the Carborundum Company) in powder form, and test samples $\frac{3}{16}$ in. OD × $1\frac{1}{2}$ in. long were hot-pressed from the powders. Norton samples are designated 6002 and Carborundum samples are designated C1.

The samples were irradiated in an inert atmosphere either in quartz ampoules or in Inconel capsules. High-temperature irradiations were made in the Inconel capsules, and the quartz-enclosed samples were irradiated at the temperature of the test-reactor water. The high-temperature tests were monitored by thermocouples pressed against the sample with leads extending out through a ceramic seal in the capsule top. Thermal-neutron dosages were calculated from the activity of a cobalt foil located against the sample.

Surface darkening of the samples during irradiation is thought to be caused by a metallic deposit. Analysis indicates that this coating consists primarily of magnesium and silicon with a trace of calcium and copper.

The irradiated samples retained good crystallinity. X-ray diffraction results from MTR samples are presented in Table 1.

Test conditions and physical property changes are presented in Table 2. Comparison between irradiated and unirradiated samples is shown in Figs. 1 and 2. The darker samples are the irradiated materials. The sample in Fig. 1 was broken in handling after irradiation.

Dimensional changes were small in these tests. The changes that were detected could not be definitely attributed to radiation damage, for boron nitride is a soft material (Mohs'-scale hardness of 2 before irradiation increased to 2.5 after irradiation) and the micrometers scraped material from the surface of the sample during measurements. Both weight and density changes were less than 1%.

The burnup values shown in Table 2 are average values obtained from calculations based on neutron flux and irradiation time. Isotopic analysis of the boron is now in progress.

Within the scope of these tests, boron nitride as a neutron shielding material has remained

Table 1. X-Ray Diffraction Results on Irradiated Boron Nitride

Index	Interplanar Spacing (Å)				
	Original Powder	Unirradiated Hot-Pressed Samples*	6002-1**	6002-4 (Inside)	6002-4 (Outside)
002	3.32	3.33	3.37	3.33	3.35
100	2.17	2.17	2.17	2.17	2.17
101		2.06	2.06	2.06	2.07
102	1.81	1.82	1.82	1.82	1.82
004		1.66	1.68	1.66	1.67
104		1.32		1.32	
110		1.25	1.25	1.25	1.25
112		1.17	1.17	1.17	1.17

*Hexagonal, a = 2.50 Å, c = 6.64 Å.
**Hexagonal, a = 2.50 Å, c = 6.74 Å. The expansion in c is 1.6%.

Table 2. Irradiation History of Hot-Pressed Boron Nitride

Sample No.	Test Facility	Temperature (°F)	Time (hr)	Flux (thermal neutrons/cm^2·sec)	Dosage (thermal neutrons/cm^2)	Weight Change (%)	Density Change (%)	Dimensional Change (%)	Average B^{10} Burnup (%)
6002-1	MTR, A-28	1600	710	5×10^{13}	1.3×10^{20}	0.9	None	None	6.0
6002-2	LITR, C-39	70	1674	1.7×10^{13}	1.02×10^{20}	*	*	*	4.8
6002-3	LITR, C-39	70	1675	1.7×10^{13}	1.02×10^{20}	*	*	*	4.8
6002-4	MTR, A-28	1780	606	4×10^{13}	8.72×10^{19}	**	0.13	0.1	4.0
6002-8	Bench test	1470	600			0.26	0.77	None	
C1-11	LITR, C-39	70	1148	1.7×10^{13}	7.02×10^{19}	*	*	*	3.8
C1-12	LITR, C-39	70	1148	1.7×10^{13}	7.02×10^{19}	*	*	*	3.8

*Physical properties not yet determined.
**Not measured because of loss of chips when sample was broken.

Fig. 1. Hot-Pressed Boron Nitride Irradiated at 1600°F to 6% B^{10} Burnup. (Confidential with caption)

undamaged when irradiated at 1800°F to a dosage of 1.3×10^{20} thermal neutrons per square centimeter.

Four gas-evolution samples have been irradiated and they also will be used for physical property measurements.

Boron Carbide

Two irradiations of hot-pressed boron carbide ART prototype tile samples have been completed. The sample designated J-1 was irradiated in the LITR C-39 facility for 1148 hr at reactor-water temperature to a calculated average burnup of 1.4%. Sample J-3 was irradiated in the LITR C-42 facility for 910 hr at 1500°F to 2.2% average burnup. Postirradiation examination is in progress.

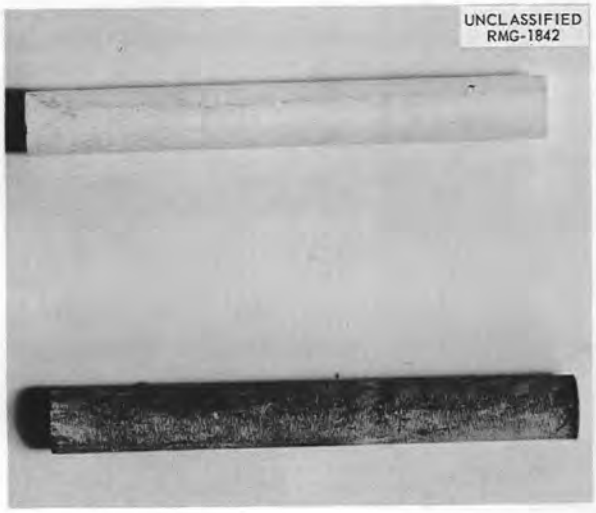

Fig. 2. Hot-Pressed Boron Nitride Irradiated at 1780°F to 4% B^{10} Burnup. (Confidential with caption)

A bench test was conducted on a third sample, J-2, by cycling it from 200°F to 1600°F for 18 cycles. The sample was examined under a microscope (20X, 40X) and no cracking was observed.

