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QUARTERLY PROGRESS REPORT
FUELS DEVELOPMENT OPERATION

APRIL, MAY, JUNE, 1961

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FUELS DEVELOPMENT OPERATION

APRIL, MAY, JUNE, 1961

By [REDACTED]

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July 15, 1961

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

Work performed under Contract No. AT(45-1)-1350 between
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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

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HW-47850	October, November, December, 1956.
HW-49803	January, February, March, 1957.
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HW-62656	July, August, September, 1959.
HW-64863	October, November, December, 1959.
HW-66094	January, February, March, 1960.
HW-67904	April, May, June, 1960.
HW-69085	July, August, September, 1960.
HW-70355	October, November, December, 1960.
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PHYSICAL METALLURGY

MECHANICAL AND PHYSICAL PROPERTIES OF IRRADIATED MATERIALS

The serrated yielding observed with unirradiated "A" nickel disappeared after irradiation. This indicates an effective pinning of substitutional impurities by irradiation defects.

Continued study of the post-irradiation strain-aging characteristics of copper, nickel, and zirconium has established that impurities play a major role in the strain-aging process.

A modified second generation capsule for measuring creep in-reactor was inserted into the reactor during the quarter. Six third generation capsules of improved design have been purchased.

The first specimen loading consisting entirely of Zircaloy-2 tensile specimens was prepared for the 280 C water, ETR 6X9 loop.

FUNDAMENTAL STUDIES

A 200 A thick layer of carbon on a UO_2 source proved to be an effective barrier to the transfer of UO_2 across a 1 mm vacuum gap during irradiation.

A series of irradiated aluminum foils show complex defect structures that vary with the pre-irradiation history of the foils, as well as the irradiation history.

Three general swelling capsules have completed their irradiation and two others are continuing to operate at a constant temperature.

The preliminary examination of bare uranium spheres irradiated at controlled temperatures has shown the formation of pores and cracks, but no change in grain structure. The observed swelling agrees well with that previously observed on clad rods irradiated with no external pressure, and on de-clad rods vacuum annealed.

Notches in flat Zircaloy-2 tensile specimens tested at 280 C resulted in a decrease in uniform elongation, but not as great as theory would predict. The effects of square shape notches were more severe than were V shape notches.

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METALLIC FUEL DEVELOPMENT

NEW FUELS DEVELOPMENT

Examination of uranium rods clad in 20 mils of Zircaloy-2 and irradiated in NaK capsules has shown that localized clad thinning is associated with areas in the clad in which the thickness is only 75-85 per cent of the nominal thickness over a circumferential gage length of 35-70 mils.

Eleven capsules, each containing two Zircaloy-2 clad fuel rods, have been discharged from the MTR and ETR. Many of the fuel rods have suffered cladding failures.

A program to investigate the ability of corrugated cylindrical fuel geometry to increase the swelling capabilities of metallic fuel elements has been initiated.

FABRICATION DEVELOPMENT

A closure utilizing a partially copper-plated Zircaloy ring electron-beam-welded into a recess in the fuel tube and subsequently bonded to the fuel is being evaluated. Some difficulty has been experienced with the copper-uranium bond, and a duplex plating of copper and nickel, with nickel adjacent to the uranium end face, is now being evaluated.

A second closure method, also using copper as a bonding medium, uses projection welding to both form the primary closure and to braze the Zircaloy cap to the uranium. The current work emphasizes the use of a tapered rather than flat end cap. End closures have been successfully welded into a tubular fuel element with "fluted" external surfaces by electron beam-welding.

Completion of braze closures by MIG welding is under study. To date, difficulties with weld profile and with braze contamination of the weld have prevented fabrication of acceptable closures. The reproducibility and control of the operation of spot welding Zircaloy supports to NPR elements has been studied. In a run of one thousand welds, produced at intervals with the machine returned to service for each run by simple control settings, 86 per cent were acceptable on the basis of shear tests showing strengths above one thousand pounds.

A series of experiments with N inner fuel elements, hot straightened in a three-roll straightener, have shown that elements can be hot straightened and will remain within acceptable warp limits after annealing or autoclaving.

A series of thirteen zirconium base alloys have been prepared for evaluation. Initial compositions were made up by double vacuum arc melting. Ingots were draw forged and hot rolled to 0.050-inch sheet and finally cold rolled to 0.040-inch sheet.

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FUEL EVALUATION

Single tube fuel elements have been irradiated in high temperature loops to exposures of 1200, 2000, and 3600 MWD/T. Measured fuel swelling on these elements ranged from one-half per cent to 5.4 per cent over a fuel temperature range of 400-470 C. Within the range of this data, fuel swelling is empirically related to the 1.7 power of burnup and temperature. Steel supports performed satisfactorily in these irradiation tests.

Comparative irradiation of uranium and uranium-2 w/o zirconium fuel elements indicates that at lower temperatures a somewhat greater swelling rate may exist in the uranium-2 w/o zirconium fuel.

Fuel element failure resulting from thermally-induced shear in the Zircaloy-2 clad has been observed. A failure occurred at the base of a flush end cap which was not bonded to the fuel.

An irradiation of two elements closed with an Fe-Be-Zr-2 brazing alloy was terminated by cap failure on one of the elements. Initial examination indicated a cladding defect. Further study, however, showed that the more probable cause was shear failure of the inner clad at the base of the end cap. This failure mechanism may have been accelerated by the extremely brittle nature of the braze employed.

A set of ten NPR inner elements, heat-treated with varying delay times and quench rates, was irradiated to 1050 MWD/T. The only major differences in fuel performance appeared to be related to charge position or specific power rather than to heat treatment.

In-reactor rupture tests continue to show that irradiated metallic uranium fuel elements suffer corrosion damage at an accelerate rate as compared to the unirradiated fuel. Fragmentation of the fuel is the most important factor contributing to the failure mechanism. Ex-reactor defect testing of irradiated metallic uranium fuels continues to confirm these observations.

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CERAMIC FUELS DEVELOPMENT

Examination of PRTR UO_2 fuel elements was performed at various stages of initial reactor operation. Each element was removed from the reactor, disassembled, thoroughly cleaned and inspected, and reassembled after the low temperature Critical Tests.

An internal desorbed gas pressure of 130 psi at 800 C was measured in an investigation of rupture hazard of PRTR swaged UO_2 fuel rods.

Three eight-foot PRTR nested tubular fuel elements (Mark II-C) were fabricated.

UC, ThO_2 , $\text{ThO}_2\text{-Al}_2\text{O}_3$, $\text{TiO}_2\text{-UO}_2$, electrodeposited UO_2 , and Micronized UO_2 were successfully compacted by high energy impaction.

Vibrational compaction efficiencies for fused UO_2 of six different particle size distributions varied considerably from the calculated efficiencies, probably because of topological characteristics of the UO_2 .

A stainless steel clad $\text{UO}_2\text{-ThO}_2$ fuel mixture, vibrationally compacted to 86 per cent TD, was fabricated for MTR irradiations.

Fused UO_2 was vibrationally compacted to 88-89 per cent TD in AISI-406 stainless steel cladding for irradiation testing at high coolant temperatures.

Inconel X, Hastelloy-F, Fansteel-80 and Zircaloy fuel rod cladding materials were employed in hot swaging studies of fused UO_2 . An improved induction heater coil design and better lubricants were investigated.

High frequency resistance welding has been demonstrated to be an effective procedure for continuously welding spacer ribs to Zircaloy fuel cladding.

A prototypic nested tubular fuel element developed a leak near the mid-point of the inner cladding on the inner fuel tube after receiving severe thermal treatment during test loop maintenance.

Zircaloy-4 continued to reveal a significantly greater resistance to hydriding than did Zircaloy-2 in irradiation tests of deliberately defected fuel elements.

Uranium dioxide was routinely compacted to bulk densities greater than 99 per cent TD by closed die forging of powders at 1000-1200 C in an explosive forming machine (impact pressure of 500,000 psi). Subsequent twelve-hour anneals at temperatures between 1300 C and 1700 C caused decreases in densities.

Inclusions in fused UO_2 were essentially eliminated by annealing in hydrogen for four hours at 1700 C. Gas evolution after annealing was one-fourth that for nonannealed samples.

Irradiation of wafers cut from single crystals of UO_2 began in the ETR.

Heterogeneous $\text{ThO}_2\text{-UO}_2$ mixtures (86 w/o ThO_2 clad in stainless steel) were irradiated in the MTR for 15 equivalent fuel power days at 600,000 Btu/hr ft².

Hot swaged UO_2 in a 12-inch long Zircaloy-4 clad, four-rod cluster was irradiated during two MTR cycles without incident.

Single crystal UO_2 spheres 0.10 to 0.25 inches in diameter were prepared and delivered for study in Euratom laboratories as part of an exchange of standard specimens and basic property information.

Uranium dioxide on a carbon substrate was examined in the electron-microscope at temperatures to 2000 C and at magnifications to 9000 X, using a specially designed hot stage.

Thermal and electrical conductivities of UO_2 specimens irradiated to 0.005 to 0.061 a/o burnup were measured at temperatures to 900 C.

Simultaneous heating of irradiated and nonirradiated UO_2 at temperatures to 3400 C further demonstrated the more refractory character of the irradiated material.

A hypothetical liquidus for the uranium-oxygen phase diagram has been established.

Centrally heated PRTR swaged UO_2 fuel rod segments have been tested under simulated in-reactor thermal-hydraulic conditions to a maximum surface heat flux of 310,000 Btu/hr ft². An estimated 60 per cent increase in average thermal conductivity of the fuel occurred during 30 minutes heating at 180,000 Btu/hr ft².

Five types of voids have been identified and correlated with specific grain structures that are developed in UO_2 fuel cores during irradiation at high central fuel temperatures.

PHYSICAL METALLURGYMECHANICAL AND PHYSICAL PROPERTIES OF IRRADIATED MATERIALSEffects of Irradiation on Non-Fissionable Materials - J. J. Laidler

Neutron Damage to Metals. The purpose of this program is to advance the theory of irradiation damage to metals by neutrons. Copper, nickel, titanium, zirconium, iron, molybdenum, and type 347 stainless steel specimens 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.

Recovery studies have been terminated for iron, nickel, copper, and zirconium. In cases where microhardness and electrical resistance recovery stages occurred over the same temperature range, the activation energies determined by the two measurements were often not in agreement. In addition, the activation energies determined by microhardness measurements for processes thought to be controlled by self-diffusion were about 20 per cent higher than accepted values for self-diffusion. The techniques used in determining the activation energy of a single process are based on the assumptions that the recovering property is linearly proportional to the number of defects and that the defects are randomly distributed. The former assumption is more nearly valid for measurement of electrical resistance than for microhardness. The latter assumption is probably valid in the case of electron-irradiation at extremely low temperatures; neutron irradiation, on the other hand, is characterized by clustering of defects. Another complicating factor in irradiations conducted at reactor ambient temperatures (300-350 K) is the fact that the distribution of defects changes during irradiation. Without refinement of the mathematical description of the recovery process, the results must be considered as not much better than qualitative. Results of these recovery studies are summarized in Table 3. 1. The high values for activation energy of the high-temperature processes observed in Fe, Ni, Cu, and Zr suggest that the processes may be self-diffusion. Only in copper, however, is the value of the jump frequency calculated from the recovery kinetics of the proper order of magnitude. The other processes probably involve complex defect-dislocation interactions.

As reported previously (HW-59300 A, March 1, 1959), yield points were observed in electrolytic nickel and "A" nickel after exposure to 1.2×10^{18} nvt (fast), and copper, iron, zirconium, and titanium after exposure to approximately 1×10^{20} nvt (fast). The development of a yield point upon irradiation appears to be independent of starting purity for face-centered cubic and close-packed hexagonal metals. This indicates that the mechanism must involve hardening of dislocation sources by radiation-induced defects, and is not a secondary effect of defect-solute atom interactions. The response of these metals to post-irradiation strain-aging treatments is, however, markedly dependent upon the purity of the material. Copper and "A" nickel,

TABLE 3. 1
DAMAGE RECOVERY IN IRON, NICKEL, ZIRCONIUM, AND COPPER

Metal	nvt(Fast)	T, °K*	T/T _m **	Activation Energy E, ev	Activation Energy for Self- Diffusion, ev	Reaction Order	Measuring*** Technique	Proposed Process
Fe	1×10^{17}	503	0.28	1.15		1.0	H	Precipitation of interstitial impurities
Fe	4×10^{18}	448	0.25	1.7		1.1	H	Breakup of defect clusters
Fe	4×10^{18}	518	0.29	Spectrum		---	H	Release of trapped interstitials
Fe	4×10^{18}	598	0.33	1.2 2.2		1.3 2.1	H R	Precipitation of interstitial impurities
Fe	1.5×10^{20}	648	0.36	a) 1.3 1.8 b) 3.7		1.2 1.0 ---	H R H, R	a) Release of trapped interstitials b) ---
Ni	1.0×10^{19}	748	0.43	3.4	2.8	1.1	R	---
Zr	4×10^{18}	548	0.32	1.0	---	---	H	---
Zr	1.5×10^{20}	838	0.49	a) 0.1 b) 4.2	2.3-2.8	---	H H	a) --- b) ---
Cu	1.5×10^{20}	598	0.44	1.8	2.0-2.1	---	H	Self-diffusion

*T = Temperature at which maximum recovery rate occurs.

**T_m = Absolute melting temperature of metal.

***H = Microhardness

R = Electrical resistance

both face-centered metals containing substantial amounts of impurities (1-2 per cent zinc formed in copper by transmutation, 0.8 per cent impurities present in "A" nickel prior to irradiation), showed strain-aging reactions when tested after irradiation. Figure 3.1b shows the effects observed in irradiated "A" nickel. It is interesting to compare this behavior with that of unirradiated "A" nickel, Figure 3.1a. The serrated yielding observed in unirradiated "A" nickel may be explained by the migration of solute atoms to extended dislocations during deformation. This effect should be quite sensitive to temperature, amount of plastic deformation, and defect concentration. The diffusion coefficient of substitutional atoms may be approximated by the equation (A. H. Cottrell, Phil. Mag. 44: 829. 1953)

$$D \approx CA^2 \nu \exp(-U/RT),$$

where C is the atomic concentration of vacancies, A the interatomic spacing, ν the vibrational frequency, and U the activation energy for vacancy migration. The concentration of vacancies created by plastic deformation is approximately $10^{-4} \epsilon$ (ϵ = strain). Figures 3.2 and 3.3 show results of tensile tests of irradiated and unirradiated "A" nickel at the same temperature and strain rate. No serrations can be seen in the flow curve for the irradiated material. From the equation given previously, it would be expected that serrations should appear in the flow curve of irradiated "A" nickel at a lower strain, due to increased vacancy concentration as a result of irradiation. That they do not indicates that the substitutional atoms in irradiated "A" nickel are effectively pinned by defects which have clustered about them, and the concentration of solute atoms which are free to diffuse is not great enough to cause transitory dislocation locking with resulting serrated yielding. Irradiated electrolytic nickel and zirconium did not show strain-aging effects, again probably due to low solute atom concentrations.

FIGURE 3.1

- (a) Tensile Test of Unirradiated "A" Nickel; Tested at 22 C,
 $\dot{\epsilon} = 8.33 \times 10^{-5} \text{ sec}^{-1}$ (all specimens in Figures 3.1 and 3.2 had same cross-sectional area)
- (b) Tensile Test of Irradiated "A" Nickel; Exposure $3.2 \times 10^{18} \text{ nvt (Fast)}$
Tested at 22 C, $\dot{\epsilon} = 8.33 \times 10^{-5} \text{ sec}^{-1}$

FIGURE 3.2

Yielding Phenomenon in Unirradiated "A" Nickel;
Tested at 20° C, $\dot{\epsilon} = 8.33 \times 10^{-5} \text{ sec}^{-1}$

FIGURE 3.3

Repeated Yielding Tests of Irradiated "A" Nickel;
 Exposure 1.4×10^{18} nvt (Fast), Tested at 22 C,
 $\dot{\epsilon} = 8.33 \times 10^{-5} \text{ sec}^{-1}$

In-Reactor Measurements - L. J. Chockie, J. J. Holmes, and J. C. Tobin

The behavior of materials under the influence of fast neutron irradiation is of prime importance in the design of nuclear reactors. The objective of the In-Reactor Measurements Program is to determine the physical and mechanical behavior of materials during fast neutron irradiation. Currently, a study of the in-reactor creep properties of Zircaloy-2 is being conducted. A prototype creep test capsule has been successfully operated for more than 10,000 hours in a Hanford Reactor. Creep data from the prototype capsule demonstrated the extent to which neutron radiation can alter the creep of Zircaloy-2. In order to continue the study of the in-reactor creep of Zircaloy-2, four creep capsules of advanced design were fabricated by an off-site vendor. The features of these second generation capsules are discussed in the Fuels Development Quarterly Report for October, November and December, 1960, HW-70355. In-reactor operation of a second generation capsule displayed a weakness in the thermocouples and excessive gamma heating in the areas surrounding the two ends of the Zircaloy specimen. The three remaining capsules were returned to the vendor for thermocouple modification and removal of some excess mass from the pull-rod sections at the ends of the specimen which had contributed to the high gamma heating.

