NAA-SR-7098
COPY

CAPSULE IRRADIATION OF UNALLOYED URANIUM AT HIGH TEMPERATURES

AEC Research and Development Report

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.

LEGAL NOTICE

Price $0.75
Available from the Office of Technical Services
Department of Commerce
Washington 25, D. C.

NAA-SR-7098
METALS, CERAMICS,
AND MATERIALS
35 PAGES

CAPSULE IRRADIATION OF UNALLOYED URANIUM AT HIGH TEMPERATURES

By

D. G. HARRINGTON
J. B. NEWTON

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.
P.O. BOX 309 CANOGA PARK, CALIFORNIA

CONTRACT: AT(11-1)-GEN-8
ISSUED: APRIL 15, 1962

DISTRIBUTION

This report has been distributed according to the category "Metals, Ceramics, and Materials" as given in "Standard Distribution Lists for Unclassified Scientific and Technical Reports" TID-4500 (16th Ed.), December 15, 1960. A total of 620 copies was printed.

ACKNOWLEDGMENT

The authors wish to thank J. H. Walter and G. G. Bentle for their invaluable assistance in the completion of this experiment.

CONTENTS

TABLES

FIGURES

ABSTRACT

Cast and wrought specimens and restrained wrought specimens of unalloyed uranium were irradiated in the Materials Testing Reactor, as the first in a series of experiments to develop fuel materials for sodium cooled reactors. Average central temperatures ranged from 710 to 1280 °F, and burnup ranged from 1230 to 2260 Mwd/MTU. The results show severe limitations of these fuel materials for Advanced Sodium Cooled Reactors, and agree substantially with data obtained from larger specimens irradiated in the SRE. The need for careful extrapolations of irradiation results from small accelerated test specimens is clearly indicated, especially in regard to diameter and length changes.

I. INTRODUCTION

The irradiation test described in this report was the first in a series to develop fuel materials for sodium cooled reactors. The objective was primarily to determine gross dimensional changes.

The irradiation behavior of nuclear fuels greatly affects the cost of nuclear power. One method of determining the irradiation stability of a fuel material is the irradiation of samples in a test reactor, such as the experiment described in this report. The specimens must be contained in a specially designed capsule which will create the environment desired. If the proper conditions are achieved and are measured, this type of testing is of great value, and has three important features not available in prototype reactor tests. The first one, obviously, is that some fuel evaluation can be accomplished for proposed concepts where no prototype reactor is in operation. The second factor is the safer study of extreme temperature and burnup conditions. The final feature is the accelerated burnup rates possible in high neutron fluxes, thereby securing much earlier results. However, the accelerated tests always require compromise in temperature or specimen size.*

In the NAA-38-1 test irradiated in the Materials Testing Reactor (MTR), the conditions in the capsule were planned to simulate those in the Sodium Reactor Experiment (SRE). The burnup rate was four times that in the SRE; and the specimen size was compromised, rather than the temperature gradient. Unfortunately, the results were not obtained in advance of actual SRE experience on the same materials. Nevertheless, the data are important for the evaluation of uranium alloys irradiated in similar MTR capsules. In addition, they provide information at higher temperatures and burnups than those achieved in the SRE.

The capsule contained five cast, five wrought, and two restrained wrought fuel pins. The restraint was provided by Type 304 stainless steel cladding which was scaled down from SRE dimensions, 0.790 in. OD by 0.010 in. wall, to 0.395 in. OD by 0.005 in. wall. The fuel specimens were 3/8 in. OD by 1.5 in. long.

---

* Loops in test reactors have the same advantages as those listed above for capsules and have dynamic, rather than static, coolant conditions with opportunity for multiple rods or plates; but the cost is a great deal more.

II. EXPERIMENTAL METHODS

A. CAPSULE DESIGN[1]

The parameters desired were fuel temperatures of 900°F surface and 1150°F central. A calculation was made to determine the heat generation required to produce the given temperature gradient. Additional calculations showed that, using the required heat generation rate, the 900°F fuel surface temperature could be achieved with a thermal barrier consisting of a NaK annulus and a stainless steel shell.

A schematic of the capsule is shown in Figure 1. A photograph of the capsule components, prior to final assembly, is shown in Figure 2. A perforated metal basket was chosen to support the 18 in. fuel column (Figure 3). There were several requirements which this basket had to satisfy. First, the fuel needed to be centered in the NaK, but allowed to swell unrestrained; second, accurate location of the thermocouples, as close to the fuel surface as possible, was desired; and third, the basket was to be easy to handle in the hot cell, for removal of the fuel. The basket was made from 0.010 in. thick Type 304 stainless steel, and perforated with 1/4 in. holes on 3/16 in. centers. There was a full length longitudinal gap in the basket to allow for diametrical expansion with no restraint. An annulus between the basket and the fuel was provided for placement of seven fuel surface thermocouples. Four longitudinal tubes, the same diameter as the thermocouples, kept the fuel centered.

B. FUEL FABRICATION HISTORY

Seven specimens of 10% enriched unalloyed uranium were from a 3/8 in. diameter, alpha-rolled and beta-quenched rod, made at Argonne National Laboratory for this experiment. The "as rolled" surfaces were maintained for the irradiations. The total impurity content was 100 to 200 ppm Al + B + Be + Co + Mg + Mn + Mo + Ni + Sn. This is high purity material.

Five specimens of 10% enriched unalloyed uranium were cast at AI. The "as cast" surfaces were maintained for the irradiations. The specimens were beta quenched. The total impurity content was the same as the alpha-rolled samples.