Cermet Shielding Materials

Composite plates of BN–Ni clad with stainless steel[1] have been irradiated at <345°F in evacuated quartz ampoules cooled by the reactor water and examined after two different burnups. Plates of CaB_6–Fe clad with stainless steel[1]

[1] J. G. Morgan *et al.*, *Solid State Semiann. Prog. Rep. Aug. 30, 1956*, ORNL-2189, p 32–34.

were similarly irradiated and examined. The boron in both materials was natural boron.

There were no structural changes as a result of irradiation to 19% average B^{10} burnup in either the CaB_6–Fe or BN–Ni material. When the irradiation was increased to 38% average B^{10} burnup, both materials suffered damage (see Table 3). The CaB_6–Fe showed slight general cracking throughout the core matrix (see Fig. 3). The BN–Ni material was more severely cracked, with

Table 3. Dimensional Stability and Hardness Changes upon Irradiation

Cermet	Length Change (% increase)	Width Change (% increase)	Thickness Change (% increase)	Density Change (% decrease)	Increase in Hardness Number	
					Core	Stainless Steel Cladding
			19% B^{10} Burnup			
CaB_6–Fe[a]	<0.1	<0.2	<0.3	<0.2	1.0[b]	25[c]
BN–Ni[a]	<0.1	<0.2	<0.3	<0.2	100[c]	25[c]
			38% B^{10} Burnup			
CaB_6–Fe[d]	0.8	1.6	2.1	1.2	36.0[b]	55.0[c]
BN–Ni[d]	0.4	1.0	5.4	1.0	204.0[c]	47.6[c]

[a]Typical sample.
[b]Diamond pyramid hardness, 4-kg load.
[c]Knoop hardness number, 1-kg load.
[d]Average of three samples.

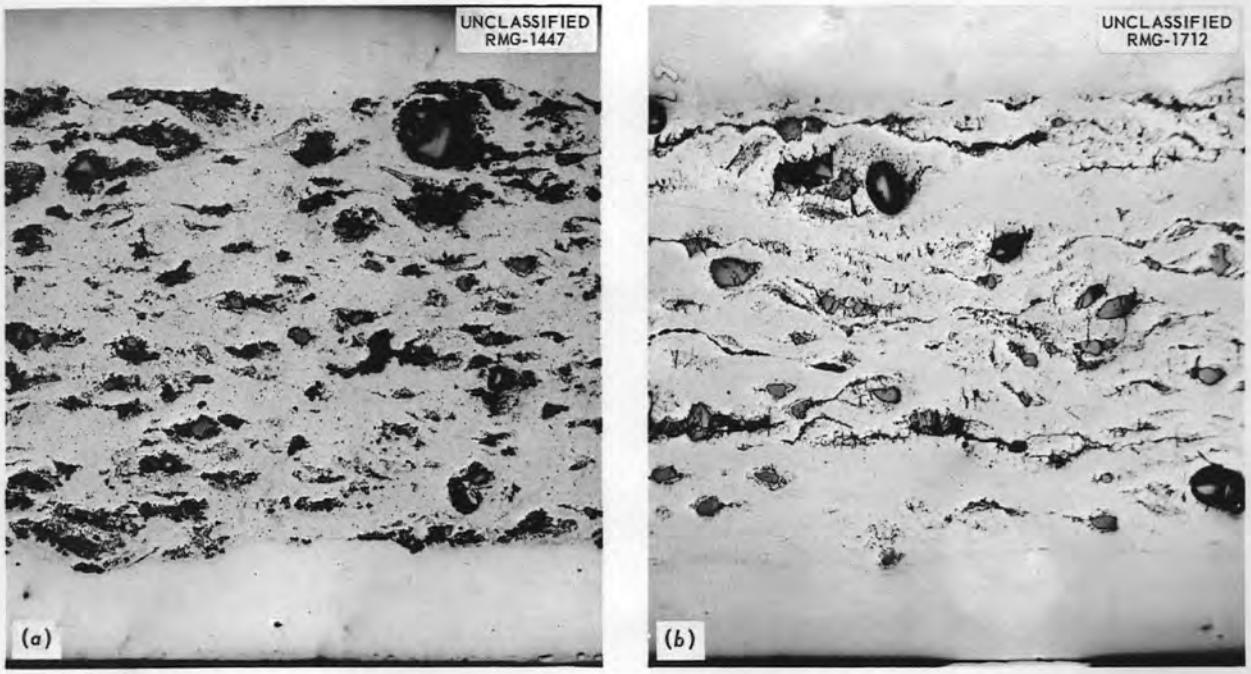

Fig. 3. Irradiated CaB_6–Fe Shielding Element. (*a*) 19% average B^{10} burnup. (*b*) 38% average B^{10} burnup. 75X. Reduced 13%. (Confidential with caption)

some separation of the cladding from the core. No significant change in the structure of the core or cladding was noted. There was no noticeable increase in fragmentation or porosity. Figure 4 is a macroscopic view showing the longitudinal splitting of the sample. Figure 5 is a comparison of the two irradiated samples at higher magnification.

Another clad cermet material that was studied contained B_4C dispersed in copper. The first three specimens for irradiation were fabricated by the Metallurgy Division. They were 0.5 in. long × 0.1875 in. wide × 0.102 in. thick. The core of these specimens was 0.082 in. thick (Cu–6.6 wt % B_4C), and was clad on two faces with 0.003 in. of copper followed by 0.007 in. of type 430 stainless steel. The core was exposed on the four edges of the specimens. The fourth irradiation contained one similar sample, 1.5 in. long, fabricated by the Allegheny Ludlum Steel Corp. The irradiation history of these samples is listed in Table 4.