In-Reactor Studies. One of these modified second generation capsules, designated as Capsule II-2, was placed in a Hanford reactor during the quarter. The specimen temperatures after reactor startup due to gamma heating alone were 375 C at one, 420 C at the middle, and 460 C at the other end. An effort to lower these temperatures by increasing internal helium pressure was made, but the helium pressure increases resulted in temperature increases rather than anticipated reductions. The irradiation of the prototype capsule and one other second generation capsule previous to the charging of Capsule II-2 had confirmed the fact that increasing internal pressure in the capsules would result in decreases in specimen temperatures. The heat transport within the capsules was also studied in a thermal mockup capsule operated from 15 to 1,000 psi of helium pressure. It was determined that convection was the controlling mode of heat transport. An increase in pressure resulted in increased convective current flow, and consequently an increase in heat transport. However, the modified capsule, Capsule II-2, did not respond with increasing pressures as expected; in fact, the results were exactly opposite from the other two capsules and the laboratory mockup. The change in pressure response can only be a result of a change in governing heat transport mode from convection to some other process. It was suggested that helium contamination might result in the observed behavior, but spectrographic analysis of helium exhausted from the capsule indicated this is not the case. Other possible explanations for the change in transport mode are: (1) a film might have formed on the capsule wall which greatly reduced the efficiency of convective flow allowing another mode to govern, and (2) a different heat transfer path may have been established by the movement of capsule parts with pressure. The pressure response was not determined prior to charging, so the extent of radiation effects on the heat transport in Capsule II-2 cannot be estimated.

The high specimen temperatures on Capsule II-2 are the result of two factors, one is the loss of convective heat transport and the other is the gamma heating of the structural components within the capsule. Initially the capsule was designed for a gamma heating value of 0.5 watts/gram of stainless steel. The actual gamma heating encountered was 1.2 watts/gram. The metal removed during the modification procedure consisted of two Fansteel-77 slugs that were placed near the ends of the specimen to adjust the thermal gradient. This weight removal at the ends of the specimen should have resulted in a temperature between 300 and 350 C. However, the loss of heat transport control resulted in actual temperatures as high as 460 C.

Only two ranges of the active three range transducer in Capsule II-2 were operable at the time of startup. After nine days of operation in the reactor the high range failed and after thirty days, the low range failed. The transducers failed in a manner identical to those of the first of the second generation capsule to be tested in the reactor by shorting of the coils. This resulted from excessive transducer temperatures. The mechanical screw extensometer in the capsule is still functioning and can be used to determine strain in this capsule.

Capsule and Instrument Development. Six creep capsule of improved design have been purchased to continue the in-reactor creep study. These six capsules are designated as third generation capsules. They differ from the second generation capsules mainly in the weight of the components contributing to gamma heating near the test specimen and in the method of strain measuring. The two three range variable permeance transducers used in the second generation capsules have been replaced with a single system employing a single range variable permeance transducer and a coupled mechanical screw micropositioner. Construction of the third generation capsules eliminated all excess metal from the area surrounding the test specimen. All parts not bearing significant loads were lightened by drilling and machining. The pull rods are hollow cylinders having wall thickness just great enough to safely strain the specimen without undergoing significant creep deformation during the test. Removal of the gear box, which is a component of the micropositioner, from the immediate area of the specimen along with the elimination of the two three range transducers further reduced the mass near the specimen. The total weight of the test section is 383 grams in the third generation capsule as compared to 732 gms in the second generation. It is predicted that the specimen temperature will be less than 200 C without electrical power to the capsule heaters.

The extensometry of third generation capsule is simpler and, hopefully, more reliable than that of Type II capsules. A transducer is mounted at one end of the specimen and the micropositioner at the other end of the specimen as shown in Figure 3.4. On the micropositioner shaft is mounted the transducer core of the variable permeance transducer, and an electrical probe for the micropositioner. As the specimen elongates the transducer windings move over the core and send a signal indicating the extent of travel to monitoring equipment outside the reactor. After a period of time the micropositioner is

FIGURE 3.4
Creep Capsule-III

activated, and the core advanced to the same relative position within the transducer as that before specimen elongation was started. The initial position is determined by both transducer readings and a reference point built into the micropositioner which makes contact with the electrical probe. Strain, therefore, can be determined with two independent measuring systems. The transducer will give continuous creep data, and the micropositioner will be used mainly to provide transducer calibration and repositioning in-reactor. In the event of transducer failure, the micropositioner alone can be used to follow the creep strain. A 2,000:1 gear box is used in the third generation capsules rather than the 20,000:1 box used in generation II. The 2,000:1 box is expected to operate in excess of 500 hours continuous use which should allow at least a 5000 hour creep test to be conducted. The lower reduction gear box has resulted in a loss of sensitivity of the micropositioner system. A strain of ± 20 microinches is the minimum resolvable strain. The variable permeance transducer is capable of 400 microinches resolution.

Irradiation and Environmental Effects in Structural Materials - A. L. Bement

As reactor concepts become more diversified, it is increasingly more difficult to keep pace with the need for information about the effects of irradiation on reactor structural materials. The purpose of this program is to investigate the combined effects of radiation and reactor environment on the mechanical properties of structural materials. Special attention will be given to the determination of mechanical property changes produced in metals by irradiation at elevated temperatures in contact with water. Materials to be investigated include zirconium base alloys and other alloys showing promise for high-temperature applications. Tensile and bend properties of these metals after irradiation under specific reactor operating conditions are to be determined. Notched tensile specimens will be employed to determine brittle failure tendencies in the irradiated materials, with the objective of predicting conditions under which catastrophic failure is likely to occur. Irradiations of Zircaloy-2 and types 304, 348, 410, and AM-350 stainless steel are scheduled to commence during cycle 39 of the ETR. The primary variables in these irradiations are shown in Table 3. 2.

The irradiations are to be conducted in the ETR 6x9 loop, which has been modified for this purpose. The outlet water temperature of the loop will be controlled at 280 C by adjusting the flow. The feed is demineralized water which is adjusted to pH 10 by LiOH additions. A flow of 3 to 7 gpm is split from the main stream for cleanup through a cation-anion resin bed.

The first specimen loading consists entirely of Zircaloy-2 tensile specimens. The fixtures for these specimens are shown in Figures 3.5 and 3.6. Figure 3.5 illustrates a tensile specimen quadrant which accommodates eighteen specimens. The quadrants are supported by means of studs on a holder assembly shown in Figure 3.6. The holder assembly is suspended in the loop from a quick-disconnect coupling, which facilitates removal and replacement of the assembly after each irradiation cycle. A total of 504 tensile specimens can be loaded into the assembly. As the Zircaloy-2

TABLE 3. 2

IRRADIATION VARIABLES IN STRUCTURAL MATERIALS IRRADIATIONS

<u>Material</u>	<u>Orientation</u> ¹	<u>Condition</u>	<u>Specimens</u>	<u>Range of Goal Exposures (nvt)</u>
Zircaloy-2	Transverse Longitudinal	Annealed 20 per cent cold work 40 per cent cold work	Tensile Notch tensile Bend	$1.6 \times 10^{19} - 1.5 \times 10^{22}$
304 SS	Transverse	Annealed 25 per cent cold work	Tensile	$5.9 \times 10^{20} - 3.9 \times 10^{21}$
348 SS	Transverse Longitudinal	Annealed 25 per cent cold work	Tensile Notch tensile Bend	$5.9 \times 10^{19} - 3.9 \times 10^{21}$
410 SS	Transverse	Annealed ² Martensitic ³	Tensile	$5.9 \times 10^{19} - 1.7 \times 10^{21}$
AM-350 SS	Transverse Longitudinal	Stabilized ⁴ Double Aged ⁵	Tensile	$5.9 \times 10^{19} - 1.7 \times 10^{21}$

¹Orientation is with respect to rolling direction.

²Annealed at 815 C, furnace cooled to 600 C, then air cooled.

³Oil quenched from 1,000 C.

⁴Solution treated at 1,040 C and water quenched

⁵Solution treatment followed by three hours at 750 C, air cooled to room temperature plus three hours at 455 C, and air cooled.

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FIGURE 3.5
Tensile Specimen Quadrant

FIGURE 3. 6
Specimen Holder Assembly

specimens reach their goal exposure, they will be discharged and replaced with types 348 and 304 stainless steel tensile specimens. In this way the assembly is kept loaded at all times.

Prior to irradiation the Zircaloy-2 specimens were pickled and autoclaved to form a continuous protective film. The solutions used for pickling are as follows:

1. Pickling solution - 4 parts by volume 48 per cent hydrofluoric acid
39 parts by volume 70 per cent nitric acid
57 parts by volume deionized water (100,000 ohm-cm or better)
2. Stop bath - 15 weight per cent aluminum nitrate monohydrate ($\text{Al}(\text{NO}_3)_3 \cdot 9 \text{H}_2\text{O}$) in 1 N nitric acid.
3. First rinse - 50 C tap water under flow.
4. Second rinse - deionized water (100,000 ohm-cm or better) under flow.

The first autoclave treatment was conducted in a water autoclave at 300 ± 25 C for 72 hours. Most of the Zircaloy-2 specimens developed a white corrosion film near the point of contact with the 304 stainless steel holder. This was attributed to contamination from residual trace amounts of pickle solution on the surface of the stainless steel. It was necessary to remove the specimens from their holders, vapor blast to obtain a fresh surface, re-pickle, and re-autoclave. The second autoclave treatment was conducted in a steam autoclave at 400 ± 10 C for 72 hours. This treatment gave a uniform, black-oxide coating of high integrity.

Since the specimens are to be irradiated near 300 C for extended periods of time, partial recovery of the cold-worked specimens and structural modifications of some of the heat treated specimens can be expected. To divorce these effects from neutron damage effects on the over-all properties; it is necessary to carefully duplicate the in-reactor loop operating variables in an out-of-reactor facility. Such a facility is available, which has three, five-foot test sections constructed of 2.30-inch, heavy-walled tubing. This loop has automatic instrumentation for controlling and recording temperature and flow, and has auxiliary facilities to adjust water quality. From the operational data logs and water samples from the in-reactor loop it will be possible to closely duplicate in-reactor operating history for the control specimens.

To date, approximately 1,900 Zircaloy-2 and 1,000 stainless steel tensile specimens, as well as blanks for the notched tensile and bend specimens have been machined. An irradiation assembly for the 6x9 loop and a mockup assembly for the critical facility have been delivered to the ETR and await charging. The design of the necessary handling tools both for transferring the specimen assembly in and out of the loop and for replacing quadrants has been completed.

Low temperature irradiations are also in progress on Zircaloy-2 and zirconium 2Sn-2Nb specimens under the conditions shown in Table 3.3. The

sample assembly employed for these irradiations is shown in Figure 3.7. Nineteen of these assemblies were charged into the ETR F6 and G6 positions. Eight of these have been discharged after reaching their respective goal exposures, and the specimens are awaiting testing.

TABLE 3. 3

AMBIENT TEMPERATURE IRRADIATIONS OF ZIRCALOY-2
AND ZIRCONIUM - 2Sn - 2Nb ALLOY

<u>Material</u>	<u>Orientation</u> ¹	<u>Condition</u>	<u>Specimens</u>	<u>Exposure Range (nvt)</u>
Zircaloy-2	Transverse	Annealed	Tensile	10 ¹⁹ -10 ²²
	Longitudinal	10 per cent cold work	Notch tensile	
		20 per cent cold work		
		40 per cent cold work		
		70 per cent cold work		
Zirconium - 2Sn - 2Nb		Annealed ²	Tensile	10 ¹⁹ -10 ²²
		Hardened from 780 C ³		
		Hardened from 740 C		
		Hardened from 700 C		
		Hardened from 660 C		

¹ With respect to rolling direction.

² Held at 850 C for one hour and furnace cooled 25 C deg/hour

³ Quenched in silicone oil after holding at respective temperatures for 4 hours

FIGURE 3.7

Sample Assembly for ETR Low-Temperature Irradiations

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FUNDAMENTAL STUDIES

Effect of Irradiation on Thin Films - T. K. Bierlein and B. Mastel

The effect of fission fragment and neutron bombardment on a variety of metals is being studied by microscopy techniques. This method of study shows great promise in the field of radiation damage because defect structures such as vacancy clusters, interstitial loops, dislocation interactions, and nucleation-growth processes can be studied.

A formal report, HW-68919, "Experimental Evidence of Fission Fragment Damage in Some Metals and Non-Metals" has been written, and describes the type of fission fragment damage observed in thin films and foils.

An experiment aimed at establishing the physical process of fission fragment track registration in UO_2 films has been conducted. A thin, 80 Å thick UO_2 source was placed at a distance of 1 mm from a 200 Å thick carbon film, and after irradiation to 4×10^{18} nvt (thermal) in an evacuated capsule, the carbon collector was examined. Similar experiments utilizing a carbon coated UO_2 source, as well as a platinum coated carbon collector, were conducted. Figure 3.8 illustrates the effectiveness of the 200 Å thick carbon layer on the UO_2 source in reducing the evaporation of UO_2 from the source onto the amorphous carbon collector. The energy transfer to the UO_2 by the fission fragments is sufficient to vaporize UO_2 from the surface of the UO_2 source, along the path of the fission fragment. The condensed particles were identified as UO_2 by electron diffraction. Only faint fission fragment tracks in the carbon collector are detectable. In the case of the collector consisting of a thin Pt coat on the 200 Å thick carbon, tracks in the collector are quite apparent, Figure 3.9. In the experiment involving a carbon coated UO_2 source opposite the Pt-coated, carbon collector, fission fragment tracks are short, and due to fragments which originated at the source. The longer tracks in the collector opposite the uncoated UO_2 source are due to fission fragments which originate from condensed UO_2 on the collector surface. Again electron diffraction identified UO_2 only on the Pt collector opposite the uncoated UO_2 source. Since the collector films are supported on conventional electron microscope grids made of copper, a considerable number of fission fragment atoms come to rest within the copper grid. Gamma ray spectrometry definitely shows fission product activity associated with these grids. Such a grid has been re-irradiated to further confirm by the increase in fission product concentration the presence of fissionable uranium atoms on the collector. The use of gamma ray spectrometry, coupled with fission fragment damage studies employing selective energy cut off filters or absorbers is in a preliminary stage. The technique may, however, be useful in establishing fission fragment energy-mass-damage relationships in specific metals.

Since fission fragment damage has been observed in thin evaporated films, as well as foils, an experiment to determine the extent of damage in a bulk specimen is in progress. High purity, annealed aluminum wires were electrolytically thinned to a diameter of approximately 0.003 inch. Uranium

dioxide was then evaporated against one-half of the wire surface and the wires were irradiated in a Hanford Quickie Facility. Current testing of control wires indicate that such wires can be sectioned after embedding in epoxy resin with an ultra-microtome equipped with a diamond knife. It is hoped that the sectioning will not completely obliterate the fission fragment damage that may persist within the wire after irradiation, and that fission fragment damage within bulk specimens may be studied by this technique.

A series of high purity aluminum specimens, in the form of thick, 0.003-inch foils and 0.0005-inch foils, in cold worked and annealed states have been irradiated to doses as high as 2×10^{20} nvt (thermal) and then thinned for examination. Some defect structures which appear in the irradiated foils are illustrated in Figure 3.10 and 3.11. In annealed foils, dislocation lines are few in number, whereas in the cold worked foils, dislocation tangles are quite prevalent. After irradiation small loops, similar to loops formed by collapse of vacancies in aluminum quenching experiments, are present in the annealed specimens, and their number increases with increasing irradiation dose. A larger number of these small loops form in the cold worked specimens than in the annealed. The most striking difference between these two types of specimens is that relatively large loops are also present in the cold worked specimens, and the size of these loops increases with increasing irradiation dose. Whether these loops are due to clustering of interstitial atoms or due to dislocation lines wrapping themselves about the smaller loops is not known. An alternative explanation may also be applicable. Impurity atoms in the aluminum foil, or defects in the foil, can lock dislocations which due to stress fields in the foil cause Frank Reed type loops to form and enclose the source. Unfortunately, crystallographic identification of the loops is not possible with the electron microscope equipment on hand. Irradiation of aluminum containing collapsed vacancy disks (quenched high purity aluminum) may afford qualitative information on the nature of the large irregularly shaped loops seen in the aluminum. Since interaction of existing loops with loops formed by irradiation may lead to a different final state, aluminum foils containing quenched-in vacancy loops will be irradiated. Foils, such as molybdenum, which have a high melting point and, therefore, less recovery of damage at room temperature, are being prepared for similar irradiations.

FIGURE 3. 8

Direct Evidence of Transfer of UO_2 Through a Vacuum Gap onto a 200-A Thick Carbon Collector Due to Fission in the UO_2 Source 77,000 X

Left: No transfer due to a 200-A thick carbon barrier between source and collector.

Right: Considerable transfer in the absence of a barrier.

FIGURE 3. 9

Direct Evidence of Fission Fragment Damage to a Platinum Coated Carbon Film Positioned 1 mm from a UO_2 Source During Irradiation 77,000 X

Left: Only short tracks and holes appear in the collector when a 200-A thick carbon coating is on the source.
 Right: Much longer tracks form due to fragments from UO_2 particles condensed on the collector in the absence of a barrier.

FIGURE 3. 10

Two Examples of Lineage of Loops in Annealed High Purity Aluminum
Irradiated to 2×10^{20} nvt (Thermal) 77,000 X

Interaction of these small loops with a dislocation line is indicated.

FIGURE 3. 11

Typical Dislocation Network Interspersed with Small Loops and
Large Rings Observed in Cold-Worked High Purity Aluminum
Irradiated to 2×10^{20} nvt (Thermal) 77,000 X

Basic Swelling Studies - T. K. Bierlein, B. S. Kosut, R. D. Leggett,
B. Mastel and J. W. Weber

When uranium is irradiated at elevated temperature, drastic increases in volume are frequently observed. This swelling is thought to be due to the accumulation of the inert fission gases, krypton and xenon, in pores in the metal matrix. It is thought that the temperature, burnup, burnup rate, fuel restraint and metal parameters such as purity, residual work, grain structure, and second phase distribution may influence the volume increase. The exact nature and magnitude of this dependence, however, is not known. This investigation is being made to establish the precise influence of these variables on swelling.

Irradiation Program. Two types of capsule have been developed to irradiate uranium at a constant temperature under conditions of low restraint-- a general swelling capsule for horizontal irradiation and a metallographic swelling capsule for vertical irradiation. The present emphasis on this phase of the swelling program is directed toward the assembly of general swelling capsules. Components for about six metallographic capsules are being stored for future use.