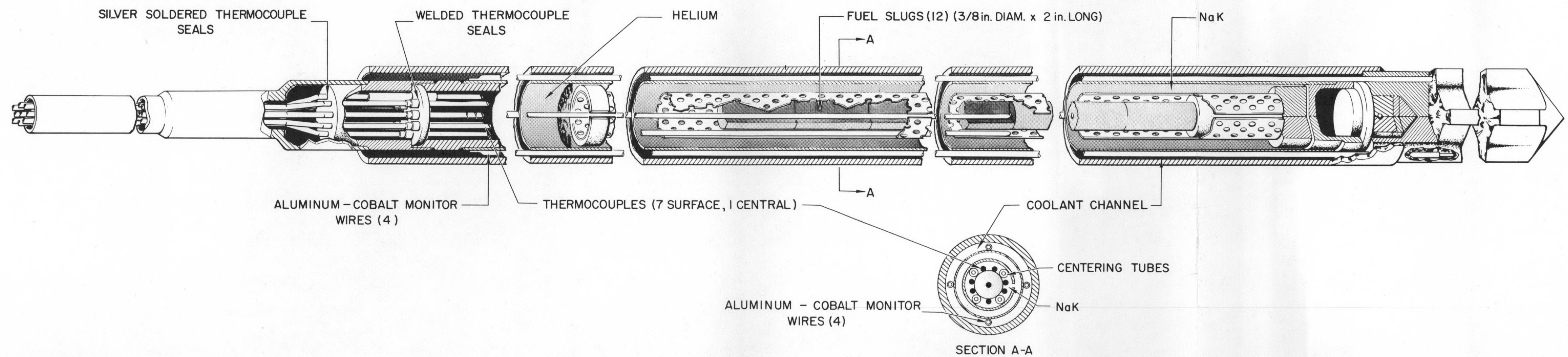


Figure 1. Irradiation Capsule Design

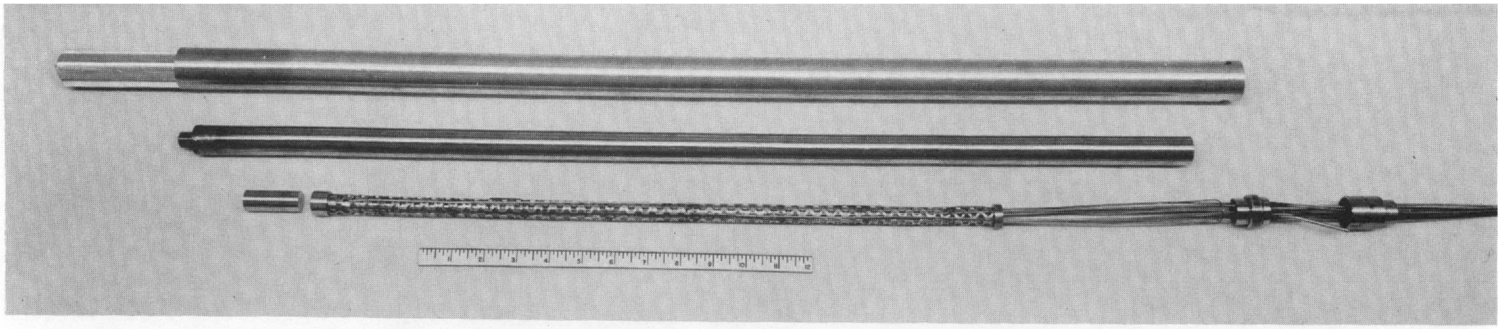


Figure 2. Irradiation Capsule Ready for Final Assembly

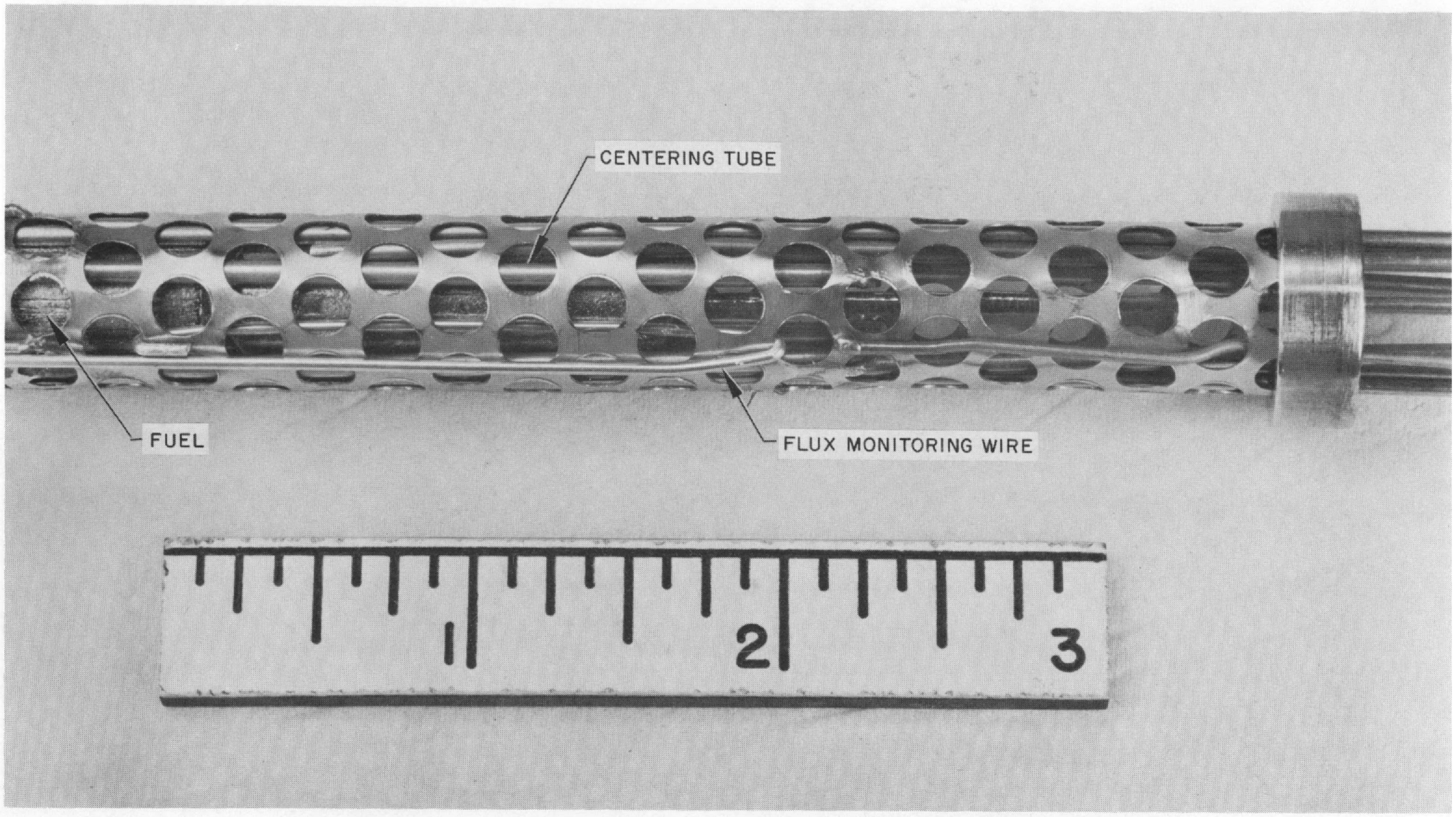
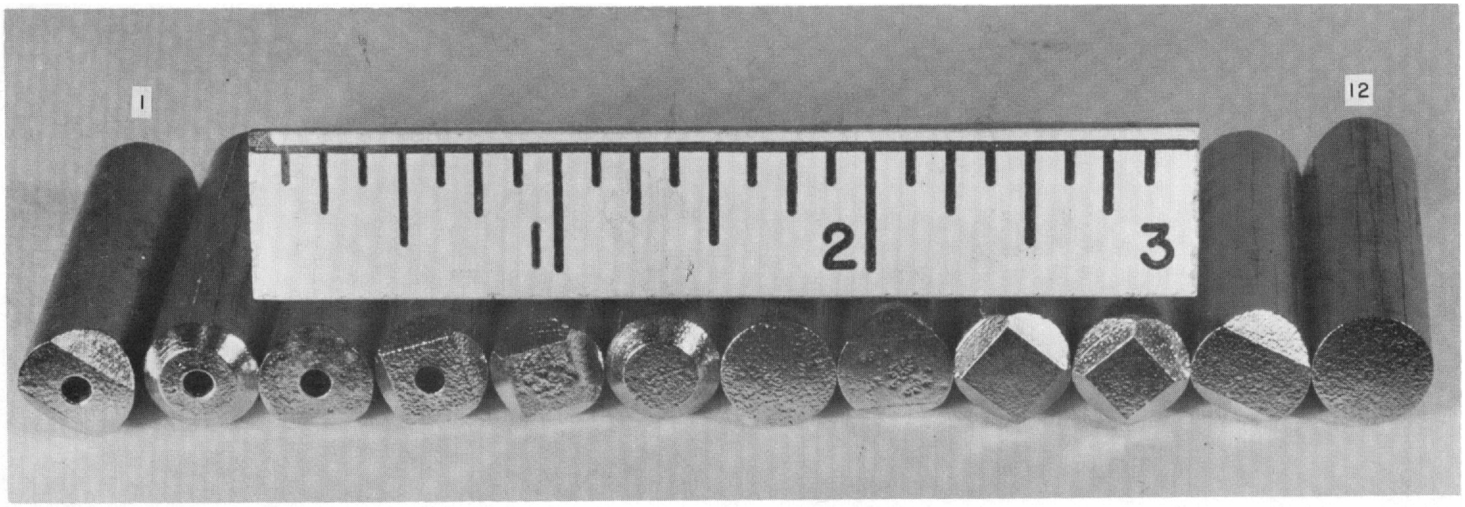
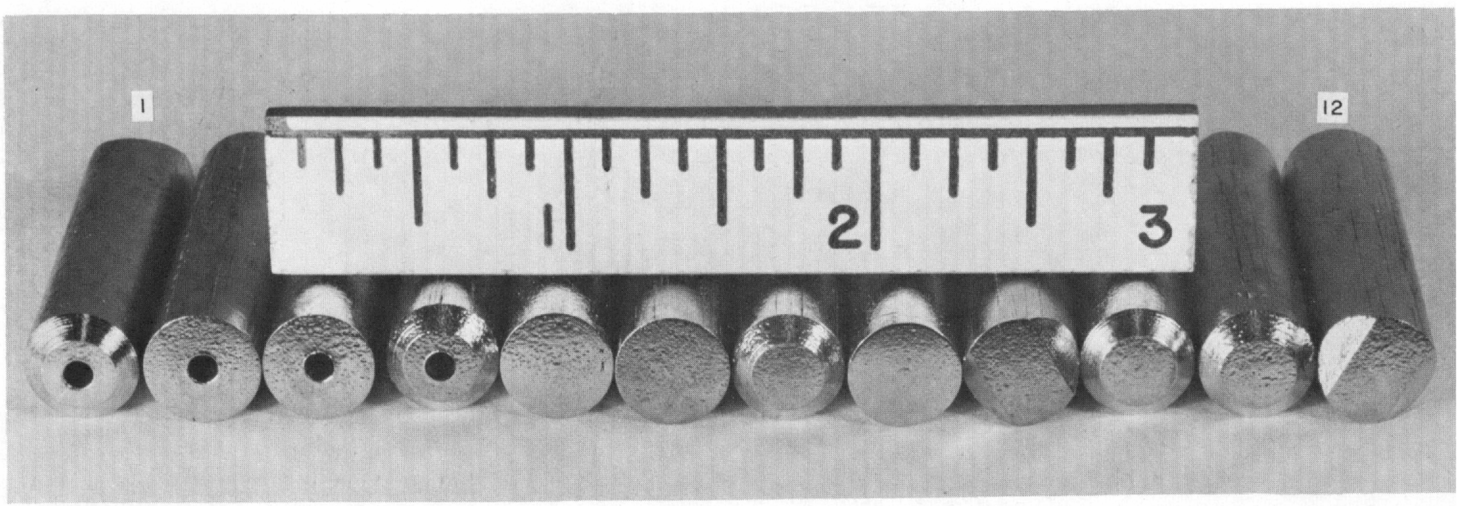