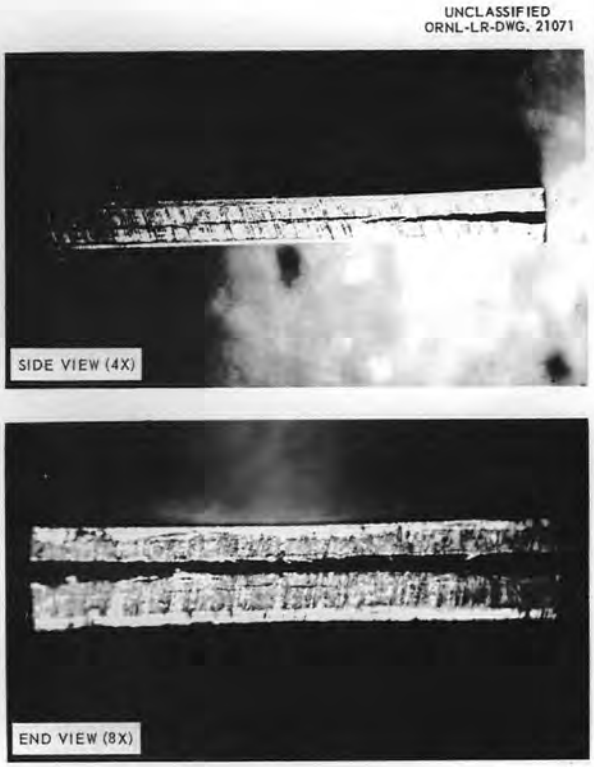

Fig. 4. Specimen of BN–Ni Clad with Type 304 Stainless Steel, Irradiated at <175°C to Estimated 38% Average B^{10} Burnup. (Confidential with caption)

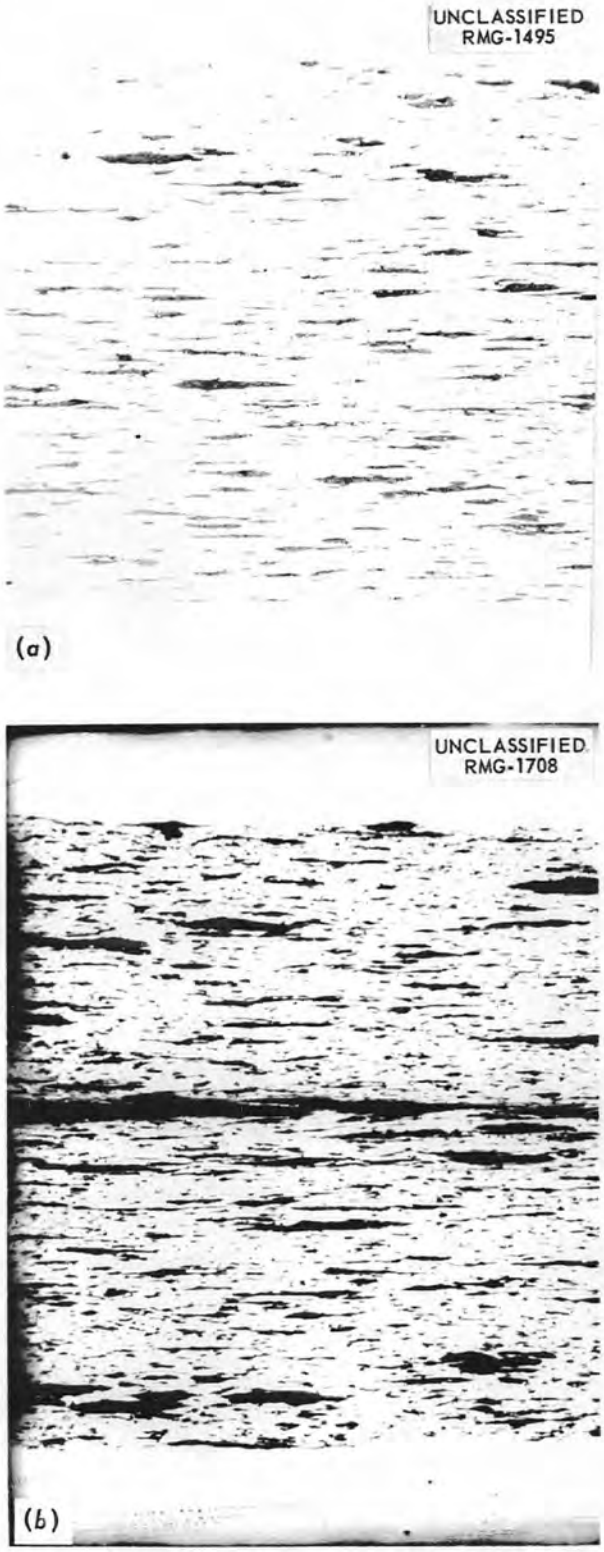

Fig. 5. Irradiated BN–Ni Shielding Element. (*a*) 19% average B^{10} burnup. (*b*) 38% average B^{10} burnup. 75X. (Confidential with caption)

Hardness measurements were made on the stainless steel cladding, copper cladding, and core after irradiation. The postirradiation hardness measurements on the core are not reported for samples which suffered severe cracking. Table 5 summarizes the results of the hardness measurements. Table 6 gives dimensional stability measurements. While damage to these samples appears to be a function of burnup and temperature of irradiation, the thermal cycling history is also important. In a composite plate of this type (essentially black to neutrons) most of the (n,α) reactions occur in the first few thousandths of an inch of sample thickness. The core becomes harder and more brittle, while the stainless steel cladding acts as a bimetallic strip, causing uneven stresses. This is believed to account for a large part of the splitting and warping.

Figure 6 is a photomicrograph of a Cu–B_4C sample irradiated at 200°C to 6% average B^{10} burnup. No cracking is evident. Figure 7 shows cracking and separation of cladding from core which occurred at a higher irradiation temperature and higher burnup.