Three general swelling capsules, #4, 5, and 6, have completed their irradiation; the samples have been recovered, and post-irradiation examination has been initiated. The latter is discussed in the section entitled "Post-Irradiation Examination".

The irradiation of capsules #7 and 8, each containing three, axially split, hollow cylinders of uranium has continued at the constant temperatures of 525 C and 575 C, respectively. Figure 3. 12 illustrates the typical thermal behavior of these capsules as the reactor goes through a shutdown. Obviously, only one point can be precisely maintained at the control temperature; all others vary as the reactor changes drastically in power level. This temperature variation is, as expected, most severe with the end specimens, but the thermocouple located 180 degrees from the control couple, but under the other half of the same cylinder, experiences as much as a 10 C variation in temperature. It is not clear whether this is a real difference in specimen temperature or whether it simply reflects the great sensitivity of thermocouple position when the reactor is up. At any rate, it imposes a 10 C uncertainty in the specimen temperature. About the only means of improving this would be to actually weld the thermocouple to the sample surface. As this is impossible from the swelling standpoint, ± 10 C will be accepted as being the best control that can presently be attained.

Since the end specimens of capsules #7 and 8 change temperature most drastically, an evaluation can be made after irradiation of the effects of thermal cycling on swelling (compare specimen #3 with #2) and also the influence of beta annealing on this effect (compare specimen #1 with #3). This comparison will be approximate and will require numerous capsules to establish definite trends, but it provides a means of obtaining valuable additional basic data at the same time that swelling is being characterized as to constant irradiation temperature, burnup and burnup rate.

Capsule #9 was completely assembled and successfully bench tested to 700 C. During the quality control testing of this capsule, there was an indication of a helium leak in the inner chamber. Although no NaK could be detected, it is not planned at this time to charge this capsule into a reactor. As a test capsule has been sorely needed for some time, capsule #9 will be used for testing purposes. It will be used in the very near future to check out a new control instrumentation system that was recently installed at the reactor. If this system proves adequate, it will be employed in the future to control the temperature of swelling capsules.

Five additional general swelling capsules, #10, 11, 12, 13, and 14, are being assembled simultaneously and are about 30 per cent complete. These each contain three axially split, hollow uranium cylinders, two of which are as-extruded and the third beta annealed. It is planned to charge two capsules per reactor test tube instead of the present practice of one per test tube.

The annealing of two laboratory capsules, #5 and 6, has been completed and three more, #4, 7, and 8, are now in the annealing facility. These capsules contain specimens identical to their in-reactor counterparts and, as near as possible, the in-reactor thermal histories of the specimens are being duplicated. Comparison will be made to determine how many of the changes that occur in the irradiated material are attributable to temperature only, and how many are attributable to a combination of temperature and a neutron flux.

An electrical resistance heater specification has been prepared (HW-68067) and an invitation for bids has been issued for thirty-five heaters to be used for future capsules.

The ambient temperature irradiation of two unmonitored NaK filled capsules, GEH-14-281 and 282, was completed after 48.7 days total irradiation in the MTR in a flux of about 5×10^{13} nv. The exposure received by the capsules was reported to be 1.7×10^{20} and 1.8×10^{20} nvt, respectively. Each capsule contains a precharacterized, coextruded and heat treated U-U diffusion couple for studying fission product diffusion from enriched uranium into depleted uranium. The capsules are en route to Radiometallurgy where they will be opened and the specimens examined. Additional capsules are being fabricated to irradiate more samples of this type.

Post Irradiation Examination. Examination in Radiometallurgy has been initiated on the specimens from general swelling capsules #4, 5, and 6. One sphere from capsule #4 was lost during the opening of the capsule and one split hollow cylinder from capsule #6 had an end removed inadvertently. The pertinent irradiation histories are summarized in Table 3.4 along with the post-irradiation description of the specimens. The macro appearance of the specimens as they were removed from the capsules are depicted in Figures 3.13 and 3.14. As can be seen, most of the specimens retained their original shape. Physical measurements of the spheres from capsule #4 showed no change in either shape or size. Because of the peculiar propensity of uranium to flow into very small crevices when irradiated with a cladding, there was some speculation that perhaps these bare spheres would sag under their own

FIGURE 3. 12

Typical Thermal Behavior of General Swelling Capsules #7 and 8

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A - Capsule #4 Specimens
2 X

B - Capsule #5 Specimens
3 X

FIGURE 3. 13

The Appearance of Irradiated Uranium Spheres
Immediately After Removal from the Capsules

End

Side

A - One-Half of Specimen 6-1
 β -Heat Treated Priot to Irradiation $\sim 2\frac{1}{2} \times$

End

Side

B - One-Half of Specimen 6-2
Irradiated As-Extruded $\sim 2\frac{1}{2} \times$

FIGURE 3. 14

A Comparison of As-Extruded and β -Heat Treated Uranium
Irradiated to 0.034 a/o Burnup at About 330 C

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weight during irradiation simulating the behavior of "silly putty". This obviously did not occur indicating that uranium has measurable strength under high temperature irradiation. Specimens 5-1 and 5-3 also retained their original form, but specimen 5-2 had split open badly on one side. The temperature of this latter sample was about 800 C when the reactor was on, so it is not surprising that severe damage resulted. The reason for the high operating temperature is not too clear. The sample was enriched to 3 per cent U²³⁵ which drastically increased the rate of heat generation, but whether this alone could account for the high temperature is unknown. The condition of the specimen suggests that one side was hotter than the other which is possible if a gas pocket had collected on this sample. This pocket would have interfered with heat transfer leading to an abnormally high temperature when the reactor was on.

TABLE 3. 4

THE IRRADIATION HISTORIES OF SPECIMENS
IRRADIATED IN CAPSULES #4, 5, AND 6

<u>Specimen Number</u>	<u>Type of Specimen</u>	<u>Volume Avg. Irr. Temp. °C*</u>	<u>Burnup a/o**</u>	<u>Remarks</u>
4-1	Nat. sphere	575	0. 28	Sample lost
4-2	Nat. sphere	615	0. 28	Badly cracked
4-3	Nat. sphere	615	0. 28	Badly cracked
4-4	Nat. sphere	575	0. 28	Badly cracked
5-1	Nat. sphere	575	0. 046	Cracked
5-2	3 per cent En. sphere	800	0. 16	Severely cracked and swollen
5-3	Nat. sphere	550	0. 046	Cracked
6-1	Nat. half cylinder	330	0. 034	Warped and bumped
6-2	Nat. half cylinder	330	0. 034	Slightly warped
6-3	Nat. half cylinder	260	0. 034	Slightly warped

* Arithmetic average of the surface temperature and center temperature.

** Estimated from reactor operations data.

The specimens from capsule #6 provided an interesting observation with regard to dimensional instability, but the irradiation temperature was too low for swelling. A pronounced difference was observed between specimens that were beta heat treated prior to irradiation, and those that were irradiated in the as-extruded (high alpha) condition. As seen in Figure 3.14 a, the beta treated specimen was badly warped and showed considerable surface bumping while the as-extruded specimens, as seen in Figure 3.14 b, essentially retained their original appearance. The explanation for this behavior is believed to lie in the fact that the beta treated specimen has very large grains (one or two across the thickness of the sample) while the as-extruded specimens had very small, though oriented grains. Thus, under these conditions and for this particular geometry, the grain size effect overshadowed the preferred orientation effect.

When the spheres were polished and examined at high magnification, numerous cracks were observed that appeared to be intergranular in nature. This cracking was more extensive in the high burnup specimens (capsule #4), see Figure 3.15 b, than it was in the low burnup spheres (capsule #5), see Figure 3.15 a. No significant difference could be detected, however, between specimens irradiated to the same burnup, but at slightly different temperature, 615 C and 575 C for capsule #4 and 575 C and 550 C for capsule #5. The uranium hollow cylinders irradiated at about 300 C to 0.03 a/o burnup exhibited no cracking. This lack of cracking in the hollow cylinders corroborates numerous previous observations on bare uranium irradiated at temperatures below 300 C to burnups as high as 0.26 a/o. It would appear that cracking is associated with high temperatures and becomes more severe as the burnup increases.

The density measurements made on the as-irradiated samples are summarized in Table 3.5. As can be seen, the spheres of capsule #4, irradiated to 0.28 a/o BU, exhibited a 7 to 8 per cent increase in volume, while those of capsule #5, irradiated to 0.046 a/o BU, exhibited an increase of 2 to 2.5 per cent. It was not possible to measure the density on specimen 5-2 because of the condition of the sample. The specimens in capsule #6 did not exhibit a definite density change. To provide a basis for comparing the volume changes observed with the spheres with the two burnup levels, the per cent volume change was divided by the per cent burnup giving an "R" value. These values are also included in Table 3.5. The "R" values for the low burnup samples are about twice those for the high burnup material. From what is presently understood about swelling, this does not appear logical. This apparent anomaly is probably associated with the cracking that occurred. A large crack that is wetted by the immersion fluid will not influence a density measurement but small cracks that are not wetted or internal cracks that do not connect with the surface will lead to low density values. In the case at hand, if the volume contribution of nonwetted cracks is assessed at about 1 per cent, the "R" values for samples of both burnups become much more consistent. All of the values then fall in the range of 20-25. The validity of this proposal concerning the contribution of nonwetted cracks to the volume change will be established by evaluating the swelling due to pores with statistical quantitative metallography.

50 X

(a) Specimen 5-3, 0.046 a/o Burnup

50 X

(b) Specimen 4-3, 0.28 a/o Burnup

FIGURE 3.15

A Metallographic Comparison of the Cracking that Occurred
in Uranium Spheres Irradiated to Two Burnup Levels

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TABLE 3.5
THE DENSITY OF SPECIMENS IRRADIATED
IN CAPSULES #4, 5, AND 6

<u>Specimen Number</u>	<u>Date</u>	<u>As-Irrad. Density, g/cc</u>	<u>$\Delta V/V$ Per Cent</u>	<u>"R"</u>
4-2	4-28-61	17.49	7.5	25
	5-3-61	17.24	9.1	30
	5-5-61	17.55	7.1	24
	Avg.		<u>7.9</u>	<u>26</u>
4-3	4-28-61	17.34	8.4	28
	5-3-61	17.46	7.7	26
	5-5-61	17.41	8.0	27
	Avg.		<u>8.0</u>	<u>27</u>
	4-28-61	17.39	8.1	27
	5-3-61	17.53	7.2	24
	5-4-61	17.72	6.1	20
	5-5-61	17.73	6.0	20
	Avg.		<u>6.8</u>	<u>23</u>
5-1	5-1-61	18.43	2.0	44
	5-6-61	18.40	2.2	47
	5-8-61	18.51	1.6	34
	Avg.		<u>1.9</u>	<u>42</u>
5-3	5-1-61	18.43	2.0	44
	5-6-61	18.31	2.7	58
	5-8-61	18.28	2.8	62
	Avg.		<u>2.5</u>	<u>55</u>
6-1	5-9-61	18.96	---	--
	5-10-61	18.96	No significant change	
6-2	5-9-61	18.98	---	--
	5-10-61	18.98		
6-3	5-9-61	18.98	---	--
	5-10-61	18.97		

*"R" is defined as the per cent volume change divided by the per cent burnup.

Several replicas of the polished and etched spheres have been prepared and the replicas are now being processed in the electron microscope laboratory. Figure 3. 16 illustrates the appearance of a typical area on specimen 4-2. Extensive porosity is apparent. As was the case for specimens subjected to post-irradiation annealing, described in previous reports, this specimen also shows a considerable range of pore sizes. In fact the large pores, approximately 10,000 Å in diameter, deviate from a spherical shape and are located at preferred sites such as at grain boundaries. These observations are quite different from those made in the UK, where pores are apparently uniformly dispersed, and are essentially of one size. Discussion of this point with S. Pugh of Harwell led to the suggestion that this discrepancy is perhaps due to higher Al and Fe contents in British uranium. Their current uranium contains two to three times as much Al and Fe as it did several years ago, and apparently swells considerably less. Pugh proposed that the increased Al and Fe were responsible for the recent UK observations of pore size and distribution. Efforts are being made to gain more information concerning the chemistry and history of British uranium.

The bright field optical micrographs of uranium spheres irradiated at about 600 C to burnups of 0. 046 and 0. 28 respectively are shown in Figure 3. 17. This usual non-irradiated, beta heat treated microstructural appearance is considerably different from the "worked" microstructure that results from low temperature irradiation. Other than cracking, no particular difference in microstructure was observed between the two burnup levels.

A precharacterized, unrestrained specimen of natural uranium, GEH-14-36, irradiated in a NaK environment at temperatures less than 150 C, has been processed for microstructural and density change. After an estimated burnup of 0. 26 a/o during which 40 complete reactor power reductions had occurred, the specimen had assumed a very wrinkled, bumpy external geometry. None of the original grain boundaries or microhardness indents were recognizable, Figure 3. 18. Repolishing and etching of the specimen surface revealed a highly worked and swirled microstructure typical of low temperature irradiated uranium, Figure 3. 19. Electron micrographs of this specimen show pores as large as 1,000 Å in diameter but the pore density is quite low. These metallographic observations are in qualitative agreement with the experimental buoyancy density measurements which indicated about a 2 per cent change in volume. This specimen is currently being repolished for microhardness and physical measurements and post-irradiation annealing at 600 C. After annealing the specimen will again be characterized by physical measurements, hardness, density and metallography.

Pore Size and Distribution. Pore size data pertaining to uranium which has been annealed at various temperatures after restrained irradiation have been tabulated on magnetic tape for computer use and analysis. Pore volume fractions and pore densities are being calculated. In addition, changes in the pore size-frequency curves are being correlated with pertinent variables.

For uranium which has undergone extensive swelling, approximately 25 per cent, buoyancy density measurements and quantitative metallography are both susceptible to considerable error. This error arises because cracks

FIGURE 3. 16

Size Ranges and Preferential Location of Large Pores in an
Unrestrained Uranium Sphere (Specimen 4-2)

The specimen was irradiated to a burnup of 0.28 a/o at
a controlled volume average temperature of 615 C. 4,500 X

(a) Specimen 5-3, 0.046 a/o Burnup

(b) Specimen 4-3, 0.28 a/o Burnup

FIGURE 3. 17

The Microstructure of Uranium Irradiated at Elevated Temperature
250 X

Macrograph, $1\frac{1}{2}$ X
As-Irradiated

Pre-Irradiated

Stereomacrographs
3.8 X

As-Irradiated

FIGURE 3. 18

Microstructural Changes in a Precharacterized Unrestrained Specimen of Uranium Irradiated to a Burnup of ~ 0.26 a/o at a Temperature of Less than 150 C. The Irradiation Included 40 Thermal Cycles.

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250 X

18,000 X

FIGURE 3. 19.

Optical Micrograph and Electron Micrograph of Specimen
Shown in Figure 3. 1 After Repolishing and Etching

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in a specimen cannot be differentiated from large interconnected pores. Accordingly, it is impossible, in many instances, to estimate the volume associated with cracks if indeed they are cracks. The problem of estimating pore void fraction from statistical counting or photometric measurements of large pores in low magnification (250 X) optical micrographs is, therefore, being investigated.

Comparison of In-Reactor and Ex-Reactor Induced Swelling. A formal report, HW-69393, has been submitted for publication. This report describes swelling in uranium rods clad with 0.030-inch thick Zircaloy-2 irradiated in a NaK environment and also the effect of annealing uranium specimens cut from a rod irradiated under conditions of high restraint. If the swelling expressed as $100 (\Delta V/V)/BU$ is compared for the two types of material as a function of temperature of irradiation and temperature of annealing, a surprising correspondence results. (Burnup of all specimens is in the range 0.19 to 0.29 a/o.) The curve in Figure 3.20, which illustrates this similarity in gross swelling, also contains recently obtained results on Zircaloy-2 clad uranium rods irradiated in NaK to a burnup of 0.43 a/o, as well as results obtained with three bare uranium spheres irradiated in NaK to a burnup of 0.28 a/o. The additional data support the conclusion that in a given burnup range ex-reactor annealing of irradiated uranium may indeed provide an estimate of expected in-reactor swelling.

Fission Product Mobility. A knowledge of inert fission gas mobility in uranium is important to the understanding of the basic mechanism of swelling. Several approaches are currently being pursued in an effort to establish this mobility. The first involves studying specimens that have had inert gas introduced into the surface by "glow discharge" or ion bombardment. The gas release characteristics are being determined and the microstructure is being examined for gas pore formation. The second approach utilizes U-U diffusion couples that contain an appreciable quantity of fission product in one-half of the couple, but none, or at least very little, in the second half. Microhardness and metallographic traverses will be made across the U-U interface.

With regard to the latter approach, two 1/2-inch-long by 1/2-inch diameter disks cut from a copper clad (unbonded) extrusion having a core of depleted uranium (0.15 per cent U^{235}) and a 0.030-inch shell of enriched uranium (3 per cent U^{235}) have been irradiated at ambient temperature in the MTR. They were located in a flux of 5×10^{13} nv and received an exposure equivalent to about 0.3 a/o BU in the shell. The changes in pore size distribution and in hardness across the U-U interface will be studied as a function of post-irradiation annealing. If measurable changes occur, reasonable estimates of the fission gas diffusion rate should be possible. This type of experiment eliminates the concern over the diffusion barrier that is unavoidably present at the surface in the "release" type of experiment that has been conducted to date.

Krypton evolution studies were continued on uranium disks that were subjected to "glow" discharge and to cathodic vacuum etching (ion bombardment) under various conditions. Two uranium discs were prepared in a cathodic vacuum etching chamber in which Kr was the etchant. One sample was etched in normal fashion at low pressure (approximately 5 microns) while the other was etched at a much higher pressure (approximately 50 microns) so that redeposition occurred. Upon being heated to above their melting point, both discs exhibited a broad peak in the evolution rate of Kr gas of about 0.07×10^{13} atom/sec in the 350-500 C temperature range and a small burst of 0.2×10^{13} atoms/sec at the melting temperature. No difference was observed between the specimen etched under normal conditions and the other specimen etched under adverse conditions, so that redeposition occurred.