Figure 3. Closeup View of Specimen Basket Assembly



a. Top



b. Bottom

Figure 4. Specimen Identification

The identification of specimens was accomplished with machined bevels at the sample ends, as shown in Figure 4. The top end could be distinguished from the bottom end. Side view pre-irradiation photographs are included with the post-irradiation photographs in Figure 7, Section III-C.

C. REACTOR HISTORY

The experiment was inserted into the MTR, in Position A-40-NE, for Cycle 117 on February 22, 1959. The operating temperatures were lower than the desired values, so the assembly was transferred to Position A-40-SE for Cycles 118 and 119, and subsequently discharged on April 6, 1959. This resulted in 15.65 days of full power operation in an advertised flux of 6.5×10^{13} n/cm^2-sec, followed by 28.53 days of full power operation in a flux of 8.5×10^{13} n/cm^2-sec.

D. HOT CELL EXAMINATION

The hot cell examination consisted of opening the capsule, reacting the NaK thermal bond, detailed visual inspection, photographs, physical measurements, and scanning of the fuel and monitor wires for gamma activity to obtain relative burnup information.

The configuration of the NAA-38 series of capsules made the use of a pipe cutting machine a feasible method of opening. A Vermette power drive of compact size, and adjustable roll type pipe cutters on the power drive, provided the necessary cutting capacity.

All opening of the capsule assembly was accomplished in an atmosphere of nitrogen which contained less than 3.5% oxygen. By utilizing this inert atmosphere, the 120 cc of NaK contained in the capsule assembly needed no special handling during the 3-1/2 hr period required to react it completely with isoamyl alcohol.

The measurements taken on the NAA-38-1 fuel pins were: (1) diameters, (2) lengths, (3) density, and (4) gamma activity of the fuel and of the cobalt-aluminum monitoring wires. The diameter and length measurements were taken with dial indicators which were graduated in 0.001 in. intervals. The indicators were calibrated on a 0.375 in. diameter stainless steel dowel pin, and were checked at the completion of each set of measurements on individual slugs.

The densities of the fuel pins were determined by the dry:wet weight method, using butyl alcohol as the liquid. By correcting for the temperature of the alcohol bath, the results of three measurements on each fuel pin were in agreement to within 3 parts in 2000.

The gamma scanning of the fuel pins was accomplished on a ball-screw driven stage that, through the use of calibrated digital readouts, could be positioned to within 0.010 in. The stage first rotated the fuel pins at 0.3 rpm, so that the side having the highest burnup was found. The longitudinal scan was then made along that side.

The scanning of the four monitoring wires was accomplished by running a scan of the No. 1 wire and then, after scanning the remaining three, making a rerun of No. 1 to check for any instrument variations. The results of the two runs on wire No. 1 were in agreement to within 4%. Two pieces, each approximately 1 cm long, were then cut from known positions along wire No. 1. These two pieces were dissolved and 1/1000 aliquots were taken. After calibrating the activity of these aliquots against a standard NBS Co^{60} source, the total integrated flux was calculated.

E. DETERMINATION OF OPERATING TEMPERATURES AND BURNUPS

One tantalum foil wrapped thermocouple was located at the fuel axis, 5-1/2 in. from the top of the fuel. Seven thermocouples were located in the NaK, adjacent to the fuel surface, at five elevations over the 18-in. fuel column. Three surface thermocouples were at the same elevation as the central thermocouple, and were located at 120° intervals. All thermocouple readings were measured and recorded by a potentiometric recorder.

Temperature profiles were drawn directly from the thermocouple measurements for the first few days of operation. However, two adjacent thermocouples at the middle of the assembly, and one at the lower end, failed early in the irradiation. These failures prevented further accurate estimates of daily temperature profiles. Consequently, overall temperature profiles were derived by superimposing normalized flux profiles, derived from monitor wire data, upon the temperatures achieved from two elevations where good temperature measurements were obtained. The correlation between the flux profiles and temperature data permitted reasonable estimates to be made of the average and maximum temperatures of all specimens.

Three means for measuring burnup were utilized: (1) advertised flux, (2) temperature data, and (3) gamma scanning of fuel and monitor wires. The burnup from advertised flux, and flux as determined by the monitor wires, was computed from the expression

$$\frac{N}{N_o} = \frac{a}{\sigma_f} \left[1 - \exp\left(\sigma_a \Phi_f t\right) \right]$$

where:

N = atoms U^{235} at time t

N_o = atoms U^{235} initially present

σ_a = absorption cross section for U^{235}

σ_f = fission cross section for U^{235}

Φ_f = average flux in fuel

The relationship between the unperturbed advertised flux, the monitor wire flux in the coolant channel, and the average flux in the fuel were determined by use of the I_2 IBM machine code which is a P_3 approximation to the solution of the Transport Equation by the Weil method.

The burnups calculated from temperature data were derived from considerations of the thermal resistances of the capsule annuli, assuming 175 Mev per fission energy dissipation by conduction in the capsule, and 200 Mev per fission total energy produced.

Gamma scans of the fuel gave activity profiles identical to the monitor wire results, and therefore were not used further in the burnup calculations.