Table 4. Irradiation History of Stainless Steel–Clad Cu–B_4C Shield Material

Irradiation Number	Number of Samples	Average Temperature of Irradiation (°C)	Average B^{10} Burnup (%)	Number of Temperature Cycles	Damage
227	3	200	6*	3	No cracking or separation of cladding from core
327	3	300	19*	7	No cracking or separation of cladding from core
127	3	420	18**	19	Longitudinal core cracking and core-cladding separation
527	1	870	12.5*	8	Longitudinal core cracking and core-cladding separation

*Calculated.
**Determined by isotopic analysis.

Table 5. Effect of Irradiation on Hardness of Stainless Steel–Clad Cu–B_4C Shield Material

Irradiation Number	Component	Average Knoop Hardness Number, 1-kg Load		Increase After Irradiation (%)
		Unirradiated	Irradiated	
227	Stainless steel cladding	139	166	19
	Copper cladding	64	76	19
	Core	76	102	34
327	Stainless steel cladding	139	215	55
	Copper cladding	64	88	38
	Core	76	85	12
127	Stainless steel cladding	139	186	34
	Copper cladding	64	65	2

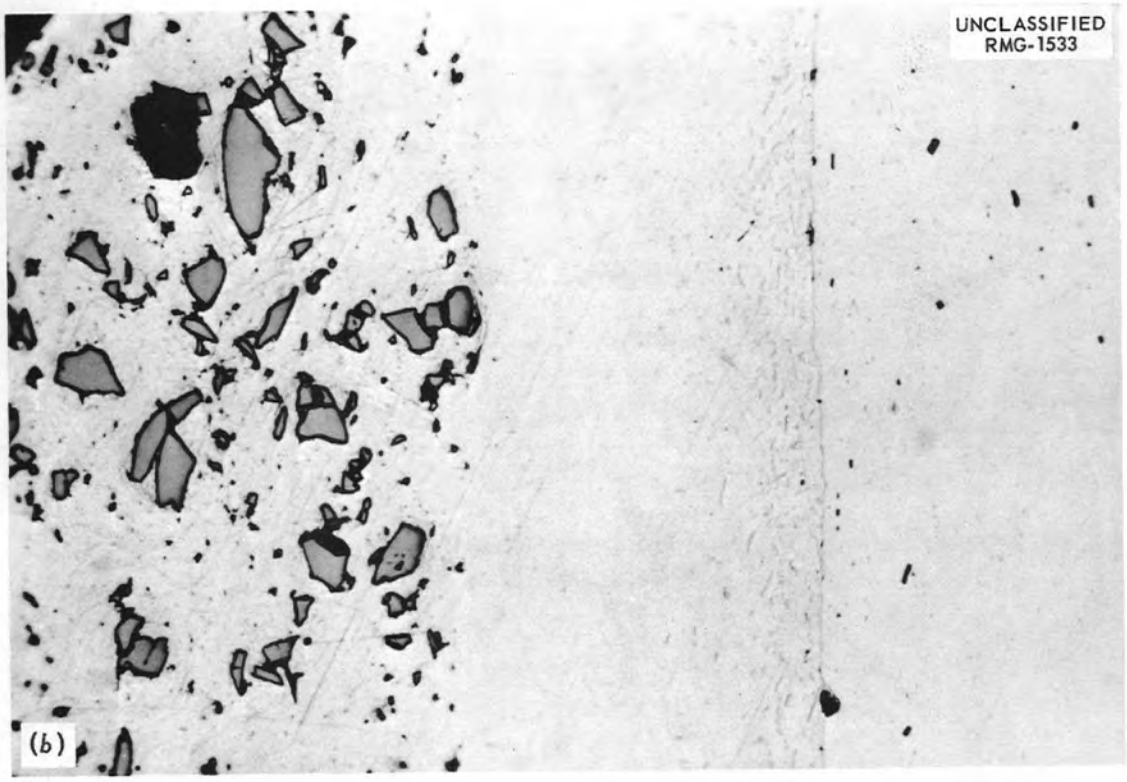

Fig. 6. Core-Cladding Interface, Cu–B_4C Element. (*a*) Unirradiated. (*b*) Irradiated. 500X. (Confidential with caption)

Table 6. Effect of Irradiation on Dimensions of Stainless Steel–Clad Cu–B_4C Shield Material

Irradiation Number	Number of Samples	Average Change (%)			
		Length	Width	Thickness	Density
227[a]	3	< 1	< 1	< 1	< 1[b]
327	3	< 1	+ 1.8	+ 2.0	− 3.0[c]
127	3	+ 0.5	+ 0.8	+ 2.3	− 2.3[b]
527	1	+ 2.5	+ 3.5	+ 12.1	− 11.6[b]

[a]Samples were slightly oxidized, which resulted in data scattering. There was essentially no change after irradiation.

[b]By displacement.

[c]Apparent density.