Another disk was "glowed" at ambient temperature for six hours at a Kr pressure of 2 mm of Hg and a current and voltage of 4 ma and 0.8 kv respectively. There were positive indications that gas was being consumed during the glow. The disk was removed from the system cut in half and one piece heated to above its melting point while monitoring the evolved gas with the mass spectrometer. A broad peak in the evolution rate amounting to roughly 0.07×10^{13} atoms/sec was observed at about 500 C and a second peak was observed upon melting. The other half of the disc was polished and examined metallographically. The redeposition layer was very thin and there was no indication of pore formation or any other unusual micro-structural change.

The "glow" discharge system, itself, was heated and the evolved Kr monitored. Two large peaks in the evolution rate were observed, one at 400 C and another at approximately 800 C with peak heights being about 11×10^{13} atom/sec respectively. This data compares favorably with previously observed data and indicates that most of the Kr consumed during glow is trapped on the walls of the chamber. Large quantities of uranium metal are vaporized from the sample during glow; most of which is condensed on the wall of the chamber. It is probable that the condensing metal acts to trap the gas. The surface areas involved here are, of course, grossly different making a realistic comparison of the gas in the system and sample difficult. In view of this large evolution from the system, a hollow uranium cylinder was made and placed on the sample disc in such a manner that the uranium that had previously collected on the walls of the chamber was collected on the walls of the cylinder. The cylinder was "glowed" in a Kr atmosphere at a pressure of 2 mm Hg for about a day. There were again indications that Kr gas was being consumed during the glow, however, as was expected, very little uranium was deposited on the walls of the "glow" chamber. After the glow, the cylinder was sectioned into eight pieces and the Kr evolution characteristics examined by heating the sections in the same manner as that previously applied to the uranium disks. This behavior was quite similar to that of the disks discussed previously, i. e., a peak in the evolution rate in 300-500 C temperature range and a second peak at the

FIGURE 3. 20

Comparison of In-Reactor and Ex-Reactor Induced Swelling

melting point. On a per unit surface area basis, the sections of the cylinder yielded only slightly larger quantities of evolved Kr gas than did the disks. However, a total of the Kr gas evolved from all eight sections would be about the same order of magnitude as that evolved from the glow chamber walls.

It is apparent that larger quantities of evolved Kr gas would lend themselves more readily to mobility studies. To this end, a new facility is being designed that will allow greater amounts of gas to be introduced into the uranium. The "glow discharge" work will be interrupted for several months until this system is designed and constructed.

Restrained Irradiations. In order to gain insight into the influence of Zircaloy-2 cladding on the swelling of uranium, Zircaloy-2 clad uranium fuel rods with selected uranium temperatures, cladding thicknesses, and exposure are being irradiated in NaK-filled, temperature-monitored capsules. Eleven capsules, each containing two fuel rods have been discharged from the MTR and ETR. The last three, GEH 14-94, 95, and 99 were discharged at goal exposures of 0.37, 0.54, and 0.53 a/o respectively. Seven capsules from this series have been opened and the fuel rods photographed, measured for dimensional changes where possible, and specimens cut for density determination and burnup analysis. Calculations from dimensional measurements (per cent $\Delta V/V_i$) and density determinations (per cent $\Delta \rho/\rho_i$) show the following changes on eight of the rods from four capsules. (Table 3.6)

TABLE 3.6

DENSITY AND VOLUME CHANGES IN IRRADIATED FUEL RODS

Capsule	Fuel Rod Cladding Thickness Inches	Per Cent $\Delta \rho/\rho_f$	Per Cent $\Delta V/V_i$	Maximum Volume Average Uranium Temp. C	Uranium* Burnup a/o	R
14-96	0.020	-1.23	1.0	415	0.07	17
14-96	0.030	-1.50	0.7	440	0.07	21
14-97	0.020	-5.4	6.7	515	0.43	12
14-97	0.030	-6.2	5.9	545	0.43	14
14-101	0.020	-0.85	0.46	250	0.3	2.8
14-101	0.020	-0.8	0.50	250	0.3	2.7
14-103	0.020	-0.64	1.0	300	0.21	3.0
14-103	0.020	-1.8	0.9	300	0.21	8.6

ρ_f - Final density

V_i - Initial volume

R - The per cent volume change divided by the per cent burnup using density changes to determine the swelling.

* - Estimated from reactor operations data.

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Capsules 14-96 and 14-97 each contained one rod with 0.020-inch thick cladding, and one rod with 0.030-inch thick cladding. In both capsules, the cladding failed on the 0.020-inch clad rods just below the weld joint of one of the end caps resulting in breakage of the end cap from the fuel rod. The cladding at the opposite end of the 0.020-inch clad rod in 14-97 had several vertical splits, some of them joined by horizontal failures in the weld zone. No cladding failures or other unusual features can be seen on the fuel rods of 14-101 or 14-103 --- all of which were clad with 0.020 in Zircaloy-2. Because of the cladding failures, the density decreases are probably more reliable than the calculated volume changes. Both fuel rods of 14-96 have higher density changes than was anticipated for this temperature and burnup. This may, however, substantiate the density changes observed in the low burnup spheres. Additional observations will be necessary to establish this point. There is also an apparent discrepancy in the density changes of the rods of 14-103. It is felt that both rods should have exhibited $\Delta \rho / \rho_f$ values of about 0.6 per cent. Both of these rods will have their density remeasured. The swelling data from the specimens in 14-97 and 14-101 are in general agreement with the data previously obtained from similar irradiation tests. It is interesting to note that no pronounced difference exists between 0.020-inch clad rods and 0.030-inch clad rods. This corroborates previous suggestions that the cladding offers little in the way of restraint to swelling.

The volume average temperatures reported here are only best estimates. A closer examination of the temperature records is being made to refine this estimate. It should also be pointed out that the reported burnup values are based on reactor operation data. Previous calculations have agreed with the chemical analyses to within 15 per cent, but the values shown here are subject to adjustment when the chemical burnup analyses are obtained.

Of the six fuel rods taken from the other three capsules, 14-98, 14-104, and 14-105, only one (one of the rods in 14-105) survived the irradiation with no apparent cladding damage. Of the remaining five, four have cladding splits and end caps missing, and are so badly distorted that no dimensional measurements can be made on them while the fifth (the second rod from 14-105) shows damage but is measurable. Length and diameter measurements will be made on the two rods from 14-105. Estimated burnups and maximum center uranium temperatures for 14-98, 14-104, and 14-105 are respectively 0.43, 0.56, 0.62 a/o and 600, 700, and 375 C. Samples for burnup analysis and density have been cut from these six rods.

Notch Sensitivity of Zircaloy-2 - K. R. Wheeler

Zircaloy-2 clad uranium test elements have shown an unpredicted tendency to fail locally under stress with little straining in the remainder of the clad. Sensitivity of Zircaloy-2 to surface irregularities such as notches or deep striations introduced during forming or handling might be the cause of this lack of uniform strain. Variations in wall thickness could also contribute to high local deformation under stress. The effect on material strength and ductility of such parameters as shape and depth of a defect, strain rate, structure, grain orientation, and temperature can be determined from tensile data. Material for testing was rolled to 0.020 inch and given a recrystallization anneal. Specimens were then machined with the long axis transverse to the rolling direction. A face was then notched parallel to the rolling direction. After machining the samples were annealed at 600 C for two hours.

The first of the tensile tests were conducted at 280 C using specimens having either a V shaped notch with a 0.005 inch-base radius or a square notch with 0.065-inch notch width. Tests were performed on an Instron tensile machine. The specimen and grips were enclosed by a split type, resistance wound, three zone Marshall furnace. The temperature was monitored and controlled by means of chromel-alumel thermocouples mounted in contact with the specimen. Strain rate was constant at 0.02 in./min cross head motion.

Preliminary analysis concerned the change in uniform elongation resulting from the presence of a defect. Uniform strain was obtained by measurement with a cathetometer of the separation of scribe marks bounding a one-inch gage section located adjacent to the notched region of the reduced section.

In the event that necking within the gage section preceded notch failure, the limit of uniform strain was taken from the high point of the Instron machine load versus cross head motion curve. Strain was then calculated from an effective gage length between grips. Theoretical values of gage section strain of a notched sample were calculated from the Instron curve of an unnotched control sample tested at the same strain rate and temperature. From the control curve, the value of σ_{ult} (nominal) was multiplied by the factor $(1-f)$ where f is the notch fraction $\left(\frac{\text{notch depth}}{\text{specimen thickness}} \right)$. The stress so obtained is the σ_t value in the gage section of the notched sample when the stress in the reduced section at the notch is σ_{ult} . The theoretical strain is then obtained directly from the control curve at the calculated σ_t values.

The curves of Figure 3. 21 illustrate the variation of uniform elongation with notch fraction, f . The per cent uniform elongation decreases with increasing notch depth as was expected, but the comparison of actual and theoretical curves shows that the decrease is less than theory predicts. This

notch strengthening effect continues with a change in notch shape. A comparison of the V notch and square notch curves of Figure 3.21 shows a definite difference between the strengthening effects of the two notch types. The square notch values more nearly approach the theoretical curve, but strengthening persists because of the stress pattern at the sharp corners. Additional testing will examine the effect of variations in V notch root radius, other strain rates, and mild notches.

Previous work with irradiated Zircaloy-2 has shown an almost complete loss of uniform strain at fairly low exposures, but little is known concerning the influence of irradiation on notched properties. Various types of specimens are being irradiated and tensile tested in order to evaluate the combined effects of irradiation and surface defects. Notched flat tensile specimens will be employed as will cladding material isolated from coextruded rods by chemically removing the uranium core.

FIGURE 3.21

Uniform Strain Variation with Fractional Notch Depth

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4. 1

METALLIC FUEL DEVELOPMENT

NEW FUELS DEVELOPMENT

The Effects of Cladding Thickness Uniformity on Zircaloy-2 Clad Coextruded Rods - J. W. Weber

Failures of the cladding of Zircaloy-2 clad uranium rods and tubes as a result of localized clad thinning have occurred in NaK and high temperature recirculating water loop irradiations. Nineteen Zircaloy-2 clad uranium rods irradiated in NaK capsules in a Hanford reactor have been examined on the basis of the possible factors influencing the localized cladding instability. Measurements of the cladding thickness were made from photomicrographs of cross-sections from three rods having nominal 30-mil and four rods having nominal 20-mil thick cladding, and a comparison of the thickness variations and the susceptibility to clad thinning was made. None of these rods had cladding splits, but three of the rods having nominal 20-mil thick cladding did show evidence of localized clad thinning, which can be seen as longitudinal striations on the external surface. Each of these obvious striations are located where the post-irradiation cladding thickness is 75-85 per cent of the nominal thickness over a circumferential gage length of 0.035-0.070 inch. However, other points with thickness variations of equally as great percentage changes over the same circumferential gage lengths are found in the 20-mil clad rods, but no visible striations occurred at these locations. Regions of similar cladding thickness changes are found where the circumferential gage lengths are 0.150 inch or larger, but there are no visible striations. Also, 75-85 per cent thickness variations over 0.035-0.070 inch circumferential gage lengths are found in the 0.030-inch thick clad rods, and no localized clad straining is evident. Thinner cladding at striations has been observed on rods in which failure of the cladding did occur but these were not considered in this comparison.

Inasmuch as the cross sections studied were taken at random and not necessarily at the point of initiation of the localized straining, other dimensional conditions may certainly exist on either side of the cross-section taken. For this reason, no positive conclusion can be made about the effect of preirradiation cladding thickness dimensions on cladding stability. However, it does appear that both thickness and variability are factor influencing the susceptibility of Zircaloy-2 cladding to localized clad instability.

To further study the effects of cladding thickness variations on the susceptibility to failure, a new series of NaK capsule irradiations of Zircaloy-2 clad fuel rods is planned. Eighteen capsules have been assembled and are ready for irradiation. Each capsule contains three fuel rods each with a different depth of 0.125-inch radius striations machined longitudinally in the cladding surface. The cladding thickness has been characterized as carefully as possible by autoradiographing and measurements.

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4. 2

HW-70547

Fuel Element Deformation Studies - K. R. Merckx

Previous reports of this series have described models for tubular fuel elements. These models accounted for the resistance of fuel material to shearing and the cyclic straining of the cladding. A formal report⁽¹⁾ describes the methods of analysis and the results of numerical calculations. Since the deformations of the fuel elements depended upon the resistance of Zircaloy cladding, the validity of the proposed material model under incremental strains applied during strain cycling is being checked. A testing frame, extensiometer, and furnace are assembled and an initial test has been run. The specimen buckled during the compressive part of the strain cycle. A new specimen geometry and increased bending rigidity of the grips are being tried to stop the buckling problem.

Corrugated Cylindrical Fuel Element - K. R. Merckx

Volume expansions of corrugated cylindrical fuel elements cause bending strains without uniform elongation of the cladding. An elastic analysis for uniform pressure loadings has been coded for any symmetric shape of corrugation and variation in cladding thickness. A program for the systematic investigation of corrugations formed by two tangential arcs has been coded but has not been debugged. Parametric study of families of constant arc length are planned.

Modified Tubular Fuel Geometry - R. G. Wheeler

Localized tensile necking and splitting of the Zircaloy-2 clad on coextruded uranium cored fuel elements may represent an ultimate exposure limit. One possible way to increase the total energy capability of the fuel element is to corrugate the outside clad so its perimeter is greater than the circumference of a circle circumscribing the equivalent cross sectional area. Thus, swelling of the core will be accommodated by bending strains in the clad rather than by relatively large clad tensile strains.

Two extrusions of irradiation quality with a cross-sectional shape illustrated in Figure 4.1 were made by Fuels Fabrication Operation. Three 18-inch long fuel elements are being made from these extrusions for irradiation testing to determine if this shape can accommodate more uranium swelling without clad failure than a circular section.

Increased Strain Rates Due to Large Temperature Cycles - K. R. Merckx

Initial tests on constant load tests during temperature cycling (150-450 C) of uranium have been run. The results of these tests have been summarized.⁽²⁾ Application of the test data required separation of the straining occurring during increasing and decreasing portions of the temperature cycle. The equipment used for the initial tests was inadequate and modifications became necessary. Modified specimen capsules, and extensiometer are being fabricated.

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4. 3

HW-70557

FIGURE 4. 1

Cross Section Through Zircaloy-2 Clad Coextruded Fuel Element

The outer clad is shaped so that swelling of the core will produce only compressive or bending strains in the clad instead of tensile strain.

Photograph Unclassified

ALICE RICHARD WASH

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Fuel Element Support Development - R. G. Wheeler

An improved fuel element centering self support for NPR inner tube fuel elements was developed. The advantages of the new, folded, support over the old, arched, support are described in Reference (3). The new support is stronger as illustrated in Figure 4. 2. It has a greater elastic range, and it does not decrease in height as the fuel element to which it is attached expands thermally. Drop tests simulating discharge of fuel assemblies from the NPR have shown it is possible to hold the outer and inner tube together during discharge with a clip spot-welded to the inner surface of the outer fuel tube. This clip engages with one of the folded type inner tube self supports to prevent the inner tube from sliding out of the outer tube.

FABRICATION DEVELOPMENT

Self-Brazed Closure for Zircaloy-Clad Fuel Elements - E. A. Smith

Procedures described in previous reports have continued to yield consistently sound bonds on specimens tested as the as-pressed condition. However, the next step in the fabrication process, viz., the salt bath heat treatment applied to impart a stable structure to the uranium, tends to crack the bond at the uranium-copper alloy interface between the Zircaloy cap and the core material. This is apparently due to the brittleness of the uranium-copper alloy formed at the contact surface between the uranium core and the copper plating on the Zircaloy cap, coupled with the considerable differential expansion between the uranium and Zircaloy components. Past experience with hot pressed fuel elements indicates that a thin nickel plate adjacent to the uranium forms a tough, ductile bond in a relatively short pressing time at temperatures in the neighborhood of 600 C. Thus it seems probable that a layer of nickel on the uranium side of the bonding layer might reduce this cracking tendency. But since nickel bonds less readily than copper with Zircaloy, a copper layer next to the cap is indicated. The obvious answer is to plate the Zircaloy cap with a coating of copper up to 0.0005-inch thick, then overlay the copper with a thin layer of nickel. Procedures for applying this double plate have been developed, and test specimens to determine the feasibility of the concept are being prepared for trial.

A 60-degree V-groove interface between cap and core continues to appear favorable, and facilities to speed up the chem-milling of this contour have been developed. Preliminary trials indicate satisfactory functioning of this equipment.

A review of the results of experiments conducted during the past six months leads to the belief that consistent quality and durability for this type of closure, which if perfected offers many advantages over the currently favored Zircaloy-2 with 5 per cent Be braze, can best be attained by simultaneously pressing the welded assembly and applying heat in the zone to be bonded. For this purpose, resistance heated dies are unsatisfactory; the heat is generated in the wrong place. Self-heating, accomplished by passing

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a heavy current longitudinally through the closure has shown promise, but providing electrical insulation in the right places without interfering with the application of pressure poses difficult problems. A new spot-welding machine has been procured to further exploit this approach, and may permit working out a satisfactory process.

A procedure more in keeping with the apparent requirements involves the use of a high strength ceramic die surrounded by an induction coil, suitably reinforced to withstand the necessary pressures. Heretofore, such a die has appeared to be beyond hope, but recent developments point to possible realization, and materials are being procured to fabricate one for trial purposes.