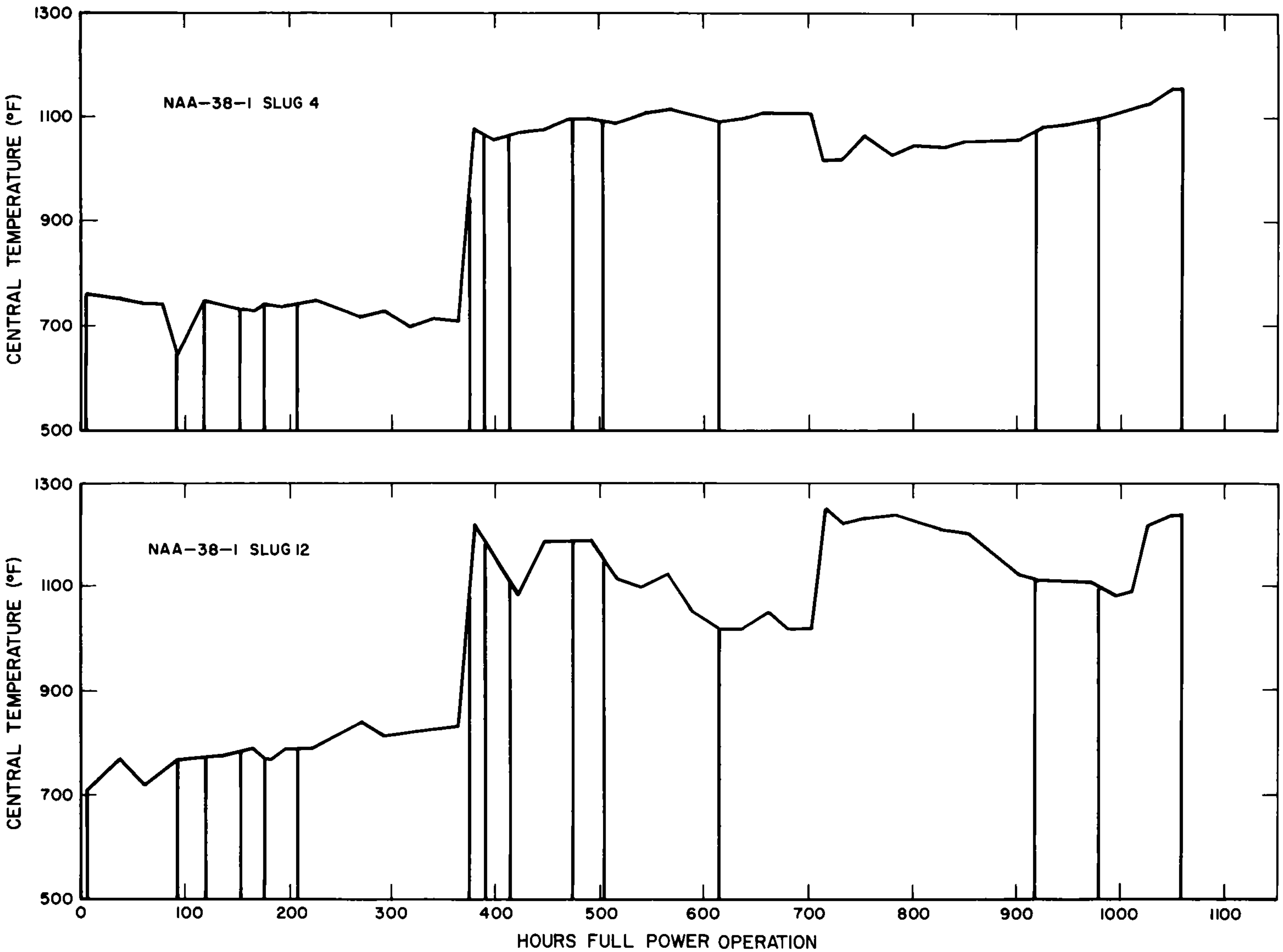


Figure 5. Central Temperature Histories for Specimens 4 and 12

III. RESULTS

A. TEMPERATURE

Central temperature histories of Specimens 4 and 12 are shown in Figure 5. These data were computed directly from thermocouple readings. Table I gives estimated temperatures for each pin, based on calibration points at Specimen 4 or 12 correlated with the monitor wire and fuel gamma scan profile data.

TABLE I

ESTIMATED SPECIMEN TEMPERATURES

Specimen Number	Average Surface Temperature After Transfer To Higher Flux (°F)	Average Central Temperature After Transfer To Higher Flux (°F)	Maximum Central Temperature (°F)
1	560	710	820
2	660	840	940
3	740	950	1050
4	830*	1070*	1160*
5	900	1150	1240
6	940	1220	1310
7	990	1270	1370
8	1000	1280	1380
9	970	1240	1360
10	920	1180	1340
11	870	1120	1300
12	830*	1070*	1250*

* Measured calibration points, all others based on monitor wire data relative to calibration points.

B. CALCULATED BURNUP

The results of burnup calculations are given in Figure 6. The peak burnup derived from monitor wire data was 3560 Mwd/MTU; whereas, from temperature data, the peak was only 2560 Mwd/MTU. This discrepancy is possibly related to flux hardening, that is, a shift of the neutron energy spectrum toward higher energy in the vicinity of the fuel region. In a previous similar test series, there was good agreement between burnups obtained from radiochemical analysis and temperature monitoring techniques.[2] For this reason, the temperature monitoring burnup values will be used henceforth in this report.

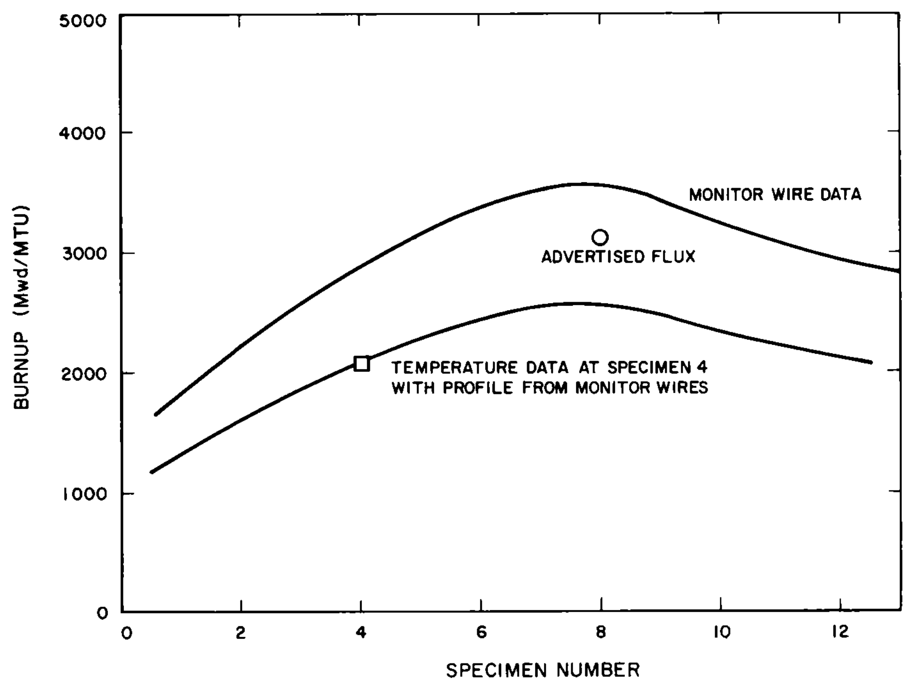


Figure 6. Burnup Profiles

C. PHOTOGRAPHS OF SPECIMENS

Figure 7 shows each specimen, before and after irradiation. The "as cast" and "as rolled" surfaces were rough before irradiation, and much more so after the irradiation. The fuel in the restrained samples tended to extrude, rather than expand the cladding diametrally.