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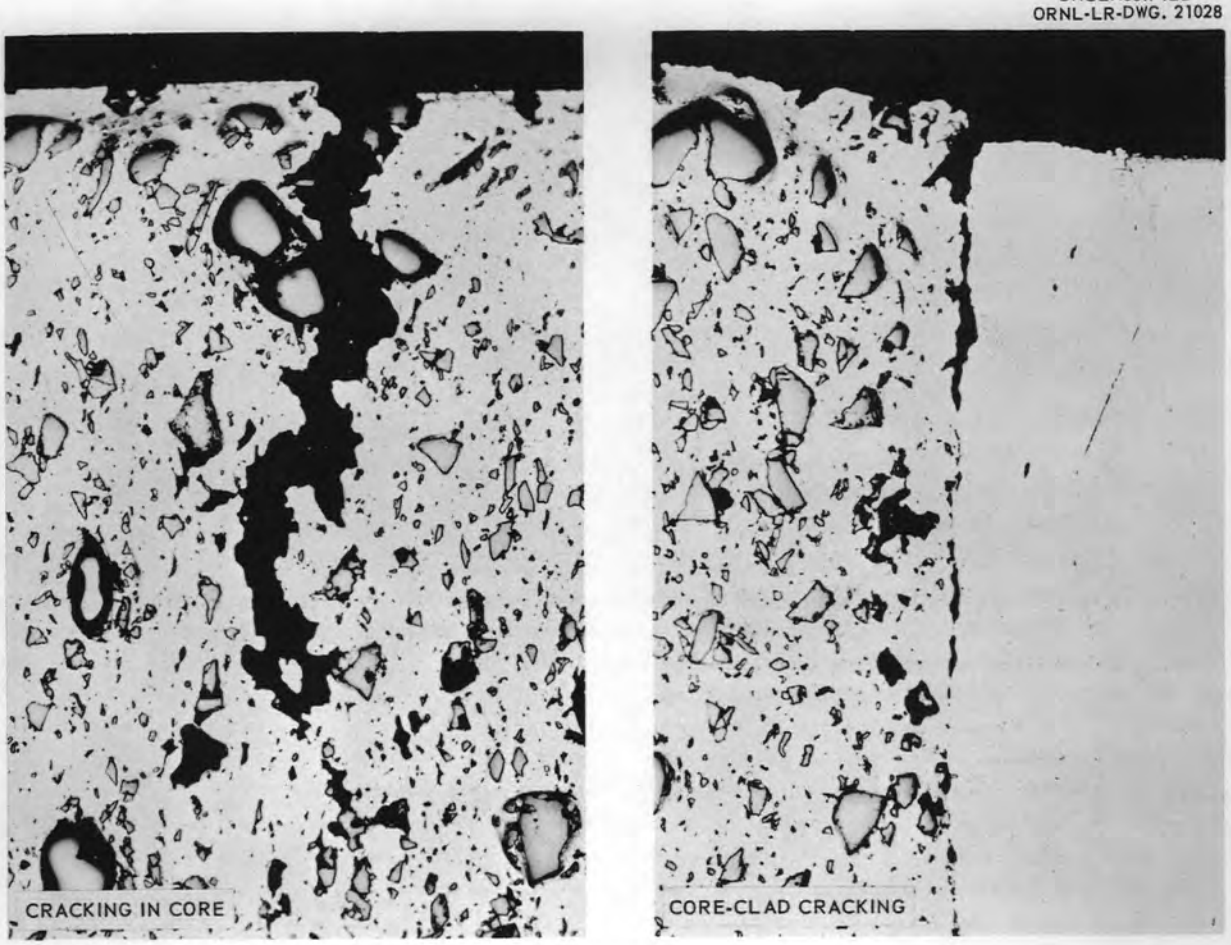

Fig. 7. Specimen of Cu–B_4C Irradiated at 420°C to 18% Average B^{10} Burnup. 250X. Reduced 10%. (Confidential with caption)

That damage due to thermal stress occurs in the absence of irradiation was confirmed by a laboratory experiment, in which the samples were cycled under a vacuum by immersion into and withdrawal from a furnace. The heating and cooling rates were as follows:

Temperature Range	Heating Rate	Cooling Rate
75–600°C	300°C/min, first min	250°C/min, first min
	175°C/min, second min	200°C/min, second min
100–870°C	960°C/min, first 30 sec	600°C/min, first 30 sec
	400°C/min, second 30 sec	200°C/min, second 30 sec
	75°C/min, second min	150°C/min, second min

One cycle consists in a temperature excursion from minimum to maximum and return to minimum. The results are shown in Table 7 and Figs. 8–10.

In the laboratory experiment, more thermal cycles were required in order to cause damage; however, the irradiated samples had experienced a different thermal gradient history, and this may account for the change with fewer cycles.

Table 7. Thermal Cycling of Cu–B_4C Tiles

Test Number	Sample	Dimensions (in.)	Number of Cycles	Temperature Range (°C)
1	Norton B_4C prototype tiles	0.25 × 0.25 × 1.0	18	100–870
2	ORNL Cu–B_4C clad with copper and stainless steel	0.1 × 0.185 × 0.5	20	75–600
3	ORNL Cu–B_4C clad with copper and stainless steel	0.1 × 0.18 × 0.5	20 20	75–600 100–870
4	ORNL Cu–B_4C clad with copper and stainless steel	0.1 × 0.185 × 0.5	20 38	75–600 100–870
5	ORNL Cu–B_4C clad with copper and stainless steel	0.1 × 0.185 × 0.5	20 58	75–600 100–870
6	Allegheny Ludlum Cu–B_4C clad with copper and stainless steel	0.1 × 0.189 × 1.5	58	100–870

Test No. 1. – The Norton B_4C showed no evidence of cracking or chipping under 500X examination.

Test No. 2. – The ORNL Cu–B_4C showed no evidence of cracking during this test. Figure 8 shows the core-cladding interface. The scratches resulted from polishing by hand. It was not desirable to mount the samples and break them out of the Bakelite for further testing.

Test No. 3. – One small crack (about 6 mils long) was noted at the interface between the stainless steel cladding and the copper.

Test No. 4. – The sample experienced complete separation of the cladding over about one-third of its area on one face. The separation occurred between the stainless steel and the copper claddings (Fig. 9). In addition, core cracking was noticeable on the ends of the samples about midway between the clad faces (Fig. 10).

Test No. 5. – The sample showed only slightly more core-cladding separation and core cracking.

Test No. 6. – In general, the Allegheny Ludlum Cu–B_4C did not suffer as severe core-cladding separation or core cracking. It was noted that, if small cracks were present in the material, they tended to propagate and open up as the result of cycling. This propagation was sometimes from the core-cladding interface and down into the core itself. The sample was not constrained during cycling. After the test the entire sample was bowed about $\frac{1}{32}$ in. in the middle, apparently due to unequal stresses caused by the bimetallic cladding.