Projection Welded Resistance Brazed Closure - P. A. Ard and W. I. Steinkamp

The projection welded brazed closure is being evaluated as a possible alternate method of closing coextruded Zircaloy-2 clad uranium fuel elements and consists essentially of projection welding a Zircaloy-2 cap to the element cladding followed by a fast resistance heating of the cap to braze the cap to the uranium surface. Attempts have been made to improve the strength of the closure by machining the face of the cap which contacts the uranium into a 60 degree wedge. The wedge shape has the advantage that the projection welds and the braze interface are not in the same plane. This modification was tried on KER inner tubes (1.050-inch OD by 0.500-inch ID) with encouraging results. A cross section of the closure can be seen in Figure 4.3.

Approximately 0.025 inch of uranium was acid milled from the end of the element to provide uranium free cladding projections and a V-groove is machined in the uranium to match the wedge on the closure cap. The cap wedge was copper immersion plated to aid in bonding the Zircaloy-2 cap to the uranium.

Approximately sixty per cent of the cap area was bonded to the uranium using a brazing force on the cap of 2,200 pounds. Approximately eighty per cent of the area was bonded using a brazing force of 2,800 pounds. The unbonded areas in both cases were on the inside of the wedge up near the ID projection weld. Results indicate that a combination of a greater brazing force and closer fit between the uranium and the cap on the inside of the wedge may eliminate the unbonded area.

The projection welds were fired between the clad and the cap with the OD projections misaligned 0.007 inch on a 0.020-inch thick cladding wall. Results indicate that this amount of misalignment may not be detrimental to the projection weld. A cross section of the closure with misaligned OD projections is shown in Figure 4.4 with a micro of the projection in Figure 4.5. The closure was not taken through the cap to uranium brazing cycle in the above tests.

Tests to date on tubular elements have included autoclaving a 4-inch long 1.050-inch OD by 0.500-inch ID element, closed on both ends with the wedge shaped cap, for 57 hours in 400 C, 1500 psi steam. The results were satisfactory with no indications of corrosion noted in the closure areas.

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FIGURE 4. 2
Height of Two Types of Supports Before and After Loading

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FIGURE 4.3

Cross Section of Projection Welded Resistance Brazed Closure
on a 1.050-Inch OD by 0.500-Inch ID Zircaloy-2 Clad Uranium
Tube Using a Wedge Shaped Closure Cap. 3 X

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Misaligned
Projection Weld

2 X

FIGURE 4.4

Cross Section of Projection Welded Brazed Closure
Showing Misaligned Projection Weld on Tube OD

Zircaloy-2
Cap

Projection
Weld Zone

Uranium

FIGURE 4.5

Micro Section of Misaligned Projection Weld Shown in Cross
Section of Projection Welded Brazed Closure in Figure 4.2 50 X

Photographs Unclassified

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Closure Welding - R. F. Klein

End closures for the new tubular fuel elements with "fluted" external surfaces were accomplished by using the electron beam welding apparatus. End caps, cut to the shape of the fuel element, were welded in place using a procedure requiring two operators. This was necessary on the present equipment because three different adjustments had to be made at the same time. One operator adjusted the high voltage control and the other manipulated the electron gun and turntable speed. Some welding difficulty was experienced because of loose end caps. Another difficulty occurred when the electron beam spot approached the points of the fluted clad. Since a greater amount of heat is concentrated at these points than on the larger radius of the concave parts of the clad surface, these points had a tendency to melt considerably faster and, at times, nearly burn out. Good welds at these points were made when the high voltage was lowered as the welding spot approached and went around the tip.

Welding of this shape element by the electron beam process could be programmed to be done automatically with little trouble.

A method of attaching a 1/4-inch diameter Zircaloy-2 stud, with a 1/8-inch diameter welding area, to the cladding of certain fuel elements was desired to test the cladding-uranium bond strength. The requirements of the attaching method were that it did not affect the Zircaloy-uranium bond and that the study would hold to the cladding as the Zircaloy-uranium bond was pulled apart. After the stud had been attached, the Zircaloy cladding immediately around the stud was to be cut leaving only the Zircaloy-uranium bond holding the stud and the cut cladding to the uranium. The attachment was achieved through the use of a stud welder. The weld produces a bond which holds together when pulled in a tensile testing device. The Zircaloy-uranium bond is pulled apart producing a hole in the Zircaloy cladding. When testing the welds, no cut was made in the cladding and an average of 1200 pounds was needed to pull the button.

The welding apparatus for investigating the Metal Inert Gas welding process as a method for making the final closure on the NPR fuel elements was modified so the entire welding cycle, exclusive of positioning the fuel element and adjusting the turntable speed, could be done automatically. The weld, on the outside diameter of the NPR inner element, from the striking of the arc, takes slightly less than three seconds to complete.

The two principle problems to deal with in the above process for this investigation are braze contamination of the weld, and weld profile. Until recently, because of the cap step-cut design, the more heat applied during welding, i. e. , the higher the voltage used in the welding process, the greater the amount of braze in the weld filler metal. Since this was the case, a good weld profile had to be obtained while using the low voltages required for the braze free weld. It was difficult to get a braze-free weld and a good profile at the same time because of this low voltage condition. The arc between the filler wire and the metal to be welded has a tendency

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to jump from the filler wire to the nearest metal, whether it be to the side or directly below the wire. This sometimes leads to the weld being placed on the top of the step or on the extreme edge of the step and not in the step as it should be. The positioning of the filler wire electrode before the weld, therefore, is extremely critical. A step cut that would allow a higher voltage to be used, would eliminate this problem and should deliver the desired weld profile. In an attempt to develop this type step, a groove was cut in the step immediately over the braze, thus actually cutting out a ring of braze a few thousandths of an inch deep. The Zircaloy-2 cladding was then bent over the groove. This formed a pocket from which the braze cannot escape to contaminate the weld metal. The initial attempts to lay a filler metal bead in this step have resulted in a weld with the braze very well contained but with a poor weld profile. Steps are being taken to correct this.

It should also be pointed out that because of the pocket formed by bending the cladding over the groove, the weld may contain a void. This was found to be the case in several specimens examined.

Welding Fuel Element Supports - W. I. Steinkamp

Although it has been demonstrated that Zircaloy supports could be welded to NPR fuel elements with adequate strength, the degree of reproducibility and the degree of weld control had not been established. A run of 1000 welds was made attaching 0.040-inch supports to 0.040-inch clad NPR inner tubes. These were shear tested with the exception of thirty-one, which were sectioned and examined for weld quality. Figure 4.6 shows a typical weld. The welding of these supports was interrupted twice to perform other jobs on the welder and then continued with no machine testing and no set up other than returning the controls to their previous settings. Shear tests showed 86.2 per cent of these welds in the 1000-1300 lb range.

All welds were made with a spring mounted low inertia holder operating at 100 lbs on the 100 KVA resistance welder. Secondary voltage of the machine was 8.8 volts.

Another much shorter run was made on NPR outer tubes attaching 0.040-inch supports to 0.024-inch clad. In this case the shear strengths fall between 1000-1200 lbs for the 40 samples sheared. A metallographic section is shown in Figure 4.7. It showed no effect on the uranium beneath the weld although the heat affected zone goes through almost seventy per cent of the clad. No curvature and no projections were used on the support tabs for the outer tube welds.

As can be seen from the shear tests and the micro sections, it is possible to securely attach supports to either the inner or outer fuel tube without disturbing the uranium.

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FIGURE 4. 6

Support Welded to NPR Inner Tube

50 X

FIGURE 4. 7

Support Welded to NPR Outer Tube

50 X

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Fuel Deformation Studies - R. G. Wheeler

The coordinates of the outside surface of tubular fuel elements were measured in three dimensions with a lathe modified by fitting it with transducers on the tool carriage and attaching an x-y recorder. Measurements obtained with this lathe were compared with plane of warp and maximum displacement as obtained by the dial-gage method.

All of eight fuel elements examined contained some corkscrew curvature and non-uniform curvature not detected by the dial-gage method. However, the magnitude and direction of warp as obtained by the dial-gage method was in good agreement with that obtained from recorded profiles of the fuel element shape. Some of the fuel elements contain local warp or kinks. If this local warp is near the center of a fuel element, total warp is large. If the local warp is near an end, total warp is small.

Diameter profiles of the fuel elements from extrusion 89 contained characteristic (three to five mil) positive and negative deviations which can be traced the full length of the extrusion. These deviations can be used to locate angular errors in the placement of the index mark at the top of the extrusion.

An inch or so of uranium at each end of the fuel elements was heated into the uranium gamma phase during end cap brazing. A diameter profile of these ends has the same general shape as the remainder of the fuel, but the end diameter is three to six mils smaller than the remainder of the fuel element even after beta heat treatment.

Deviations in the inside clad wall thickness follow closely the deviations in uranium wall thickness. The outside clad, however, has a negative correlation with uranium wall thickness. A tendency for the fuel to bow convex in the direction of greatest uranium wall thickness was observed.

Measurement of the residual longitudinal stress in the jacket from a portion of fuel element 89-147 shows the outside jacket can exert a significant bending moment. In the one specimen examined, the observed 0.015-inch warp would increase to 0.023 mils on removal of the outside jacket. Improved equipment, capable of measuring both the inside and outside surface of NPR fuel elements, will be of great aid in determining the cause of bowing and in interpreting irradiation behavior of these elements.

Results of this work are reported in References (4) and (5).

Fuel Straightening - J. P. Pilger

Experiments on 24-inch long, N inner fuel elements indicate that hot straightening in a three-roll straightener (Hanford design) produces an element that will meet N inner tube assembly straightness specifications. Six HLO extrusions were cut into 24 elements, each element 24-inches long. The elements were randomly divided into three groups of eight elements each. All elements were subjected to the same beta heat treatment of 730 C salt - 7 min, air delay - 20 sec, 300 C salt - 5 min, 20 C water - till cool. Straightening treatment varied as follows:

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Group A

1. Vacuum Anneal at
600 C - 4 hours
2. Hot Roll Straighten
3. Autoclave 72 hours
4. Autoclave 72 hours

Group B

1. Hot Roll Straighten
2. Vacuum Anneal at
600 C - 4 hours
3. Autoclave 72 hours
4. Autoclave 72 hours

Group C

1. Hot Roll Straighten
2. Autoclave 72 hours
3. Autoclave 72 hours

Each element was heated to 575 C in an argon atmosphere. When temperature was reached (10-11 minutes) the element was placed in the straightener. The straightener roll temperature ranged from 20 to 45 C. The element was cooled to roll temperature (8 to 11 minutes) while revolving about its longitudinal axis. Table 4. 1 indicates the double throw warp, and plane of warp measured between each step in the fuel processing. Figure 4. 8 illustrates a schematic diagram of the three-roll straightener. All straightening on the above elements was done with 2800 pounds force available on the movable roll. Groups A and C were straightened by seating the movable roll against positive stops. An element would then experience the straightening force only if it were warped. Adjustment of this stop is critical. Group B was straightened by allowing the full 2800 pounds to act on the fuel element continuously during straightening. The data from Group B is included to illustrate the improvement in straightness obtained by exerting full force on the element continuously during rolling. No attempt has been made to date to show correlations between warp behavior and straightening variables. The data in Table 4. 1 shows that it is possible to straighten fuel that is warped more than 0.030 inch (double throw), autoclave it, and assemble an element which will be within assembly straightness specifications.

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FIGURE 4.8

Schematic of Three-Roll Straightener

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TABLE 4. 1

ABBREVIATED SUMMARY OF STRAIGHTENING RESULTS

(All warp is recorded as double-throw warp.)

Tube No.	As Extruded Warp/Plane	Beta Heat Tr. Warp/Plane	Hot Straight Warp/Plane	Stress Relief Warp/Plane	1st Autoclave Warp/Plane	2nd Autoclave Warp/Plane
Group A (24 inches long)						
1-35	0.050/270°	0.061/90°	0.010/180°	0.066/90°	0.013/235°	0.022/235°
2-35	0.045/240°	0.062/245°	0.005/300°	0.057/240°	Clad defect - no autocl.	
3-11	0.055/185°	0.053/190°	0.014/265°	0.036/180°	0.012/265°	0.010/265°
3-59	0.012/180°	0.025/180°	0.004/150°	0.025/180°	0.008/230°	0.009/230°
4-82	0.047/100°	0.033/100°	0.003/70°	0.034/100°	0.005/100°	0.005/40°
5-11	0.056/190°	0.110/260°	0.009/290°	0.095/255°	0.020/310°	0.020/310°
5-35	0.043/270°	0.092/275°	0.008/220°	0.096/275°	0.019/275°	0.022/275°
6-59	0.015/270°	0.011/270°	0.005/280°	0.013/270°	0.006/290°	0.006/320°

Group B (24 inches long)

1-59	0.030/170°	0.048/175°	0.008/175°
2-59	0.022/190°	0.016/195°	0.003/60°
3-35	0.055/185°	0.053/190°	0.002/180°
4-34	0.031/198°	0.028/230°	0.003/290°
4-58	0.010/50°	0.027/265°	0.006/305°
5-83	0.021/90°	0.040/110°	0.005/50°
6-11	0.111/270°	0.093/270°	0.010/240°
6-83	0.015/270°	0.011/270°	0.007/255°

Data not yet available

Group C (24 inches long)

1-11	0.080/300°	0.156/90°	0.018/225°	Not applicable	0.040/240°	0.040/240°
1-83	0.024/90°	0.031/170°	0.008/320°		0.010/320°	0.009/10°
2-11	0.032/90°	0.057/100°	0.015/260°		0.022/260°	0.022/260°
2-83	0.019/170°	0.013/150°	0.004/200°		0.004/200°	0.002/40°
3-83	0.040/270°	0.020/270°	0.008/330°		0.014/330°	0.012/330°
4-10	0.115/195°	0.087/270°	0.012/130°		0.018/165°	0.020/230°
5-59	0.014/190°	0.029/260°	0.007/250°		0.010/250°	0.011/330°
6-35	0.022/170°	0.034/165°	0.006/230°		0.008/230°	0.008/230°

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Heat Treatment Studies - R. N. Johnson

Two NPR inner tube extrusions have been made for heat treatment and warp studies. Both extrusions are from the same original 11-inch billet; one was extruded with Zircaloy-2 cladding and the other was without cladding. Twelve 24-inch long elements were cut from each extrusion and all were heat treated at 730 C for seven minutes followed by an 18-second air delay and quench into 300 C nitrate salt bath. Average as-extruded double throw warp was 18 mils for the unclad tube and 11 mils for the Zircaloy-2 clad tube. Average post-heat-treatment warp was 38 mils for the unclad tube and 27 mils for the clad tube, indicating that the Zircaloy-2 cladding does not contribute to the warp and may be effective in restraining the warp to a certain extent. All orientation of the elements has been kept with the top of the die as the reference point. Wall thickness variation of both extrusions, shift of the coextrusion, the plane of warp of both extrusions before and after heat-treatment, and the vector direction of the change in warp upon heat treatment were related to this reference point. Ten out of the twelve Zircaloy-clad fuel elements showed a correlation between the lead shift and the pre-heat-treatment warp while eight of the twelve showed a correlation between the lead shift and the post-heat-treatment warp. These were the best correlations found. No other significant correlations were found in either the Zircaloy-clad tube or the unclad tube. A possibility exists that the warp may be due to differences in preferred orientation between two sides of a tube. As-extruded samples of the tubes are being studied by X-ray diffraction to determine if these differences exist. The Zircaloy-clad tubes are also undergoing a second beta-heat-treatment to determine if the amount of warp or plane of warp will change and if any better correlations will begin to present themselves. unbonded end closures have been welded on the Zircaloy-clad fuel elements and hot straightening and autoclaving tests will be run after the second heat-treatment.

Difficulty has been encountered in obtaining reproducible X-ray results on beta-treated uranium. It has been established that the difficulty is due to the large grain size of the beta-treated material and the lack of a statistical number of grains to be scanned by the X-ray beam. The minimum surface area of beta-treated material needed for reproducible X-ray results is being determined. A recent study of a beta-treated rod showed that reproducible data was not obtainable from the surface area presented by a single cross section. The rod was then alpha-recrystallized at 640 C giving a decrease in grain size as shown by metallography. When again studied by X-ray diffraction, the rod now gave excellent and reproducible results for the same surface area as was scanned previously. The available literature indicates that the major texture of uranium is unaffected by recrystallization; recrystallization is being studied as a possible means of obtaining reproducible and significant X-ray results without having to scan excessively large areas of uranium samples.

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Zirconium Base Alloy Melting and Fabrication - R. G. Nelson

The quarterly report for third quarter 1960⁽⁶⁾ describes the electrode preparation and primary melting of a series of thirteen zirconium base alloys. Melting and fabrication to 0.040-inch thick cold rolled sheet has been completed on all alloys (Figure 4. 9).

Forging of the primary ingots into electrodes for re-melting proved to be unsuccessful because of the poor first melt ingot quality and excessive oxygen pickup during forging. Best results were obtained by quartering the first melt ingots longitudinally and welding these together end to end to form the electrode for the second melt.

Five ingots were made from forged primary ingots. Oxygen values on the finished sheet ranged from 600 to 1800 ppm. The remaining eight ingots were melted from quartered primary ingots with oxygen values of 700 to 1000 ppm. Suppliers' analysis of the zirconium sponge was 855 ppm oxygen. Nitrogen ranged from 6-40 ppm with 17 ppm in the starting zirconium.

Final melting of quartered sections and forged bars was performed into the 2.89-inch diameter crucible using 2000 amps and 30-35 volts. Resulting ingots were sound with good side wall condition; 1/32 to 1/16 inch of metal removed from the surface gave a 100 per cent clean up to most of the ingots.

Forging was done on the 700-ton press from a portable salt pot set at 880 C. Slabs 3/4-inch thick by 3-inches wide by length (Figure 4. 10) were draw forged in 1/4-inch reductions. The slabs were vapor blasted, pickled, cropped and surface irregularities ground out.