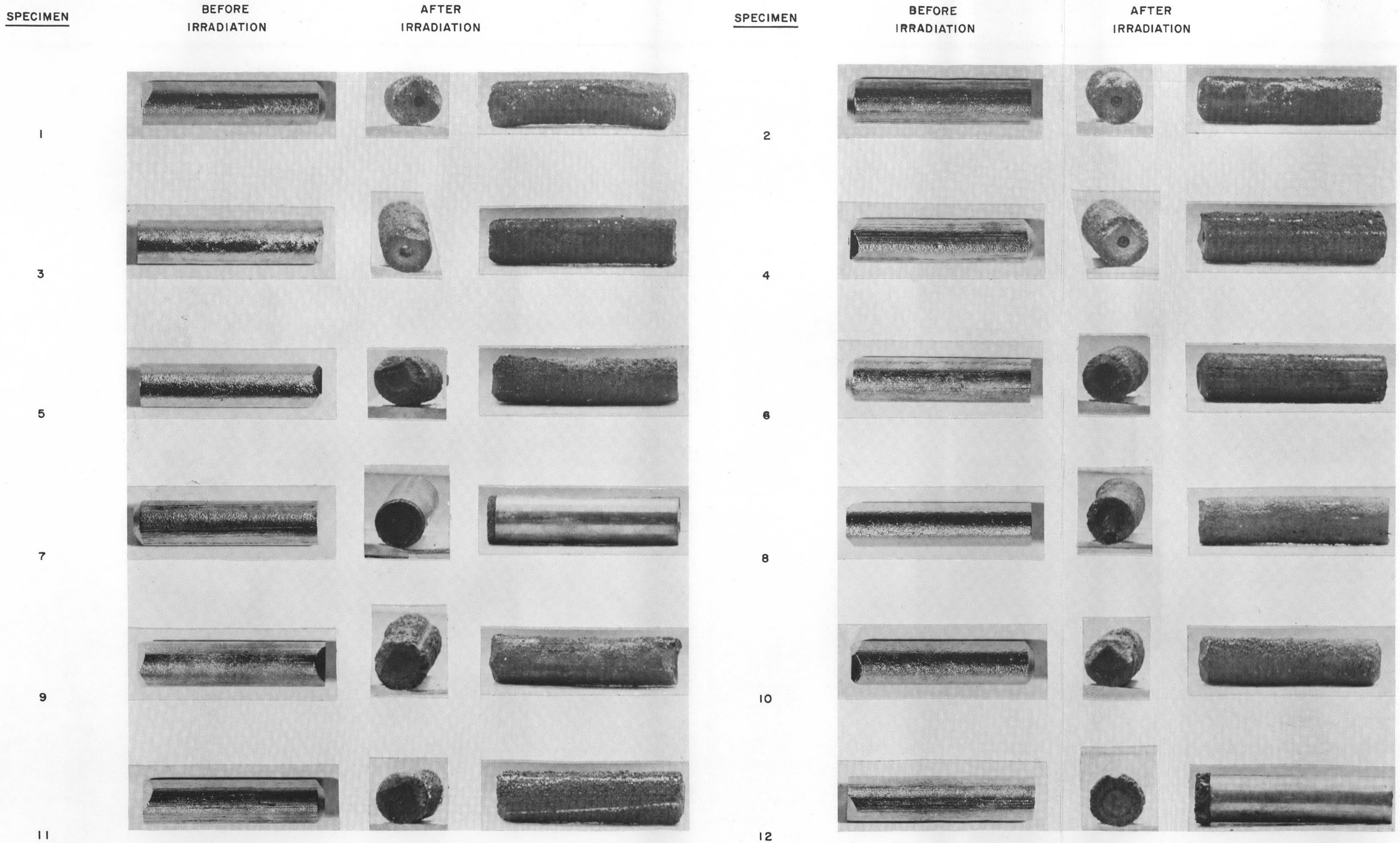


Figure 7. Photographs of Specimens Before and After Irradiation

D. DIAMETER, LENGTH, AND VOLUME INCREASES

The diameter, length, and volume changes which occurred as a result of the irradiation are presented in Table II.

TABLE II

DIAMETER, LENGTH, AND VOLUME CHANGES

Specimen	Description*	Increase (%)		
		Diameter†	Length	Volume§
1	Cast	4.0	5.0	0.9
2	Alpha rolled	5.0	4.9	5.0
3	Cast	5.9	2.7	6.0
4	Alpha rolled	8.5	4.9	14.0
5	Cast	6.7	2.3	8.8
6	Alpha rolled	6.4	4.1	12.2
7	Alpha rolled and jacketed	3.8	5.1	10.1
8	Cast	7.6	4.8	13.1
9	Alpha rolled	9.3	5.1	13.5
10	Cast	7.8	2.2	11.3
11	Alpha rolled	9.3	3.9	15.1
12	Alpha rolled and jacketed	3.7	7.2	11.1

* All specimens beta quenched

† Based on average of 20 postirradiation dial gauge readings

§ From density measurements

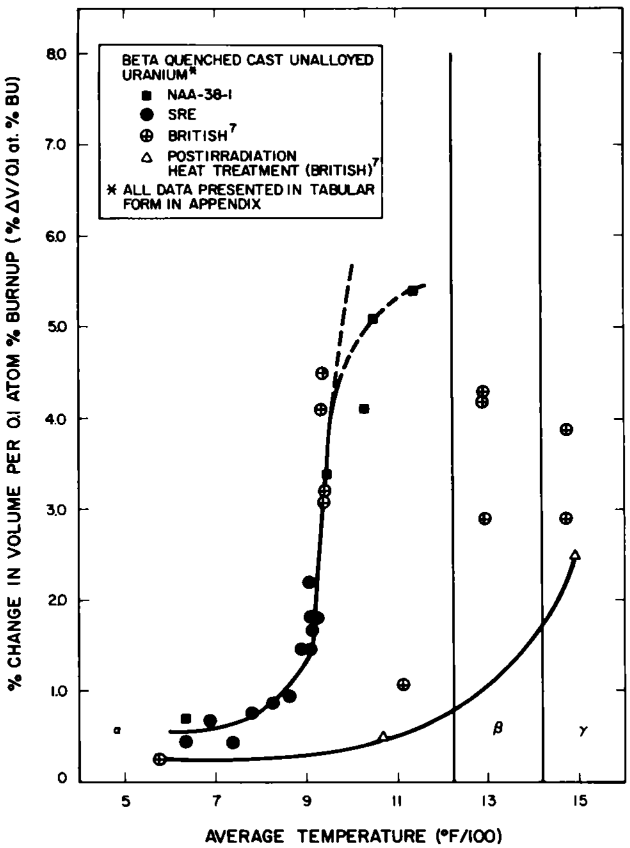


Figure 8. %ΔV/0.1 at.% BU vs Average Temperature for Cast and for Postirradiation Heat Treated Unalloyed Uranium

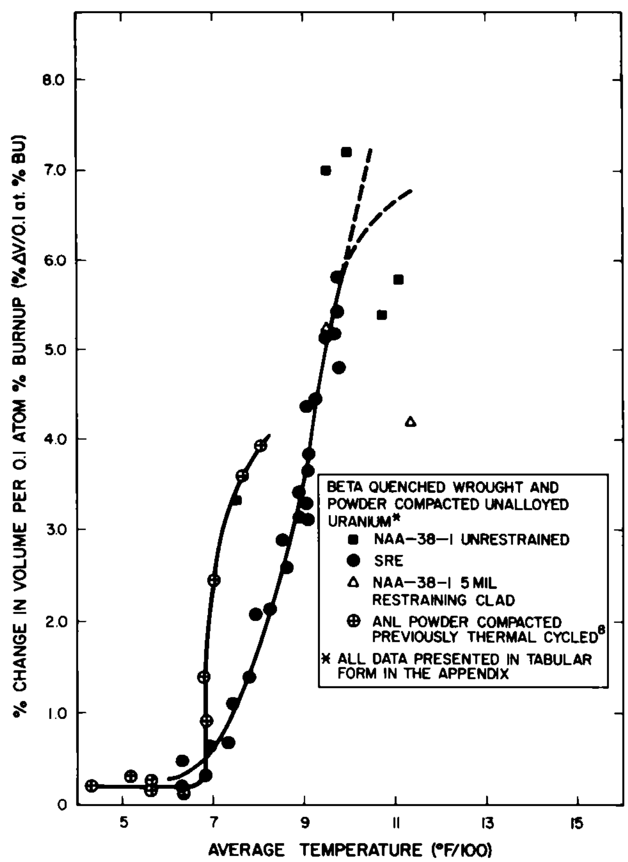


Figure 9. %ΔV/0.1 at.% BU vs Average Temperature for Wrought and for Powder Compacted Unalloyed Uranium

IV. DISCUSSION AND RESULTS

As an aid in evaluating the results of this test, comparisons with irradiation data from tests irradiated in the SRE have been made.[4,5,6] Also, some British[7] and ANL[8] irradiation data on unalloyed uranium were included. In plotting the data, two normalizing factors were consistently used. One was that of normalizing all percent dimensional changes to one-tenth total atom percent burnup, which gave ordinates such as $\% \Delta V/0.1$ at. % BU. Since swelling is a strong function of burnup, extrapolations of these data beyond actual burnups achieved is not recommended. The other factor was that of using the average temperature between the fuel surface and center for abscissas, as recommended by G. G. Bentle.[3] This temperature is believed to be the most representative to use when plotting data from a variety of samples and conditions. All of the data placed in the graphs are tabulated in the Appendix, in Tables IV to VII.