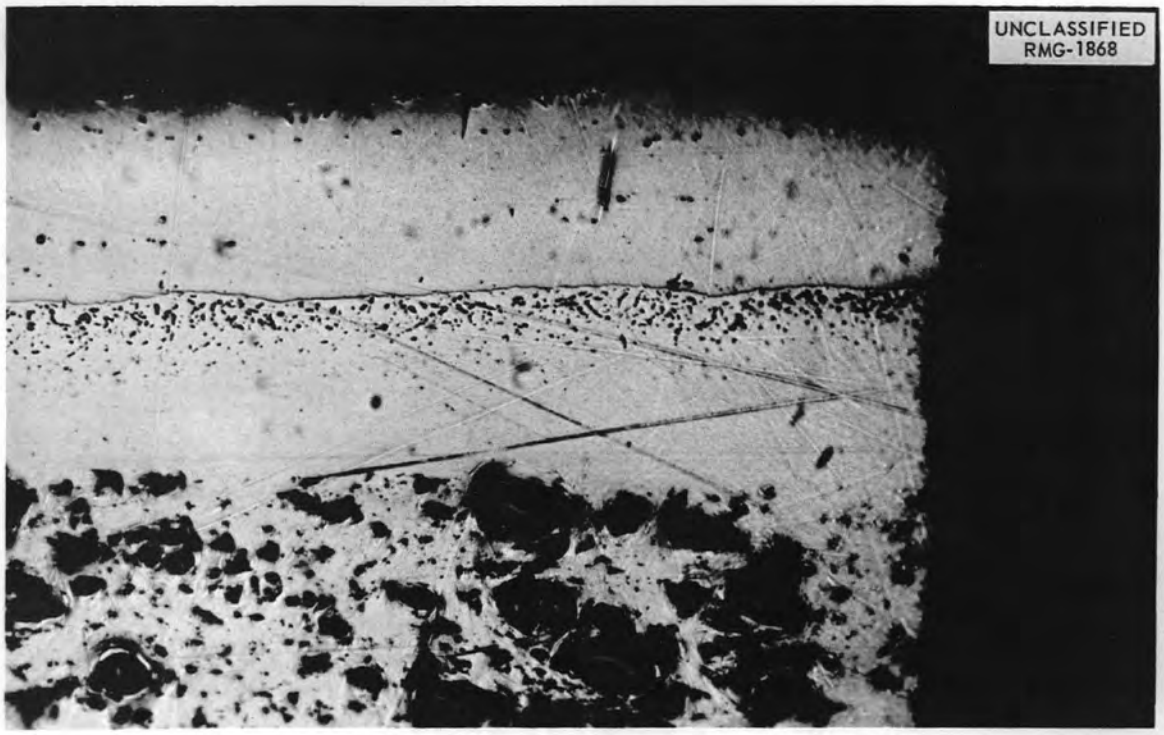

Fig. 8. Unirradiated Stainless Steel–Clad Cu–B_4C Specimen After Thermal Cycling – Test No. 2. 250X. (Confidential with caption)

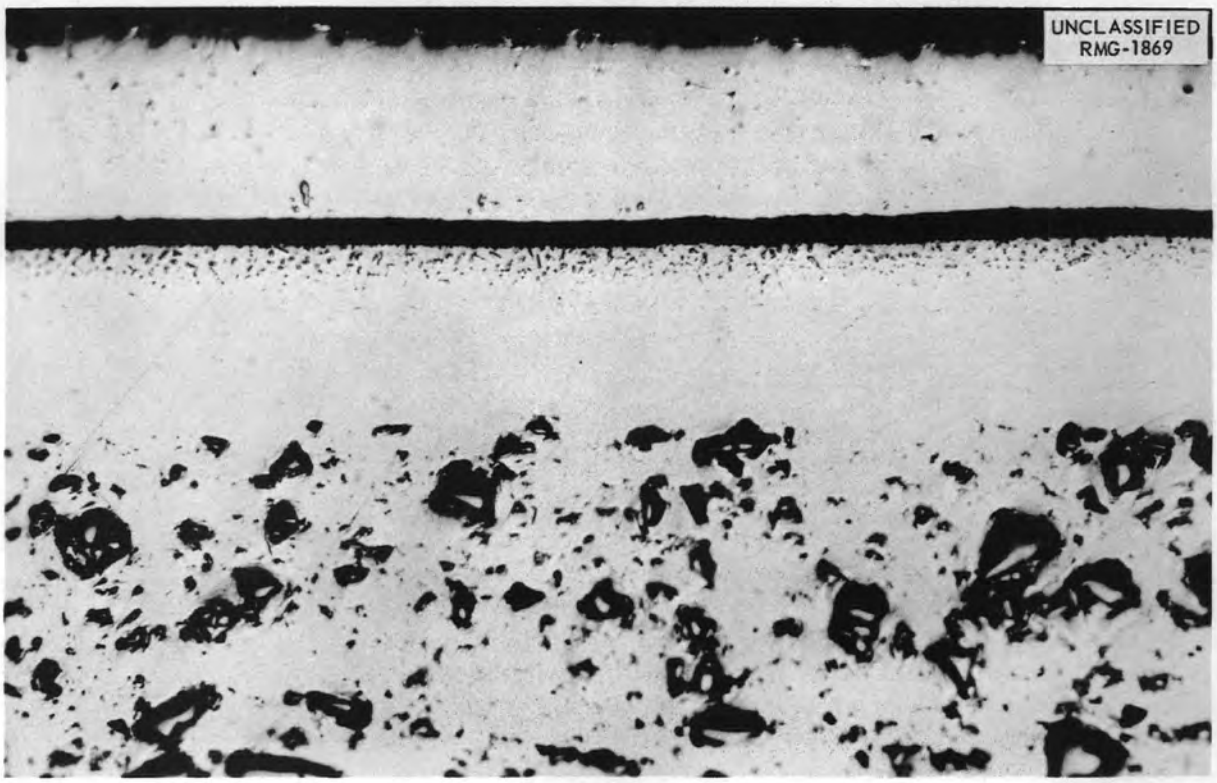

Fig. 9. Unirradiated Stainless Steel–Clad Cu–B_4C Specimen After Thermal Cycling – Test No. 4. 250X. (Confidential with caption)