Hot rolling was done from an electric fired air atmosphere muffle furnace set at 750 C. One pass per reheat was made using 30 mils per pass to a thickness of 300 mils, followed by a 10 per cent reduction per pass. All sheets were annealed at 750 C after the final pass in the same furnace.

The hot rolled and annealed sheets were vapor blasted, pickled, and surface irregularities spot ground.

Cold rolling was done in 10 per cent per pass reductions from 0.050 inch to 0.040 inch giving a total of 20 per cent cold reduction.

Zircaloy-2 Free Sink Tube Tests for the Structural Materials Development Operation of MDO - F. B. Quinlan

This investigation was conducted to establish an optimum set of conditions for the free sink swaging of Zircaloy-2 tubing. The variables to be optimized were the throw of the dies and the rate at which the material is fed into the dies.

The salient features of the swaging dies, as shown in Drawing H-3-9694, are the 7-8 degree entrance angle, the 4 per cent side relief, and a deep throat that allows for a 25 per cent diametral reduction in one pass.

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Large diametral reductions of thin-walled tubing constitute a very severe test of the tube's physical and metallurgical properties. Swage reductions of the order of 25 per cent should be made under reasonably controlled conditions, otherwise failure of the tube is bound to occur regardless of its properties. The conditions as shown in the following list are of paramount importance:

1. Be sure that the work is held in good alignment with the axis of the swage.
2. A rigidly supported, non-galling guide bushing (maple, oak, textolite, etc.) should be mounted on the door of the swage, especially when hand feeding is used.
3. Galling and rippling of the tube surface may also occur if the tube is allowed to whip as it emerges from the back end of the swage dies.
4. Feed rates as high as 30 inches per minute may be employed if the die throw (Figure 4. 11) does not exceed 10 per cent of the tube diameter. Hand feeding tends to exceed this rate.
5. A die throw of five per cent has been found to give the most consistently good results.
6. A nine degree blow duration (Figure 4. 12, a and b) gives excellent results. Larger blow durations, up to 25 degrees, may be employed but the friction drag on the work piece must be reduced accordingly.
7. Excessive friction drag on the tube will cause spiralling (Figure 4. 13, a, b, c) of the surface. When mechanical feeding is employed, the grips should be lubricated to prevent excessive drag.

In swaging a tube to a smaller diameter there is usually a certain amount of elastic recovery. The degree to which this recovery takes place is dependent upon the work hardening properties of the metal plus the amount of cold work that was in the stock prior to swaging. Within certain limits there is also a size effect; larger tubes tending to show a slight increase in the per cent of spring back. For example, 9/16-inch diameter tubes swaged with 7/16-inch dies showed a recovery of 1.5 per cent and 3/4-inch diameter tubes swaged with 9/16-inch diameter dies showed a recovery of 2.2 per cent.

The aforementioned seven points apply to a four-die swaging operation only. When a two-die swage is used, the side relief should be increased to five to six per cent and the blow duration time kept to a minimum. Furthermore, in a two-die swage there is a much greater tendency for fatigue cracks to form in the tube wall. This tendency to form longitudinal fatigue cracks is caused by the greater degree of side relief. But if the side relief is dropped much below five per cent, galling of the tube and dies occurs along the parting edges of the dies.

With the above recommendations, Zircaloy-2 free sink tube tests may be made with the assurance that a tube of good quality will take a 25 per cent reduction in diameter, in one pass, and still have a smooth inner and outer surface.

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FIGURE 4.9
0.040 Inch Cold-Rolled Sheets of 13 Zirconium Base Alloys

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FIGURE 4. 10

Typical Draw Forged Slabs of Zirconium Base Alloys

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FIGURE 4.11

The Radial Displacement of the Die Block and Hammer Block is Referred to as the Throw

FIGURE 12a

FIGURE 12b

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FIGURE 4. 13 a
Spiralling Due to Excessive Drag

FIGURE 4. 13 b

Spiral Failure Due to Excessive Drag and Too Great a Feed Rate

FIGURE 4. 13 c
Example of Properly Swaged Tube

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FUEL EVALUATION

Single-Tube Fuel Elements - J. W. Goffard

The single component tubular element, designed and fabricated for testing in the KER loops, was described in the previous quarterly. ⁽⁷⁾

Three charges of four elements each have been successfully irradiated to 1200, 2000, and 3600 MWD/T. The 1200 exposure elements looked good after discharge, except for a slight "sooty" appearance on the clad. The 2000 exposure elements also looked good after discharge, except for the lack of the usual black luster of loop tested Zircaloy-2 material and some white areas on the braze-weld closures. The surfaces and closures of the 3600 exposure elements were black, but the inner bores of the two downstream elements were visably distorted.

The irradiation induced volume increase of the elements was measured in the reactor basin following discharge. This was determined by pre- and post-irradiation weighing of each element in the reactor basin. The final weighings were repeated at least one month after discharge to allow for gamma heating decay of the fuel and its unknown effect on buoyancy of the fuel.

There was no significant difference in the data obtained from the two weighings. Swelling obtained from weighing the elements is tabulated below:

<u>Exposure</u>	<u>Tube Position</u>	<u>Vol. Mean Fuel Temp.</u>	<u>Element Swelling</u>	<u>Fuel Swelling</u>
1200 MWD/T	Upstream to Downstream	430 C	0.45%	0.5%
		450	1.1	1.3
		460	1.1	1.3
		465	1.1	1.3
2000 MWD/T	Upstream to Downstream	435	2.2	2.6
		450	2.5	3.0
		465	2.6	3.1
		470	2.5	3.0
3600 MWD/T	Upstream to Downstream	405	2.4	2.8
		425	3.5	4.2
		435	4.6	5.4
		440	4.4	5.3

An empirical expression for the fuel swelling observed in the above elements has been derived which relates the swelling to the 1.7 power of the burnup and the temperature (T-340 C) in the range 400-470 C.

Radiometallurgical examination of the single tube fuel element is in progress. Hot extraction of hydrogen from a clad sample taken from the 1200 MWD/T charge (sooty appearance) indicated 70 ppm hydrogen

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a quantity that is about twice the expected. Pre- and post-irradiation densities of a fuel sample from the downstream element of the 1200 MWD/T charge revealed a 1.35 per cent increase in volume which compares very well with 1.3 per cent as determined by fuel element displacement measurements in the basin.

Sections of the closures of the 2000 MWD/T charge have been made and some fine radial cracks across the uranium fuel in the braze closure heat-affected zone have been found. The cracks terminate at the clad. No deficiencies have been noted in the Be-Zr eutectic braze closures or in the coextruded bond in the heat-affected zone. A cross-section of a downstream element of the 2000 exposure charge revealed some distortion of the inner tube bore. A more advanced stage of inner clad buckling was noted in the basin on the two downstream elements of the 3600 exposure charge. Four NPR inner fuel tubes with 0.040-inch outer clad and 0.025-inch inner clad were discharged from a KER loop at 1000 MWD/T. Swelling determinations made by displacement measurements in the reactor basin indicate an average of 0.4 per cent swelling.

Fuel Component Development - R. G. Wheeler

Steel fuel-centering supports were used on coextruded Zircaloy-2 clad single-tube fuel elements irradiated to 1200, 2000 and 3600 MWD/T. On discharge from the high-temperature, pH 10 water loops, the supports were undamaged and coated with a black, adherent oxide. All-steel supports attached to the clad with Zircaloy-2 rivets have been superceded for use on NPR outer fuel tubes by Zircaloy-2 supports covered with a crimped on steel shoe. The irradiation tests served to demonstrate the efficacy of steel or steel clad supports for high-temperature water service.

Tubular Fuel Elements - G. T. Geering

Post-irradiation examination of four KER tube-in-tube fuel elements exposed to 3200 MWD/T has been completed. The elements, each consisting of an inner and an outer fuel tube, were irradiated in a high temperature, water-cooled loop. The irradiation ended when a clad split occurred in one of the inner tubes. The water entering through the clad split caused extensive internal fragmentation and corrosion of the unalloyed uranium fuel. Metallographic examination showed that the Zircaloy-2 clad had failed due to localized clad thinning resulting from the expansion of the irradiated uranium. This type of failure had been seen previously in Zircaloy-2 clad uranium rods exposed to 2500 MWD/T. (8, 9) The corrosion damage was not serious enough to block coolant flow or affect the outer tube of the fuel element.

In addition to the failed element, one uranium core and two U-2 w/o Zr core elements were irradiated. The failed element and one of the U-2 w/o Zr elements were examined in detail. The dimensional changes of these two elements were reported in preceding quarterly reports, (6, 7, 10) and are summarized in Table 4.2. There was a marked difference in the

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surface appearance of the elements after testing. The uranium elements had a roughened appearance, but the U-2 w/o Zr elements looked unchanged. Some white oxide corrosion product formed on the support of both elements. Neither element had a measurable change in length or warp. Metallographic sections were taken from the midlength of both tubes of each element. In addition, extensive metallography was done on the damaged area of the uranium element. Cracks were present in all the sections which contained uranium. However, no cracks were found in the U-2 w/o Zr. No grain structure could be distinguished in the microstructure of the uranium or the U-2 w/o Zr. The U-2 w/o Zr contained a dispersed second phase before irradiation. After irradiation, no second phase appeared in optical or electron microscope examination.

TABLE 4. 2
VOLUME CHANGE AND CORE TEMPERATURE DATA FOR
KER TUBES EXPOSED TO 3200 MWD/T AVERAGE

		B. U.	Per Cent $\frac{\Delta \rho}{\rho}$	Per Cent $\frac{\Delta V}{V}$	Max. Core Temp. °C	Nominal Tube Dimensions Inches
A-1	Uranium	0.42	-5.8	5.0	410	18.0 Length 1.050 OD
B-1	U-2 w/o Zr	0.38	-4.7	3.5	430	0.500 ID 0.020 Clad Thickness
A-2	Uranium	0.54	-3.5 ± 0.14	2.7	330	18.0 Length 1.820 OD
B-2	U-2 w/o Zr	0.45	-6.1 ± 0.14	4.9	350	1.450 ID 0.020 Clad Thickness

The volume increase of the fuel was determined for each tube and correlated with maximum core temperatures and burnups. Volume increase was determined by two methods: (1) density measurements and (2) dimensional measurements. The density specimens and dimensional measurement were taken at the midlength of the tubes. In Table 4. 2, the results of both methods are compared and the burnup values, operating temperature and dimensions are given for each tube. The data reported in Table 4. 2 indicates that at lower temperatures U-2 w/o Zr may increase in volume more rapidly than uranium.

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Clad Failure Due to Thermal Stress - G. T. Geering

Three Zircaloy-2 clad uranium tubes and one Zircaloy-2 clad U-2 w/o Zr tube were tested in a high temperature water-cooled loop. The tubes were 1.430 x 0.520-inch OD and ID, and 16-inches long. The clad was 0.020-inch thick.

The test was intended to determine the dimensional changes of the tubes at an exposure of about 2000 MWD/T. In addition, a previously untested end closure was used. A flat ring of Zircaloy-2 was tightly fitted into the recessed end of the fuel tubes. The ring was welded to the fuel clad to complete the closure. No clearance was allowed between the flat ring and the fuel core. The irradiation ended when the clad of one tube failed adjacent to the end ring. The exposure of the test charge was 1700 MWD/T average at failure. The failed tube was sectioned longitudinally exposing the interior of the end closure. In Figure 4. 14 the cut specimen is shown in the 0 and 180 degree positions. A shear line is visible on the outside of the tube in position 0 degree. About midway along the shear line, a complete separation of the clad occurred. A weld crack also appears in Figure 4. 14. This crack resulted from the pressure of corrosion products under the end ring. In position 180 degrees, Figure 4. 14, the clad shear area appears on the far right, adjacent to the end ring - uranium interface. A photomicrograph of the sheared area is shown in Figure 4. 15. The clad of the inner bore of the tube has also failed due to the build up of corrosion products.

The unfailed end of the tube was sectioned in the same manner as the failed end. An identical clad shear zone was found. The sheared area appears in the upper left of Figure 4. 16. Incipient failure was also found on two additional elements which were examined.

Three fuel elements, including the failed one, were examined in detail. Post-irradiation diameter and length measurements were made. The fuel densities were measured and the burnup of each tube was determined by radiochemistry. The density decreases, calculated volume increases, and core temperatures for three of the tubes are given in Table 4. 3. The tube with the lowest temperature showed a markedly lower volume increase. The volume increase of the U-2 w/o Zr tube appears to be slightly greater than for the uranium tubes.

TABLE 4. 3

VOLUME CHANGE AND CORE TEMPERATURE DATA FOR KSN-1
FUEL TUBES EXPOSED TO 1700 MWD/T

Fuel Element Number		B. U. Per Cent	Per Cent $\frac{\Delta \rho}{\rho}$	Per Cent $\frac{\rho V}{V}$	Max. Core Temp °C
21	U	0.20	-0.9	1.7	450
6	U-2 w/o Zr	0.25	-2.2	3.5	456
17	U	0.26	-2.6	3.1	468

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0 Degree Position 2X

180 Degrees Position 2X

FIGURE 4. 14

Clad Shear Failure in a Tubular Fuel Element

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FIGURE 4.15

Incipient Shear Failure in Fuel Element End Cap Region

Ion Etch

FIGURE 4.16

2 X

Incipient Clad Shear Failure in KSN-1 Tubular Fuel Element

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Irradiation of Fuel Elements with Brazed Closures - J. C. Tverberg

During the past quarter two irradiation experiments were performed on fuel elements with brazed closures. In addition, Radiometallurgical examination was started on another test.

The first irradiation experiment was with KER size tube-tube fuel elements in the ETR 6X9 loop. These elements are 1.830-inch OD by 1.450-inch ID for the outer tube, and 1.050-inch OD by 0.505-inch ID for the inner tube. Each element was 18.5 inches in length. The core was natural uranium, jacketed with coextruded Zircaloy with end caps brazed in position with a 12 Fe + 4 Be + 84 Zircaloy-2 brazing alloy. This test operated at a very high specific power. At the time of discharge, these elements had accumulated 1200 MWD/T (0.14 per cent burnup or 6.8×10^{19} fission/cm³) on the outer elements and the inner elements had accumulated 825 MWD/T (0.1 per cent burnup or 5×10^{19} fission/cm³). Basin examination following discharge indicated that these elements were in excellent condition. Warp measurements, taken in Hanford's Radiometallurgical facilities, indicate that the elements warped 0.125 inch toward the higher flux region of the reactor core. The points of maximum warp correspond to the inflection points on the flattened cosine flux curve. Metallurgical examination is now in progress.

Radiometallurgical examination of the experiment GEH-4-57, 58, described in the last quarterly report, (7) has continued. Briefly, this test was irradiated in the GEH-4 loop of the MTR to test the brazed closure concept. These elements generated low specific power.

After 1050 MWD/T exposure (0.11 a/o burnup or 6×10^{19} fission/cm³), one of the fuel elements ruptured. As mentioned in the last quarterly report first examination indicated that the rupture may have been due to a cladding defect. This rupture occurred at the startup of the fifth irradiation cycle after nine reactor startups. Radiometallurgical examination, however, characterized the rupture as a shear type failure with the following characteristics:

1. The element failed by a shear-type failure mechanism through the cladding of the inner bore. This failure was initiated by cracking of the braze layer between the uranium and Zircaloy cap. Repeated thermal cycling due to the nine reactor startups and shutdowns coupled with a cracked bond between the core and cap caused the repeated shear forces which eventually caused fracture of the cladding.
2. The coextruded bond was completely separated for 180 degrees in the heat-affected zone above the rupture blister. Bond separation appeared to progress by cracking through both the Zircaloy and along the coextruded bond.

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3. Several cracks were observed in the inner weld above the rupture blister. These cracks were superficial in nature and did not enter into the braze zone.
4. Uranium cracking in the heat-affected zone was quite evident. Of particular interest were the cracks which extended 45 degrees from the end cap to the outer jacket. Bond separation appeared to start below the juncture of the 45 degree crack and the jacket.
5. Swelling was quite evident in these elements. The necked-in region of the closure heat-affected zone had expanded to a diameter exceeding that of the end cap. The length decreased slightly.

A sequel to this test, GEH-4-63, 64, was placed in the MTR on July 7, 1961, to irradiate a closure made with the 5 Be + Zircaloy-2 brazing alloy. Irradiation conditions were essentially the same as GEH-4-57, 58. These elements will be exposed for six irradiation cycles, then examined in the Hanford Radiometallurgy facilities.

Irradiation of Variable Heat Treatment Elements - R. S. Kemper

Ten 15-inch long elements of NPR inner tube stock, processed with variable beta heat treatment, were irradiated to an average charge exposure of 1050 MWD/T, and the weighted average tube outlet temperature was 286 C. The elements operated without any unusual incidents. Changes in outer diameter, length, and warp have been determined in the examination of these elements. Inner diameter measurements are currently in process. The results of dial gage measurement of warp with the element supported on rollers are given in Table 4.4. These data indicate that the plane of warp has not changed during exposure and that the most unstable elements were those water quenched (Positions 2 and 8) and slowly cooled (Positions 3 and 7). No evidence of bumping or surface distortion was observed. The changes in length and outer diameter are shown in Figures 4.17 and 4.18. The major differences appear to be due to charge position or specific power; all the elements increased in outer diameter and shortened in length.