A. VOLUME INCREASE

There was excellent agreement between the MTR and the SRE results for cast unalloyed uranium, as can be seen in Figure 8. Although a curve drawn through the MTR data points only would indicate more swelling in the 700 to 900°F range than the SRE data indicate, it is presumed that this is due only to the absence of MTR points in that region. Four British data points at 932°F agree well with the Atomics International SRE and MTR results. Additional British[7] information shows less swelling in the beta and gamma phases than in the high alpha-phase temperatures used in the NAA-38-1 test.

As for wrought unalloyed uranium, the MTR data are more scattered than is the case for cast material; but, in spite of this, they tended to agree with SRE experience, as shown in Figure 9. Incidentally, the NAA-38-1 point at 750°F, while at a temperature 150°F below the SRE data, ran at an estimated 850°F maximum average temperature near the end of the irradiation. The lower temperature NAA-38-1 pins were located near the ends of the assembly; and, as a consequence, they experienced larger temperature variations, due to flux shifts in the MTR. ANL powder compacted unalloyed uranium are included in Figure 9 for comparison.

TABLE III

EFFECT OF RESTRAINT ON VOLUME, DIAMETER, AND LENGTH CHANGES FOR WROUGHT UNALLOYED URANIUM

Specimen Number	Description	Average Irradiation Temperature (°F)	Total Atom % Burnup	$\frac{\% \Delta V}{0.1 \text{ at.\% BU}}$ *	Volume Change in Restrained Specimen (% Decrease)	$\frac{\% \Delta D}{0.1 \text{ at.\% BU}}$	Diameter Change in Restrained Specimen (% Decrease)	$\frac{\% \Delta L}{0.1 \text{ at.\% BU}}$	Length Change in Restrained Specimen (% Increase)
SRE R-57-8	Not Restrained	950	0.085	5.1		1.1		4.0	
MTR 4	Not Restrained	950	0.20	7.0	21	4.2	57	2.4	50
MTR 12	Restrained	950	0.20	5.5		1.8†		3.6	
MTR 9	Not Restrained	1105	0.23	5.8	27	4.0	60	2.2	5
MTR 7	Restrained	1130	0.24	4.2		1.6§		2.1	

* From density measurements, cladding removed

† Cladding removed. The maximum increase in diameter of the jacket was 0.005 in., 1.3%.

§ Cladding removed. The maximum increase in diameter of the jacket was 0.002 in., 0.5%.

The volume changes in the restrained specimens were 21 and 27% less than unrestrained specimens at the same temperatures (Figure 9 and Table III). At 950°F, an unrestrained specimen from SRE grew in volume 7% less than the restrained MTR specimen and 27% less than the unrestrained MTR specimen. Five possible explanations for the higher swelling rates indicated for the MTR specimens over SRE specimens, at an average temperature of 950°F are: (1) higher MTR maximum temperatures, (2) higher burnups, (3) higher burnup rates, (4) smaller MTR specimen size, resulting in less self-restraint, and (5) magnitude of thermal cycles. Insufficient theory or data are available to prove which effect or effects are predominant.

For a more comprehensive comparison of volume changes in cast, wrought, powder compacted, and postirradiation heat-treated unalloyed uranium, the curves without data points from Figures 8 and 9 have been combined in Figure 10. The ANL powder compacted uranium reached the knee of the swelling curve at about the same temperature as the wrought uranium, but increased much more rapidly thereafter. In contrast, the cast uranium is somewhat more stable than the wrought in the 700 to 900°F range. The dashed lines showing decreased positive slope at the top of the irradiation swelling curves in Figures 8, 9, and 10 seem reasonable from two points of view:

1) In the bend-back region (decreasing positive slope above 950°F), approximately half of each specimen had temperatures in the 1000 to 1220°F region; and much greater swelling would have been observed if the curves did not bend back.

2) The average central temperature of several of the MTR pins was in the beta phase; and, assuming less swelling in the beta phase, these swelling curves should bend back, even if the actual swelling rate were discontinuous across the phase transition.

The postirradiation heat treatments did not show equivalent swelling vs temperature (except in the gamma phase, as shown in Figure 8).

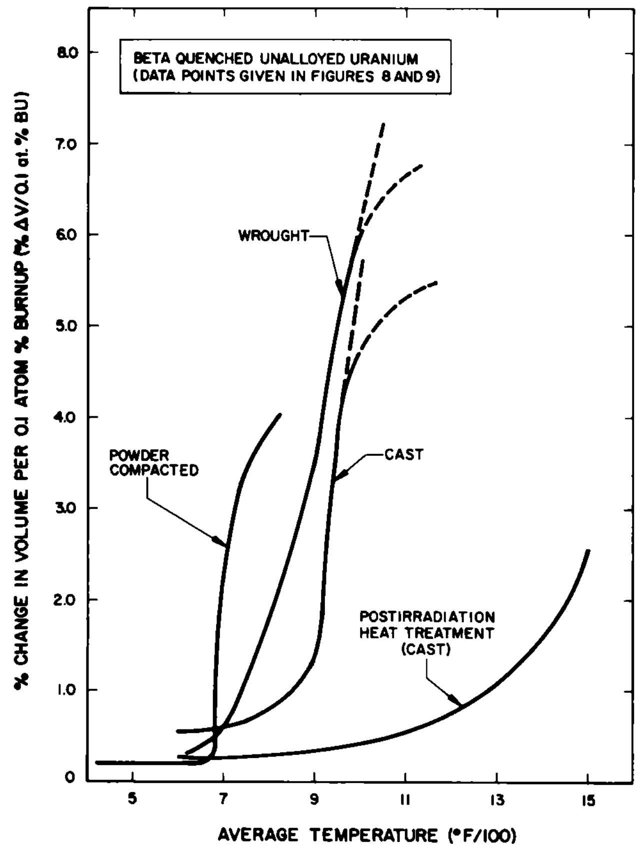


Figure 10. Comparison of %ΔV/0.1 at. % BU vs Average Temperature for Wrought, Cast, Powder Compacted, and Postirradiation Heat Treated Unalloyed Uranium (Curves only)

B. DIAMETER CHANGE

Figure 11 shows the percent diameter change per 0.1 atom percent burnup vs average temperature for SRE and NAA-38-1 tests. It can be seen that the diameter increases for wrought uranium were, in general, greater than for cast uranium, in the case of both the MTR and SRE data. However, the percent changes in the MTR results were much greater than for the SRE. This can be attributed to the larger MTR volume increases discussed previously, where applicable, and to a large degree to the surface roughening and the smaller MTR specimen size. That is, the same surface roughness on 3/8-in. diameter MTR specimens as on 3/4-in. diameter SRE specimens would give twice the percent change. Of course, this change would be added to the percent diameter change from internal swelling. It should be noted that this surface condition does not affect the volume swelling data in the same manner, since these data are