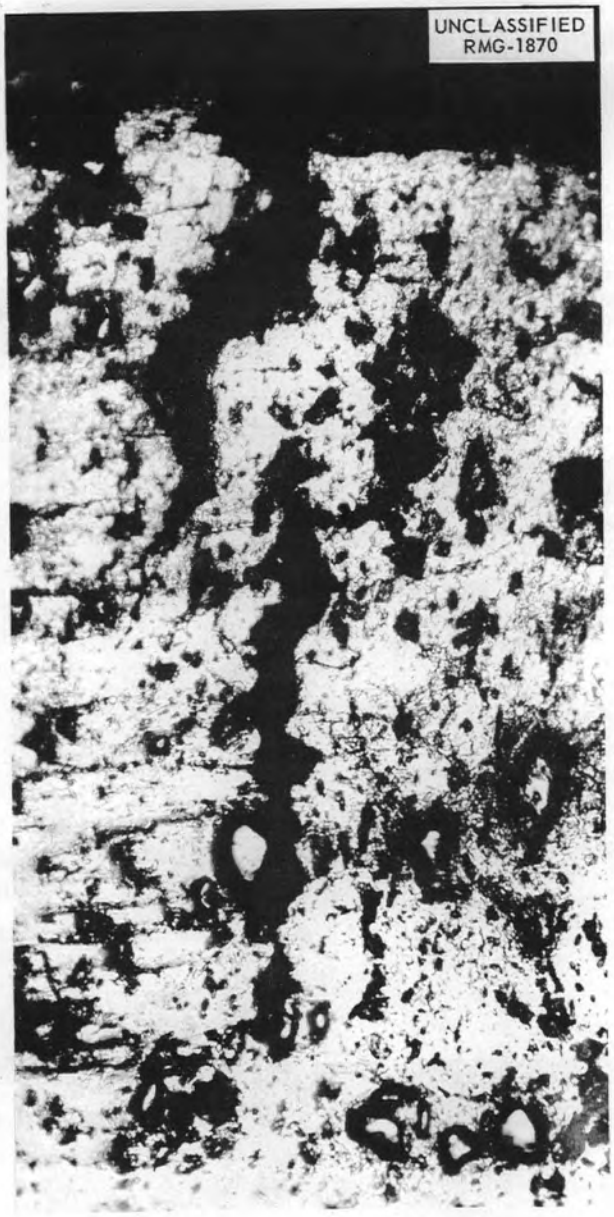

Fig. 10. Unirradiated Stainless Steel–Clad Cu–B_4C Specimen After Thermal Cycling – Test No. 4. 250X. (Confidential with caption)

REACTOR EXPERIMENTS

G. W. Keilholtz

The study was continued on the effect of radiation on static corrosion of structural materials by ANP fused fluoride fuels. The results indicate that corrosion of Inconel by fluoride fuels is not affected by exposure to radiation. An in-pile loop containing ANP fluoride fuel was operated at 1600°F in the LITR for 235 hr with the reactor operating. The experiment was terminated when radiation damage to an organic lubricant in a bearing interfered with operation of the pump. The loop is being examined in the Solid State Division hot cells for evidences of corrosion. The study was continued on the effect of radiation on chemical analysis of fluoride fuels. As much as 13% error was observed in analysis of chromium at radiation levels of 3000 r/hr. Work is being continued on the analysis of chromium as well as of nickel and iron. An in-pile experiment is being designed for a study of the effect of radiation on new high-temperature moderator materials in the ETR.

EFFECT OF RADIATION ON STATIC CORROSION OF STRUCTURAL MATERIALS BY FUSED SALT FUELS

W. E. Browning
H. L. Hemphill
R. E. Adams
J. E. Lee, Jr.

The study of the effect of radiation on the corrosion of structural materials by fused salt fuels has been continued. Ten capsules of Inconel or Hastelloy B containing fluoride fuels were irradiated in the MTR for irradiation times up to six weeks at 1500°F. Twelve capsules have been cut open in the Solid State Division hot cells and examined for evidence of corrosion. Three control capsules have been given heat treatments simulating in-pile conditions and examined metallographically for evidence of corrosion. These tests indicate that corrosion of Inconel by fluoride fuels is not affected by exposure to radiation. Details of these tests have been reported elsewhere.[1-3]

ANP LITR VERTICAL LOOP

W. E. Browning
H. E. Robertson
R. E. Adams
M. F. Osborne
R. P. Shields
D. E. Guss
J. E. Lee, Jr.

The improved pump design previously described[4-7] was tested in the laboratory under conditions as nearly like those in the reactor as possible. The Inconel pump was operated for 3525.5 hr at 1570°F with $NaF-ZrF_4-UF_4$ (50-46-4 mole %, fuel 30). The performance of the pump in this test was satisfactory in all respects. The test was terminated when the supply of cooling water was interrupted by failure of a water main. A pump of the same design was incorporated into a miniature in-pile loop which was operated in the LITR for 235 hr with the reactor operating. The test was run at 1600°F with $NaF-ZrF_4-UF_4$ (60.4-27.4-11.8 mole %, fuel 41). The maximum axial temperature difference was 250°F. The Reynolds number for the fuel was 3000. The dilution factor was 7.3. The maximum power density was 580 watts per cubic centimeter and the total power was 9000 watts. The experiment was terminated when radiation damage to an organic lubricant in a bearing interfered with operation of the pump.