TABLE 4.4

WARP MEASUREMENTS OF ELEMENTS FROM PRODUCTION TEST IP-378-A

Charge Position	Warp ⁽¹⁾		Change	Rotation of Plane ⁽²⁾ Degrees
	Pre	Post		
1-upstream	47	47	0	-10
2	25	32	+7	+15
3	12	29	+17	-10
4	18	18	0	-15
5	29	32	+3	+20
6	22	19	-3	-15
7	47	62	+15	+10
8	8	28	+20	0
9	35	27	-8	-30
10	6	5	-1	0

(1) Double throw-mils

(2) Difference between planes of maximum warp, + clockwise, - counter-clockwise

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FIGURE 4.17

Length Change of Elements , PT-IP-378 A, 15-Inch Initial Length

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FIGURE 4.18

Outer Diameter* Change of Elements, PT-1P-378 A
(*Average of measurements at each end and middle)

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Fuel Element Rupture Tests in the ETR - R. L. Call and D. C. Kaulitz

The sixth and seventh tests of the current series of in-reactor rupture tests were performed at the ETR in April and in June, respectively. These tests complete the current series of tests which were being run to determine the failure delay time or "incubation period" and the corrosion rate of the uranium in Zircaloy-2 clad coextruded fuel elements. Both unirradiated and irradiated fuel elements have been used in these tests.

The GEH-P7 loop in the ETR in which these tests were performed, the rupture mechanism, the rupture detection instruments, and the results of the first five tests of the series have been described in previous quarterly reports. (6, 9, 10, 11) The fuel elements used in the sixth and seventh tests were similar to those used in the fifth test except that the defected fuel element of the sixth test was pre-irradiated to 400 MWD/T before testing, and the one for the seventh test, to 1670 MWD/T before testing. Briefly, each test used two 12-inch long Zircaloy-2 clad, coextruded, natural uranium tubes with an OD of 1.43 inches and an ID of 0.52 inches. The defected element was the upper element in each test.

In the sixth test, the first indication of failure of the defected fuel element was a 1/4-pound increase in the loop ΔP which occurred twelve minutes after the defect had been opened to the coolant water. Twenty-two minutes later the ΔP had increased to 3/4 pound. Four minutes later the delayed neutron monitor and the gamma scintillator began showing increased activity in the loop. The ΔP continued to rise slowly, as did the delayed neutron monitor and the gamma scintillator activity, until the test was terminated by shutting down the reactor one hour after the test began. Temperature conditions at the defect during the test were:

Local bulk water temperature	264 C (507 F)
Cladding surface temperature	290 C (556 F)
Cladding-core interface temperature	352 C (666 F)
Core center temperature	481 C (897 F)

In the seventh and final test, the first indication of failure occurred nine minutes after the defect was opened to the coolant water. At that time, the loop ΔP began increasing, and the rupture monitoring instruments also began showing loop activity. In another five minutes the loop activity had increased to the point that the reactor was shut down. The fuel elements from neither the sixth nor the seventh test have as yet been examined radiometallurgically and, therefore, no complete analysis of the results of the series of tests is possible. However, the following conclusion appears to be warranted on the basis of the data now available. Irradiated material fails more rapidly (has a shorter incubation period and corrodes more rapidly following the incubation period) than does unirradiated material of the same geometry. This is probably caused primarily by the internal cracking of the irradiated uranium. The cracks thus formed provide an increased surface for the attack of the water once the cladding has been penetrated.

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Rupture Program for Metallic Fuel Elements - F. E. Young

A new phase of the fuel element rupture program will be initiated with the availability of the PRTR Fuel Element Rupture Facility. A systematic study will be made to obtain fundamental information of the rupture kinetics and mechanism of the N Reactor fuel element - a concentric tube-tube Zircaloy-clad uranium assembly.

The tests will be a continuation of the rupture tests conducted in the ETR. The new PRTR rupture facility will, however, allow pre-irradiated N Reactor fuel elements to be ruptured in-reactor. Four pre-defected inner component elements are presently being irradiated for use in future tests. Additional inner and outer components will be pre-irradiated as material and irradiation facilities are available.

Defect-Tests of Irradiated Coextruded Fuel Material - J. W. Goffard

Two 3200 MWD/T exposure coextruded KER inner fuel tubes, one unalloyed uranium and one U-2 w/o Zr alloy, were defect-tested in 300 C, 1650 psi water by Coolant Systems Development Operation in their IRP loop. The unalloyed uranium fuel tube incubated 7-1/2 minutes before failing and the test loop filter accumulated 20R activity just 4-14 minutes later. Loop cool-down was started at this time. Five minutes later the loop temperature had decreased to 160 C, fuel corrosion had become insignificant, and the filter activity had increased to 40R. Examination of the coextruded uranium Zircaloy-2 clad fuel tube revealed that it had broken into three major pieces and six lesser pieces. The 18-inch long tube had a spiral cleavage from one end to the other and both unbonded end caps had separated.

The U-2 w/o Zr alloy fuel tube incubated 10-1/4 hours before failure indications. Eighty-eight minutes later the loop filter activity had reached 20R and cool-down was started. After four minutes the temperature was 200 C and fuel corrosion had become insignificant. The final filter activity was 23R. Examination of the failed alloy fuel tube revealed that it had broken into two pieces at the cross-section through the defect. The observed order-of-magnitude difference in the "rate of failure" of the U and the U-2 w/o Zr fuel material is attributed to the alloy addition--since all other test parameters (including irradiation history) were identical.

Thermocoupled Fuel Element Tests - R. L. Call

A series of three tests using thermocoupled fuel rods has been charged into a high pressure, high temperature, recirculating loop in one of the Hanford reactors. These tests were performed to measure the temperature effects of "crud-film" deposition on Zircaloy-2 clad fuel elements. A description of the fuel elements and the results of the first two tests of the series are given in previous quarterly reports. (6, 11)

Briefly, the results of the first two tests are as follows: the first thermocoupled rod was irradiated for four weeks in 260 C water and for three weeks in 30 C water, after which, a rupture occurred in the loop and the test

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was terminated. The thermocoupled fuel rod was examined in a radio-metallurgy cell and no evidence was found to indicate that it had failed. The second thermocoupled fuel rod was irradiated for two weeks in 260 C water when again rupture signals were received from the loop. The rupture signals occurred 24 hours after a reactor startup. The thermocoupled rod was examined in a radiometallurgy cell and the cladding was found to be split over both of the thermocouples lying just under the cladding. The third thermocoupled fuel rod failed 6-1/2 hours after the reactor started up. It failed in the same way as the second thermocoupled rod.

In neither of the last two tests was the cause of the failure definitely determined. It is believed that either strain incompatibility between the uranium and the cladding during startup caused the cladding over the peripheral thermocouples to fail, or the cladding was thinned excessively when the holes for the thermocouples were drilled. A new thermocoupled fuel rod is being designed which should eliminate the danger of failure from either of the above causes.

FACILITIES

Structural Materials Loops in the ETR - D. C. Kaulitz

A loop modification is being made at the ETR, Idaho Falls, Idaho, to provide a pressurized water loop in which structural materials can be irradiated under controlled and measured conditions of temperature, pressure, and flux.

The modifications include re-piping existing facilities so the present 6X9G7 loop can be used in parallel with a new loop for the 6X6M3 position. This permits using the existing 6X9G7 position for the structural program and the new 6X6M3 loop for the fuel program. Much of the equipment for the existing 6X9 loop will be used in common by the two in-pile loops. Both loops are designed to operate at 2000 psi, 300 C at 150 gpm.

The loop modifications will be complete about September 1, 1961. Samples for the structural loop, are fabricated and ready for insertion. The new in-reactor tube for the 6X6 position is made of Zircaloy-2. A fuel test will be inserted in that position about October 1, 1961.

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CERAMIC FUELS DEVELOPMENTVibrational Compaction Studies - J. J. Hawth

Densities of vibrationally compacted, fused UO_2 were compared with densities that are theoretically achievable⁽¹⁾ using mixtures of spheres having similar particle size distributions.

Fused UO_2 having an average particle density of 98.5 per cent TD was crushed, screened and blended to prepare mixtures having six different particle size distributions. One-kilogram of each mixture was vibrated to maximum density in 0.505-inch ID Zircaloy tubes on a 5000 pound-force shaker. The compaction efficiencies calculated for spheres are compared with measured values for fused UO_2 in Table 5.1. Results show that for mixtures of particles having the same size ratios between fractions, the mixture having larger particles in each fraction can be compacted to a higher density. Compaction efficiencies that are higher than predicted reflect either the effect of particle shape, or more favorable particle size distribution caused by vibration, or both. Compaction efficiencies lower than predicted may be explained on the basis of the topological characteristics of the material-- some particles have highly irregular surfaces that will not allow penetration of fines. The effective particle density is thus lower than that indicated by fluid displacement.

Two one-foot long fuel elements containing mechanically dispersed UO_2 and ThO_2 were fabricated for irradiation in the MTR. One fuel core consisted of 10 w/o ThO_2 in enriched UO_2 (1.38 per cent U^{235} in U). The other consisted of 13.7 w/o UO_2 (enriched to 10 per cent U^{235} in U) blended with ThO_2 . The particle size compositions of the fuel are as follows:

I			II		
<u>UO_2 - 10 w/o ThO_2</u>			<u>ThO_2 - 13.7 w/o UO_2</u>		
	<u>w/o</u>	<u>Mesh Fraction</u>		<u>w/o</u>	<u>Mesh Fraction</u>
UO_2 :	67.5	-3+6	ThO_2 :	68.8	-3+6
	17.5	-28+100		17.5	-28+100
	5	-200			
ThO_2 :	10	-100	UO_2 :	13.7	-100

The fuel materials were compacted to 86 per cent TD (90 per cent compaction efficiency) in 1.12-inch OD x 0.030-inch wall stainless steel tubing.

Fused UO_2 was vibrationally compacted to 88-89 per cent TD in a four-foot long, 0.564-inch OD x 0.009-inch wall, AISI-406 stainless steel tube. Magnetic Force welding was used for the closures that withstood applied accelerations greater than 80 g's, both before and during compaction of the oxide. Autoclaving at 400 C, 1000 psi for 69 hours had no observable effect on the thin-wall tubing.

TABLE 5.1

Size Fraction	Approx. Diam. Ratio, d/d_c	w/o	Mesh Size	
			I	II
C	1	91.4	-6+8	-10+14
M	0.414	6.5	-14+20	-28+35
F	0.225	2.1	-20+28	-48+65
Compaction Efficiency				
Calculated			81	81
Measured			89	87
C	1	88.9	-6+8	-10+14
M	0.217	10.0	-20+28	-48+65
F	0.083	1.1	-48+65	-100+150
Compaction Efficiency				
Calculated			83	83
Measured			93	87
C	1	77.3	-6+8	-10+14
M	0.178	10.1	-28+35	-48+65
F	0.013	12.6	-325	-400
Compaction Efficiency				
Calculated			96	96
Measured			93	90

Three 8-foot long nested tubular fuel assemblies (Mark II-C) were fabricated for testing and evaluation in the PRTR. Straight spacer ribs were attached to the 0.060-inch wall Zircaloy-2 cladding by a continuous, high-frequency resistance welding process. Fused UO_2 was crushed, sized and vacuum outgassed six hours at 750 C to minimize gas release during subsequent irradiation. The fuel was vibrationally compacted to a bulk density of 87-89 per cent TD. End caps for the tubular segments were attached by a double closure technique involving resistance seam welding followed by inert gas, tungsten arc welding. The fuel components, after the final closure, were etched, autoclaved and visually inspected. Rib welds and closure were inspected using ultrasonic and radiographic techniques, respectively. Planned studies involving the three assemblies include extensive out-of-reactor and in-reactor testing. One assembly will be flow tested in a PRTR mockup coolant channel. One assembly is scheduled for irradiation in the PRTR in the near future, and one will be used for critical testing and/or PRTR irradiation.

Hot Swaging of UO_2 - P. L. Farnsworth

A marked improvement in the surface finish of hot swaged fuel rods was achieved using "BMI-r" and "604-A" hot metal-working lubricants. The swage dies were swabbed with either of these lubricants just before a fuel rod was hot swaged. Previously, lubrication was achieved by periodically injecting a water suspension of colloidal graphite through spray nozzles into the die cavity during hot swaging. Fuel rods lubricated with graphite-water exhibited good surface finish. However, post-swaging examination of the swaging machine revealed that the water in the lubricant had seriously corroded the swaging machine components.

An induction coil to heat fuel rods for hot swaging was modified to achieve more efficient heating. This coil differed from the original coil in that it consisted of three relatively short coils, connected in parallel, rather than of one long, continuous coil. Each section of the new coil contains approximately 60 inches of copper tubing to match the length of the secondary transformer coil. The superior heating efficiency of the three-section coil permitted heating of Zircaloy-clad fuel rods to 850 C at a feed rate of 5 feet per minute, whereas the maximum feed rate at which the same fuel rod could be heated to 850 C in the continuous coil was one foot per minute. The continuous coil and the new three-section coil are shown in Figure 5. 1.

Inconel-X, Hastelloy-F, and Fansteel-80 (Nb-1 per cent Zr) clad fused UO_2 fuel rods were heated to 850-1200 C and hot swaged. The fuel rods were fed through the induction coil and the swage at two feet per minute. Both the Inconel-X and Hastelloy-F clad fuel rods were swaged successfully. The Fansteel-80 clad fuel rods, however, failed during swaging, presumably because of atmospheric contamination and embrittlement of the Fansteel-80 cladding during heating. Future experiments with this cladding material will require a protective, non-oxidizing atmosphere surrounding the heated fuel rods. The results of this experiment are summarized in Table 5. 2. The highest fuel density in each cladding material was

achieved at the lowest swaging temperature (700 C). The lower densities reached at the higher swaging temperature were presumably a result of the lower cladding strength at temperature. Experiments to establish the optimum heating and swaging temperatures for fused UO_2 clad in Zircaloy are now in progress.

TABLE 5. 2
EFFECT OF CLADDING MATERIAL AND TEMPERATURE ON
DENSITY OF HOT SWAGED FUSED UO_2

Cladding Material	Heating Temperature ($^{\circ}\text{C}$)	Swaging Temperature ($^{\circ}\text{C}$)(estim.)	Density Achieved (Per Cent TD)	Remarks
Inconel-X	850	700	93	
	1000	850	92	
	1100	950	91	
	1200	1000	90	
Hastelloy-F	850	700	92	
	1000	850	91	
	1100	950	90	
Fansteel-80	1100	950	92	Rod split

Feed Rate - 2 ft/min

Photomicrographs (Figure 5. 2) of selected areas of Inconel-X clad, fused UO_2 that was heated to 1200 C and swaged at about 1000 C show that abutting particles were deformed plastically during hot swaging. This effect is not apparent in fused UO_2 swaged at 600 C.

Photomicrographs of cross-sections of the Inconel-X, Hastelloy-F, and Fansteel-80 cladding materials, before and after hot swaging, are shown in Figures 5. 3, 5. 4, and 5. 5, respectively. Only Inconel-X cladding swaged at 700 C exhibited evidence of cold work. This sample was swaged in the temperature range 650-870 C. In this range of temperature, Inconel-X exhibits hot shortness. The other swaging temperatures were above 870 C. No excessive precipitation of a second phase in the grain boundaries was apparent in any of the Inconel-X samples. The Hastelloy-F sample that was swaged at 700 C shows an undesirable amount of intergranular carbide precipitate. The extreme brittleness of the Fansteel-80 during hot swaging is emphasized by the extent to which cracks were formed on both the inner and outer surfaces.

Nineteen 8-foot long, Zircaloy-clad fuel rods containing -60 mesh fused UO_2 were successfully hot swaged for assembly into a PRTR fuel element.

A. Continuous Coil

B. Three-Section Coil

FIGURE 5. 1

Induction Heating Coils Used for Hot Swaging

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FIGURE 5.2

Fused UO_2 (-60 Mesh),
Heated to 1200 C and Swaged
(Estimated "In-Swage" Temperature = 1000 C)

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FIGURE 5.3

Inconel-X Cladding, Before and After Hot Swaging

FIGURE 5.4

Hastelloy-F Cladding, Before and After Hot-Swaging

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FIGURE 5.5 A
Fansteel-80 (Nb- 1 Per Cent Zr) Cladding Before Hot-Swaging

FIGURE 5.5 B

Fansteel-80 (Nb - 1 Per Cent Zr) Cladding After Hot-Swaging.
Heated to 1100 C and Swaged (Estimated "In-Swage" Temperature = 950 C)

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Gas Evolution from Swaged UO_2 - J. A. Christensen

Gas pressures developed within a swaged UO_2 PRTR fuel element were measured to determine whether or not sorbed gases represent a rupture hazard during irradiation. Pressures were measured with a strain gauge transducer connected via a 1/16-inch stainless steel capillary to a 30-inch segment of swaged fuel rod. Coupling between the steel capillary and the Zircaloy end cap was effected through a zirconium-iron eutectic. The rod contained (-20+100) mesh, sintered and crushed UO_2 swaged to a bulk density of 85 per cent TD. The internal pressure was continuously monitored as the rod was heated in an argon atmosphere tube furnace between room temperature and 800 C. The rod failed between 800 C and 900 C, probably because of a rupture in the eutectic. The maximum pressure measured was 130 psi at 800 C. The plot of unit gas evolution versus temperature is shown in Figure 5.6.

High Frequency Resistance Welding - L. R. Vancott and L. E. Mills

Studies of high frequency resistance welding during the past year have shown the process to be applicable and practical for continuously bonding 0.075-inch wide spacer ribs to Zircaloy fuel element cladding.

Characteristics of spacer ribs welded by this process are:

(1) weld joint shear strength in the range 708 to 990 pounds per lineal inch of rib, (2) no loss of corrosion resistance of the vapor-blasted weld area after the weld is completed in air (Argon consumption of 50 cu ft/8-foot rib was eliminated), (3) no distortion of empty or filled cladding tubes after welding of straight or spiral ribs, and (4) no subsequent machining required of ribs over 0.135-inch high (these can be applied to specified height ± 0.002 inch).