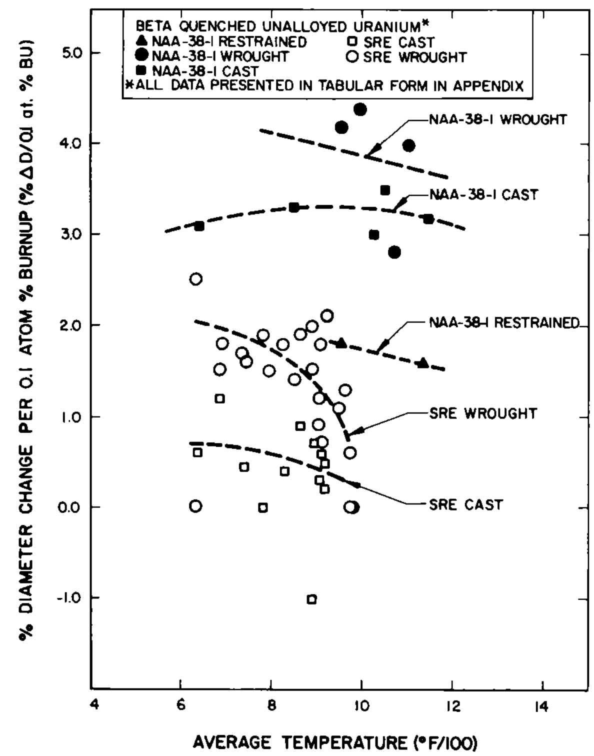


Figure 11. %ΔD/0.1 at. % BU vs Average Temperature for SRE and NAA-38-1 Unalloyed Uranium Irradiations

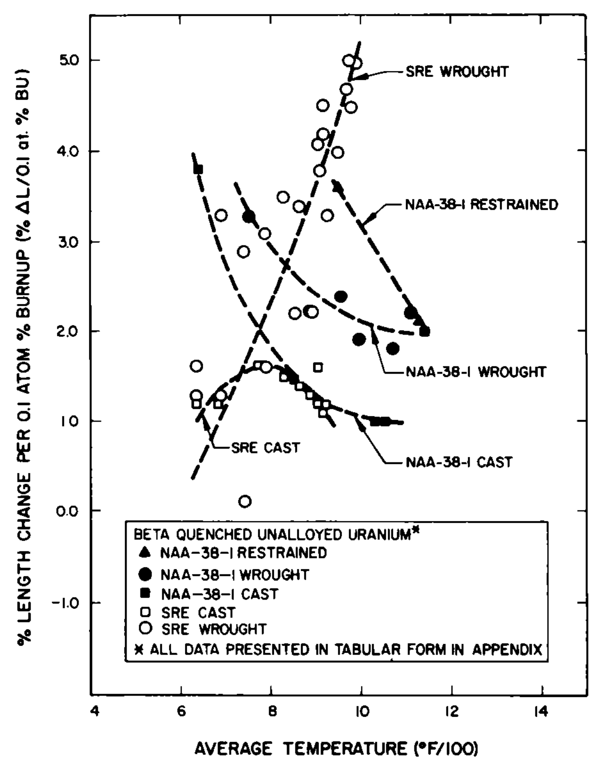


Figure 12. %ΔL/0.1 at. % BU vs Average Temperature for SRE and NAA-38-1 Unalloyed Uranium Irradiations

based on immersion density measurements. It should be noted also that the negative slope to the SRE %ΔD/0.1 at. % BU data is at least partially due to the fact that surface roughening was a minimum at the maximum surface temperature.

The MTR restrained specimens had an initial 5-mil NaK bond, compared to a 10-mil bond for the SRE slugs, and restraint occurred only after a swelling of greater than 5 mils. In the case of slugs from SRE Rod R-57 (See Table V, Appendix I), restraint from diameter increases had just begun on the worst slugs. The maximum increase in SRE cladding diameter measured was 0.4% at a burnup of about 1000 Mwd/MTU. Only small changes in the diameter of the restraining jackets of the MTR pins occurred (1.3 and 0.5% maximum for Specimens 7 and 12, respectively). In view of the much higher burnup in the MTR test, the change in cladding diameter is less than half the amount one would predict from the SRE results. One partial explanation might be the fact that the MTR specimens at the same average temperatures did have greater temperature gradients, and hence higher central temperatures, permitting more axial extrusion. Moreover, the maximum MTR temperatures were about 200°F higher than SRE temperatures.

Considering the fuel only, and forgetting jacket expansion, Figure 11 and Table III show that restraint had a significant beneficial effect on MTR pin diameter increases. But the SRE unrestrained specimen, R-57-8, was as good as the restrained MTR specimen at the same temperature, for possible reasons listed earlier under the section on volume change.

C. LENGTH INCREASE

A comparison of percent length changes per 0.1 atom percent burnup for SRE and MTR specimens is given in Figure 12. There is fair agreement for cast uranium, but opposite trends for alpha-rolled uranium.

In reference to Table III, at 950°F the restrained specimen, No. 12, grew 50% more in length than the unrestrained specimen, No. 4. At the same time, at 1130°F, restrained pin No. 7 grew in length only 5% more than unrestrained pin No. 9. There are no logical explanations of the differences evident.

V. SUMMARY

An instrumented test capsule containing cast, wrought, and restrained wrought heat treated unalloyed uranium was designed, constructed, irradiated in the MTR, examined, and evaluated. Average central fuel temperatures were from 710 to 1280° F, and burnups ranged from 0.13 to 0.24 total atom percent. The hot cell examination included photographs and measurement of density, diameter, and length.

The maximum volume increase was 15.1% for the alpha-rolled uranium, and occurred at a central temperature of 1120°F average and 1300°F maximum and at a burnup of 0.21 at. %. The maximum volume increase was 13.1% for the cast uranium, and occurred at a central temperature of 1280°F average and approximately 1380°F maximum.

Two specimens, restrained with 5 mils of stainless steel, grew in volume 21 and 27% less than unrestrained specimens at the same temperatures.

Diameter and length changes differed from SRE results, with only partial explanations evident.

Some SRE, ANL, and British data on unalloyed uranium were included in the analysis for comparative purposes.

VI. CONCLUSIONS

As a result of the work done on this test the following conclusions can be made regarding unalloyed uranium:

1) High purity unalloyed uranium is not a suitable fuel material for sodium cooled reactors, because of high volume swelling rates in the temperature range of interest.

2) The volume increases due to irradiation, determined in the accelerated MTR test with 3/8-in. diameter specimens, compare favorably with SRE results on 3/4-in. diameter slugs.

3) Diameter percentage increases in the MTR were considerably greater than extrapolations from SRE results. This is primarily attributed to surface roughness, which leads to higher percentage changes in smaller specimens. Nevertheless, the correct order was maintained between cast and wrought material; that is, diameter increases for wrought uranium were greater than for cast uranium, in both reactor tests considered independently.

4) There was disagreement between the MTR and the SRE results on percentage length changes, and no obvious explanation is evident, other than scatter in experimental results.

5) The variety of good and poor correlations found between the irradiation results of large SRE specimens and small MTR samples show the need for careful extrapolations of irradiation data.

6) In order to accurately separate the effects of burnup rate, specimen size, temperature, temperature gradient, thermal cycling grain size, purity, heat treatment, etc., many more specimens, irradiated under better controlled and measured conditions, would be required.