Almost all the components of the experiment performed satisfactorily throughout the test; however, some of the thermocouples were damaged by differential thermal expansion of the fuel tube as fission power was applied, and some heater leads failed during rapid temperature changes. The loop has been disassembled in the Solid State Division hot cells. Examination of the pump revealed that performance was satisfactory except for the ultimate bearing failure.

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[1] W. E. Browning, R. E. Adams, and H. L. Hemphill, *ANP Quar. Prog. Rep. Dec. 31, 1956*, ORNL-2221, p 300.

[2] W. E. Browning *et al.*, *ANP Quar. Prog. Rep. March 31, 1957*, ORNL-2274, p 250.

[3] W. E. Browning *et al.*, *ANP Quar. Prog. Rep. Sept. 10, 1957*, ORNL-2387.

[4] W. E. Browning, M. F. Osborne, and H. E. Robertson, *ANP Quar. Prog. Rep. Dec. 31, 1956*, ORNL-2221, p 299.

[5] W. E. Browning *et al.*, *ANP Quar. Prog. Rep. March 31, 1957*, ORNL-2274, p 254.

[6] W. E. Browning *et al.*, *ANP Quar. Prog. Rep. June 30, 1957*, ORNL-2340, p 275.

[7] W. E. Browning *et al.*, *ANP Quar. Prog. Rep. Sept. 10, 1957*, ORNL-2387.

[redacted]

Examination of the thermocouples showed that their failure was due to mechanical deformation such as would result from differential thermal expansion of the fuel tube. The heater failures were traced to the electrical leads inside the experimental assembly. Work is under way to eliminate in future loops the conditions which led to failures of these components. Metallographic examination of the fuel tube, radioassay and chemical analysis of the fuel itself, and examination of the fuel container system are in progress. This work is reported in further detail elsewhere.[4-7]

EFFECT OF RADIATION ON CHEMICAL ANALYSIS OF CHROMIUM, IRON, AND NICKEL IN ANP FLUORIDE FUELS

R. P. Shields R. E. Adams
W. E. Browning

A study has been continued of the effect of radiation on the chemical analysis of ANP fluoride fuels. Spectrophotometric measurements of chromium concentration by the diphenylcarbazide method were performed while the samples were being irradiated with beta radiation at levels up to 3000 r/hr. A correlation between analytical result and irradiation level was observed amounting to as much as 13% error in the analysis after a 60-min exposure to the radiation field. The results of this test have been reported in more detail.[8] Work is being continued on the analysis of nickel and iron in fluoride fuels. Samples of fuel having known concentrations of these elements have been irradiated and are being used for these tests.

ETR IRRADIATION OF HIGH-TEMPERATURE MODERATOR MATERIALS

W. E. Browning R. P. Shields

Preparations are being made for irradiation of high-temperature moderator materials in the ETR. Conceptual design of the experiment has been developed in collaboration with members of the Reactor Projects Division. Mockup tests have been conducted in the laboratory to simulate heat transfer conditions in the experiment during irradiation and to test assembly techniques. Materials to be tested include beryllium oxide, zirconium hydride, and yttrium hydride. This work has been reported in greater detail elsewhere.[9]

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[8] R. P. Shields and R. E. Adams, *ANP Quar. Prog. Dec. 31, 1956*, ORNL-2221, p 305.

[9] W. E. Browning and R. P. Shields, *ANP Quar. Prog. Rep. June 30, 1957*, ORNL-2340, p 280.

HOT CELL OPERATIONS

A. E. Richt E. S. Schwartz S. E. Dismuke

The Solid State Division hot cell effort has been of a service nature, and the results are reported in this document and elsewhere.

Specific reports representing the work of the metallographic section of the Hot Cell Group have been issued.[1-9]

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[1] A. E. Richt and R. M. Wallace, *Metallographic Examination of Static Corrosion Capsules Nos. 260, 291, 295, 240, 262, 290, 258, 259, 299, and 337*, ORNL CF-56-10-96 (Oct. 23, 1956).

[2] A. E. Richt and R. M. Wallace, *Metallographic Examination of Static Corrosion Capsules Nos. 248, 264, 320, 341, 247, and 276*, ORNL CF-57-2-30 (Feb. 5, 1957).

[3] A. E. Richt, *Metallographic Examination of Static Corrosion Capsules Nos. 225, 227, 232, 237, 245, 252, 253, 274, 275, 301, 302, 303, 316, 317, 333, and 345*, ORNL CF-57-3-51 (March 12, 1957).

[4] A. E. Richt and E. J. Manthos, *Metallographic Examination of Irradiated Cermet Shielding Materials — Report No. 1*, ORNL CF-56-10-59 (Oct. 12, 1956).

[5] A. E. Richt, *Metallographic Examination of Irradiated Cermet Shielding Materials — Report No. 2*, ORNL CF-56-12-105 (Dec. 18, 1956).

[6] A. E. Richt, *Metallographic Examination of Irradiated Cermet Shielding Materials — Report No. 3*, ORNL CF-57-2-33 (Feb. 7, 1957).

[7] A. E. Richt, *Metallographic Examination of Irradiated Cermet Shielding Materials — Report No. 4*, ORNL CF-57-5-46 (May 10, 1957).

[8] E. J. Manthos *et al.*, *Disassembly and Examination of MTR In-Pile Loop 3*, ORNL CF-57-3-14 (March 4, 1957).

[9] E. J. Manthos *et al.*, *Disassembly and Examination of the BeO Moderator Capsule Test ORNL-10-4*, ORNL CF-57-3-53 (March 13, 1957).

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