APPENDIX I

TABULATED MTR, SRE, BRITISH, AND ANL IRRADIATION DATA ON UNALLOYED URANIUM

TABLE IV

TABULATED NAA-38-1 UNALLOYED URANIUM IRRADIATION DATA

Specimen Number	Description*	Average Irrad. Temp.† (°F)	Average Central Temp. (°F)	at.% BU	% Diameter Increase	% Length Increase	% Volume Increase (from density)	% Volume Increase (calculated from dimensions)	$\frac{\% \Delta V}{0.1 \text{ at.\% BU}}$	$\frac{\% \Delta D}{0.1 \text{ at.\% BU}}$	$\frac{\% \Delta L}{0.1 \text{ at.\% BU}}$
1	a	635	710	0.13	4.0	5.0	0.9	13.0	0.72	3.1	3.8
2	b	750	840	0.15	5.0	4.9	5.0	14.9	3.3	3.3	3.3
3	a	845	950	0.18	5.9	2.7	6.0	14.5	3.4	3.3	1.5
4	b	950	1070	0.20	8.5	4.9	14.0	21.9	7.0	4.2	2.4
5	a	1025	1150	0.22	6.7	2.3	8.8	15.7	4.1	3.0	1.0
6	b	1070	1220	0.23	6.4	4.1	12.2	16.9	5.4	2.8	1.8
7	c	1130	1270	0.24	3.8	5.1	10.1	12.7	4.2	1.6	2.1
8	a	1140	1280	0.24	7.6	4.8	13.1	20.0	5.4	3.2	2.0
9	b	1105	1240	0.23	9.3	5.1	13.5	23.7	5.8	4.0	2.2
10	a	1050	1180	0.22	7.8	2.2	11.3	17.8	5.1	3.5	1.0
11	b	995	1120	0.21	9.3	3.9	15.1	22.5	7.2	4.4	1.9
12	c	950	1070	0.20	3.7	7.2	11.1	14.6	5.5	1.8	3.6

* a. Cast and beta quenched
b. Alpha rolled and beta quenched
c. Same as b, but with 5-mil stainless steel restraining jacket

† Sum of average fuel surface and average fuel central temperatures (during period in higher flux position) divided by two.

TABLE V

TABULATED SRE UNALLOYED URANIUM IRRADIATION DATA[4,5,6]

Specimen Position	Description*	Average Irrad. Temp.† (°F)	at.% BU	% ΔV (from Δρ)	% ΔV / 0.1 at.% BU	Central Fuel Temp. (°F)	% ΔD / 0.1 at.% BU	% ΔL / 0.1 at.% BU
MC-1-1								
1		635	0.047	0.1	0.213	695	2.5	1.25
2		687	0.060	0.2	0.333	765	1.5	3.3
3		738	0.072	0.5	0.694	830	1.7	2.9
4		782	0.080	1.1	1.375	881	1.9	3.1
5		828	0.084	1.8	2.143	935	1.8	3.5
6	Alpha Rolled	862	0.085	2.2	2.588	970	1.9	3.4
7		890	0.083	2.6	3.132	1000	2.0	2.2
8		903	0.076	2.5	3.290	1005	1.2	3.8
9		910	0.066	2.4	3.636	997	1.8	4.1
10		912	0.055	2.1	3.818	985	0.7	4.2
11		912	0.042	1.3	3.095	965	0.7	4.5
12		905	0.032	1.4	4.375	945	0.9	4.1
1		635	0.047	0.2	0.426	695	0.6	1.2
2		687	0.060	0.4	0.666	765	1.2	1.2
3		738	0.072	0.3	0.417	830	0.4	N.R.
4		782	0.080	0.6	0.75	881	0	1.6
5		828	0.084	0.7	0.833	935	0.4	1.5
6	Cast	862	0.085	0.8	0.941	970	0.9	1.4
7		890	0.083	1.2	1.446	1000	-1.0	1.3
8		903	0.076	1.1	1.447	1005	0.7	1.2
9		910	0.066	1.2	1.818	997	0.6	N.R.
10		912	0.055	1.0	1.818	985	0.5	1.1
11		912	0.042	0.7	1.666	965	0.2	1.2
12		905	0.032	0.7	2.188	945	0.3	1.6
R-57								
1		635	0.053	0.26	0.490	695	0.0	1.6
2		692	0.067	0.42	0.627	770	1.8	1.3
3		748	0.081	0.90	1.111	840	1.6	0.1
4		795	0.089	1.84	2.067	902	1.5	1.6
5		851	0.094	2.69	2.862	958	1.4	2.2
6	Alpha Rolled	890	0.095	3.26	3.431	998	1.5	2.2
7		923	0.093	4.16	4.473	1033	2.1	3.3
8		950	0.085	4.37	5.141	1052	1.1	4.0
9		963	0.072	3.74	5.194	1050	1.3	4.7
10		973	0.062	3.37	5.435	1046	0.6	5.0
11		974	0.047	2.74	5.830	1027	0.0	4.5
12		977	0.036	1.74	4.833	1017	0.0	5.0

* All specimens beta quenched, 0.75 in. OD by 6.0 in. long.
† Sum of surface and central temperatures divided by two.

TABLE VI

TABULATED BRITISH CAST UNALLOYED URANIUM DATA[7]

Specimen Number	Description*	Measured Irradiation Temperature† (°F)	at.% BU	% Δ V (from Δ ρ)	$\frac{\% \Delta V}{0.1 \text{ at.\% BU}}$
36	Arc Melted	932	0.13	4.0	3.10
36		932	0.34	10.9	3.20
42		1472	0.10	2.9	2.90
42		1472	0.37	14.1	3.90
43		1292	0.10	2.9	2.90
43		1292	0.39	20.1	4.20
60		1292	0.27	11.5	4.30
72		932	0.26	8.1	3.10
73		932	0.26	11.8	4.50
74		932	0.26	10.8	4.10
218		1112	0.41	4.4	1.07

Postirradiation heat treatment of a specimen irradiated at 572°F to 0.4 total atom % burnup gave the following results:

Temperature (°F)	%ΔV	$\frac{\% \Delta V}{0.1 \text{ at.\% BU}}$
572 (as irradiated)	1	2.5
1067	2	5
1490	10	25
1832	20	50

* All specimens beta-quenched 0.1 in. by 0.1 in. buttons

† Surface temperature nearly equal to central temperature

TABLE VII

TABULATED ANL POWDER COMPACTED UNALLOYED URANIUM DATA[8]

Specimen Number	Description	Average Irradiation Temperature† (°F)	at.% BU	% Volume Increase (from $\Delta\rho$)	$\frac{\% \Delta V}{0.1 \text{ at.\% BU}}$
AE-10		518	0.43	1.29	0.30
AE-6		437	0.44	0.95	0.216
AE-18		563	0.53	0.93	0.175
AE-2		563	0.58	1.19	0.205
AE-15	Powder Metallurgy plus thermal cycling* before irradiation. 0.25-in. diameter by 0.75-in. long pin.	635	0.39	0.48	0.123
AE-7		680	0.64	8.96	1.40
AE-19		689	0.64	5.76	0.90
AE-16		706	0.64	15.36	2.40
AE-4		761	0.66	23.76	3.60
AE-20		302	0.27	0.25	0.093
AE-17		806	0.70	27.5	3.93

* 250 to 700 thermal cycles between 110 and 500°C.
† Sum of surface plus central temperature divided by two. From calculated data.

REFERENCES

1. D. G. Harrington, "MTR Metal Fuel Tests for Advanced Sodium Graphite Reactors," NAA-SR-Memo 3215 (January, 1959)

2. B. R. Hayward and L. E. Wilkinson, "Radiation Behavior of Metallic Fuels for Sodium Graphite Reactors," NAA-SR-3411 (August, 1959)

3. G. G. Bentle, "A Study of Irradiation Swelling in SGR Fuels," NAA-SR-4893 (September, 1961)

4. J. L. Ballif et al., "Sodium Reactor Experiment Fuel Irradiation Experience," NAA-SR-3887 (September, 1959)

5. J. L. Ballif et al., "The Engineering Evaluation of a Mixed Alloy Experimental Fuel Element Irradiated at Elevated Temperatures in the SRE," NAA-SR-3888 (June, 1960)

6. Unpublished AI Data.

7. R. S. Barnes et al., "Swelling and Inert Gas Diffusion in Irradiated Uranium," Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, (Geneva, August, 1958) P 81

8. J. H. Kittle and S. H. Paine, "Effects of Irradiation on Powder Compacts of Uranium and Some Uranium Base Alloys," ANL-5664 (June, 1957)