Uneven heating rates of the rib and cladding, caused by the difference in effective cross-sectional areas, produce a slight cold lap. Hot rib metal that is squeezed over relatively cold cladding metal does not become completely fused to the cladding. The uneven heating pattern is illustrated in the micrograph (Figure 5.7 bottom) taken under polarized light. A typical weld nugget that covers 75 to 90 per cent of the contact width is shown under bright field illumination of the same cross-section (Figure 5.7 top). The reliability of the process was improved. Approximately 15 minutes are required for welding each rib. New electrode configurations and materials are expected to further increase reliability.

PRTR UO_2 Fuel Elements - M. K. Millhollen

Shortly after the low temperature PRTR critical tests, all UO_2 19-rod cluster fuel elements were visually examined and found to contain small deposits of ion exchange resin and other material. All UO_2 fuel elements were disassembled, cleaned, and reassembled to avoid the possibility of in-reactor fuel rod damage caused by such deposits.

Disassembly was accomplished by (1) cutting off the outer wrap wire, (2) knocking out the pin holding the bottom hanger fitting to the center rod, (3) removing the bottom hanger fitting, and (4) unscrewing the nuts fastening the fuel rods to the top hanger fitting. All fuel rods, hangers, and nuts were ultrasonically cleaned, steam cleaned, and wiped with alcohol-saturated towels. The elements were reassembled using new outer wire wraps and bottom hanger pins.

Following the first PRTR power operation (< 10 Mw) all swaged UO_2 19-rod cluster fuel elements were again visually examined under six feet of water. No element exhibited structural damage or loosening of wire wraps. A few elements were lightly scratched during discharging, but no scratch appeared to have any significant depth. Small collections of reddish colored crud were visible on ~ 75 per cent of the fuel elements. All of the elements were recharged into the PRTR.

Hydriding of Zircaloy - G. R. Horn and J. L. Bates

"Zircaloy-4" (low nickel Zircaloy-2) continues to reveal a significantly greater resistance to hydriding than does Zircaloy-2. The latest evidence comes from an irradiation test of a seven-rod cluster (GEH-12-20) that included purposely defected Zircaloy-2 and "Zircaloy-4" clad fuel rods. Post-irradiation examination of a "Zircaloy-4" clad rod, defected at the center with a 0.006-inch diameter hole, revealed less than 50 ppm hydrogen in the cladding near the defect. Examination of an adjacent, similarly defected Zircaloy-2 clad rod revealed greater than 500 ppm hydrogen near the defect. The pre-irradiation hydrogen concentration in each cladding material was approximately 50 ppm. The fuel element was irradiated 17.8 equivalent full power days with a maximum surface heat flux of $\sim 275,000$ Btu/hr-ft². During most of the irradiation, the bulk inlet temperature of the cooling water was ~ 100 F.

The second Hanford defect test (HD-2) was discharged from KE Reactor after successfully completing 20 weeks of irradiation. The element will be metallographically examined to determine the effect of long term irradiation at relatively low heat generation on the hydriding of Zircaloy-2 and "Zircaloy-4" (low nickel Zircaloy-2). The 12-inch long four-rod cluster element contained sintered and crushed, natural UO_2 swaged to a bulk density of 90 per cent TD. Two rods were clad in Zircaloy-2 and two were clad in "Zircaloy-4". One each of the Zircaloy-2 and "Zircaloy-4" clad rods was defected with an 0.006-inch diameter hole near the center of the rod. The test element was irradiated in low temperature water which contained 7.7-12.3 ppm O_2 . The irradiation included several power cycles. Very small amounts of fission products were released to the coolant.

The third Hanford defect test element (HD-5) was fabricated for irradiation at a heat generation rate $\sim 2\text{-}1/2$ times that of HD-2.

FIGURE 5.6 (Revised 11/15/61)

Total Gas Evolution From Swaged UO_2 Fuel Rod

Bright Field
100 X

Polarized Light
100 X

FIGURE 5. 7

Micrograph of Cross Section of High Frequency Resistance Welded Rib

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Nested Tubular Fuel Element Irradiation (GEH-4-22) - M. K. Millhollen and G. R. Horn

A prototypic Mark-II C nested tubular fuel element⁽²⁾ was discharged after 16 equivalent full power days of irradiation in the ETR because of fission product leakage. During the irradiation period, the fuel element underwent ten reactor scrams from full power and recoveries that produced severe thermal shocks in the element. Further severe thermal treatment of the element occurred after several days of irradiation when the in-reactor section of the test loop was inadvertently isolated during a reactor outage. The pressure in that section increased from 200 psi to 1100 psi in 15 minutes due to gamma heating. The loop was vented and coolant flow was again started, but the loop was isolated twice more for short periods before the outage was finished. Pressure did not rise above 500 psi during the last two isolations.

After the loop isolations, the fuel element performed normally while the reactor ran 17 hours at power levels between 45 and 80 Mw. The reactor later was shut down because of core problems. Subsequently, the reactor power was raised rapidly to 100 Mw. After two hours at 100 Mw the reactor was scrammed because of fission gas leakage from the test element.

The maximum surface heat flux of $\sim 350,000$ Btu/hr-ft² occurred through the surface of the inner cladding of the inner tube. The calculated total exposure was ~ 450 MWD/T UO₂.

Visual examination of the discharged fuel element revealed no obvious defects (Figure 5.8). The element was remotely disassembled, visually examined, and leak tested. A leak was located on the inner cladding of the inner fuel tube approximately midway between the ends. Further examination will be performed at Hanford.

Fused UO₂ Quality - D. W. Brite

The microstructure of fused UO₂ containing inclusions was remarkably homogenized by annealing four hours in hydrogen at 1700 C. Inclusions 0.06 mm in diameter before annealing disappeared, leaving voids 0.008 mm in diameter or smaller. Photomicrographs of the microstructure appear in Figure 5.9. The inclusions are judged to be uranium metal from their rate of tarnishing in air and their etching behavior, which corresponds to that of uranium metal inclusions in fused UO₂ identified by G. R. Cole of SRL with the aid of an electron microprobe analyzer.

The existence of a sub-stoichiometric UO₂ having an O/U ratio as low as 1.96 in samples quenched from the liquid phase has been reported.⁽³⁾ The annealed UO₂ has an O/U ratio of 2.001 as determined by the controlled potential coulometric method⁽⁴⁾ which, however, determines only the ratio of U (VI) to U (IV) in the dissolved sample, and consequently does not detect sub-stoichiometry in UO₂.

The amount of gas extracted from the annealed fused UO_2 by vacuum outgassing was approximately one-fourth as great as from the nonannealed UO_2 . Results of vacuum gas extraction analyses appear in Table 5. 3. The total volume of gas extracted in either case was less than the 0. 1 cc/gm UO_2 currently specified for fused UO_2 .

TABLE 5. 3

VACUUM GAS EXTRACTION ANALYSES OF FUSED UO_2

(1000 C, 15 minutes, -2+6 mesh particles)

	<u>Sample Before Annealing</u>	<u>Sample After Annealing 4 Hours, at 1700 C in H_2</u>
Gas content, cc/gm UO_2 (STP)	0. 09	0. 021
ppm N_2 (weight basis)	4	0. 4
ppm H_2	3	1
ppm CO	1	2
ppm CO_2	8	2
ppm O_2	< 1	0. 01

High Energy Impact Forming of Ceramic Fuels - D. W. Brite

Uranium dioxide powder can be compacted to a bulk density in excess of 10. 86 g/cc (>99 per cent TD) by high energy impactation using a Dynapak machine. A Bridgeman anvil apparatus, modified for impact application and capable of exerting an impact pressure of 500,000 psi is used with the machine. (5) The UO_2 , contained in a stainless steel capsule having an out-gas hole in one end, is heated to 1000-1200 C, transferred to the cavity of a die attached to the Dynapak machine, and quickly compacted by the high energy impact of a punch in the die. An expendable mild steel ring retains the sample between the faces of truncated conical punches. Pressures were calculated from data from strain gauges attached to the punch.

A series of experiments was performed to simultaneously reveal the effects of temperature, pressure, and O/U ratio on the density obtained by high energy impactation of Micronized UO_2 powder. A statistical analysis of the data is being conducted. Tentative conclusions are: (1) temperature variations in the range 1000-1200 C and O/U ratio variations in the range 2. 01-2. 26 do not cause significant changes in the compacted density of uranium dioxide at impact pressures of approximately 500,000 psi; and (2) at lower impact pressures, greater compacted UO_2 densities are obtained at higher temperatures and/or higher O/U ratios.

FIGURE 5.8
Nested Tubular Fuel Element (Test GEH-4-22) After Irradiation

Typical Microstructure Before Annealing

Typical Microstructure After Annealing Four Hours in H_2

FIGURE 5. 9

Microstructure of Fused UO_2 Containing Inclusions
(As Polished, 100 X)

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Swelling corresponding to a decrease of density of approximately two per cent occurred when UO_2 compacted by high energy impact was annealed at 1300-1700 C. One sample, compacted to a bulk density of 98.5 per cent TD and having an O/U ratio of 2.04, was reduced in hydrogen at 800 C to an O/U ratio of less than 2.01. The density remained practically the same, 98.3 per cent TD. Part of the sample was then heated in hydrogen for 12 hours at 1300 C, and the density decreased to 96.2 per cent TD. Another portion was heated in hydrogen for 12 hours at 1500 C, and the density decreased to 96.4 per cent TD. Two high energy impact-formed UO_2 specimens, compacted from -200 mesh and -325 mesh sintered and crushed UO_2 to densities of 97.9 and 98.1 per cent TD, respectively, were heated in hydrogen for 12 hours at 1700 C. The densities decreased to 96.4 and 95.6 per cent TD, respectively. The expansion upon annealing high energy impact formed UO_2 may be caused by the release of residual stresses, or may be due to expansion of gases trapped in the UO_2 during compaction. Further experiments are planned to determine the reason for this behavior.

Micronized UO_2 powders containing 0.5 and 0.1 w/o TiO_2 were compacted by high energy impact at 1100 C and 500,000 psi in unsealed stainless steel capsules to densities of 97.3 and 97.8 per cent TD, respectively. Photomicrographs of the compacted UO_2 , Figure 5.10, reveal no evidence of reaction (such as the formation of a liquid eutectic phase) between the TiO_2 and the UO_2 . The TiO_2 may, in fact, inhibit the densification of the UO_2 , as a density of 10.85-10.90 g/cc (99 per cent TD) is usually attained with Micronized UO_2 alone under the conditions of temperature and pressure used for the experiment.

Electrodeposited UO_2 crystals of approximately -28+48 mesh size were high energy impact formed at 1100 C and 500,000 psi in an unsealed stainless steel capsule to produce a bulk material having a particle density of 10.88 g/cc (99.2 per cent TD). Photomicrographs and electron micrographs of the compacted UO_2 , shown in Figure 5.11, reveal extensive plastic deformation of the crystals.

Minus 6 mesh fused uranium monocarbide was sealed in vacuum in a 3/4-inch OD 304L stainless steel capsule by electron beam welding. The capsule was placed inside a similar, one-inch OD capsule. Alumina insulation was vibrated into the 1/16-inch annular spacing, and the outer capsule was sealed by a tungsten arc fusion weld. The double-sealed capsule was induction heated at 1200 C for five minutes, placed in the Dynapak die, and compacted at 500,000 psi. The impacted UC attained a density of 13.28 g/cc (97.4 per cent TD). Photomicrographs of the UC before and after high energy impact forming appear in Figure 5.12. Plastic deformation of the UC crystals is apparent. Extensive reaction occurred between the UC and the 304L stainless steel.

Thoria, alumina, and a 50 m/o thoria-alumina powder mixture were compacted in unsealed stainless steel capsules at 1200 C and 500,000 psi. Densities attained were 9.59 g/cc (95.9 per cent TD), 3.37 g/cc (84.9 TD), and 6.33 g/cc (90.0 per cent TD), respectively. The thoria was obtained by

calcining thorium nitrate, and the alumina was Linde-A metallographic polishing powder, which has an average particle diameter of 0.3 micron. The compacted thoria had a glassy appearance, and was recovered in fragments 1/8 to 1/4 inch in diameter. Photomicrographs of the compacted materials appear in Figure 5.13.

Spherical UO₂ Crystals - D. W. Brite

Spherical fused UO₂ single crystals were provided for several European sites. The crystals will be used for measuring diffusion rates of fission gases in UO₂ and for other basic property studies. The crystals were annealed in hydrogen at 1700 C to relieve stresses and assure stoichiometry before grinding them into spheres, and were analyzed individually for microstructure, O/U ratio, density, and purity. Diameters of the finished spheres were between 0.25 and 0.10 inch.

Spheres were also prepared from electrodeposited UO₂ crystals and from high energy impact-formed UO₂ by similar techniques.

Fuel Evaluation - G. R. Horn, J. L. Bates, and P. L. Farnsworth

Single crystal wafers of UO₂ (0.35 inch by 0.35 inch by 0.055 inch) are being irradiated in the ETR core. Measurements of physical properties and microstructure will be made on these specimens at several levels of integrated exposure.

Two ThO₂-UO₂ fuel mixtures were successfully irradiated in the MTR. The tests were performed to evaluate the extent of homogenizing of physical mixtures of two isomorphous fuel materials during irradiation at high central fuel temperature. Both test capsules were irradiated approximately 15 equivalent full power days with a maximum surface heat flux of ~600,000 Btu/hr ft². Fuel mixtures were contained in 12-inch long stainless steel tubes, 1.1 inches OD. The fuel in one case (GEH-4-61) was ThO₂ - 90 w/o UO₂ enriched to 1.38 w/o U²³⁵ in U and in the other case (GEH-4-65) was ThO₂ - 14 w/o UO₂ enriched to 10 w/o U²³⁵ in U. Post-irradiation examinations are scheduled.

The first significant irradiation of hot swaged UO₂ (Zircaloy-4 clad, 4-rod cluster containing fused material) was terminated after two scheduled MTR cycles in the GEH-4 facility. The element (GEH-4-60) attained an exposure of 550 MWD/T_U while operating with a maximum surface heat flux of ~320,000 Btu/hr ft². The 12-inch long, 4-rod cluster comprised three rods swaged at 800 C to a bulk density of 92 per cent TD, and one rod cold swaged to a bulk density of 88 per cent. The cladding of the hot swaged rod was 0.023 inch thick Zircaloy-4. Post-irradiation examination has been scheduled.

UO₂ - 0.1 Per Cent TiO₂

UO₂ - 0.5 Per Cent TiO₂

FIGURE 5. 10

Microstructure of UO₂-TiO₂ Compacted by High Energy Impact
(250 X, Polished and Etched)

Optical Micrograph, Polished and Etched

5000 X

20,000 X

Electron Micrographs, Fracture Surface, Positive Replica

FIGURE 5.11

Microstructure of High Energy Impact Formed, Electrodeposited UO_2

Before High Energy Impaction

After High Energy Impaction

FIGURE 5.12

Microstructure of Fused Uranium Monocarbide
(100 X, Polished and Etched)

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Microstructure of ThO₂, Al₂O₃, and 50 m/o ThO₂-Al₂O₃
Compacted by High Energy Impact (100 X, As Polished)

FIGURE 5.13

Al₂O₃

ThO₂

HW-70557

5.24

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High Temperature Electron Microscopy - J. O. McPartland, J. L. Daniel, and S. J. Walter

High temperature capability of the JEM-6A electron microscope was extended to 2000 C for transmission microscopy. Temperatures to 2000 C can be reached in less than 15 minutes with the specimen under continuous observation. No significant limitations were imposed by expansion or by magnetic deflection caused by heater current changes.

The specimen is supported on the grid-heater element, a 200 mesh, 1/8-inch diameter tungsten screen disk provided with solid tabs for electrical connections (Figure 5. 14). The grid-heater is clamped in a modified JEM-6A high temperature stage. Effects of expansion are minimized by the geometry of the grid, symmetrical support, and the relatively great depth of field of the electron microscope.

Temperature calibration of the high temperature stage is preliminarily based on an average of values recorded during previous experiments. A more accurate final calibration of each grid-heater is made by optical pyrometry in an auxiliary vacuum system after microscopy is completed. The calibrated grid temperatures, plotted as a function of heater current, are accurate to ± 100 C, neglecting effects of electron beam heating.

The high temperature stage was used to observe the behavior of a thin layer of UO_2 vacuum deposited on a carbon supporting film laid on the tungsten grid-heater. The UO_2 recrystallized at an increasing rate as the temperature was raised, until rapid melting occurred. Crystal patterns and melting behavior were reminiscent of observations reports from work on the high temperature optical microscope. A series of micrographs and electron diffraction patterns was prepared for further study and evaluation, (Figure 5. 15).

UO_2 Thermal and Electrical Conductivity - J. L. Daniel and S. J. Walter

Measurements of thermal and electrical conductivity were completed on the last of the current series of UO_2 samples in the cooperative research program involving HLO and Battelle Memorial Institute. The series has included UO_2 pieces fabricated by a variety of methods, and several specimens subjected to low-temperature irradiation in Hanford reactors. Initial curves for most of the samples examined are shown in Figures 5. 16 through 5. 19. Adjustments of curve shapes and positions may be made when detailed evaluation of data has been completed.

Variables other than those shown on the curves may have a significant effect on interpretation of the results. Among these variables are: oxygen/uranium ratio; crystal species present; size, shape, distribution and orientation of pores and particles; and physical condition of the samples before, during and after thermal and electrical measurements. Evaluation of these factors for all or selected specimens is currently in progress.

High Temperature Studies of Irradiated UO_2 - J. A. Christensen

Simultaneous comparison of irradiated and nonirradiated UO_2 at temperatures near the melting point provided further evidence of greater crystal stability at high temperatures for irradiated specimens. Irradiated and nonirradiated specimens of approximately the same size and shape were heated simultaneously on a resistance heated tungsten strip. The irradiated oxide (0.20 a/o burnup) was selected from a specimen which had previously been observed to melt at 2880 C, 90 degrees above the melting point of nonirradiated UO_2 . When the temperature was raised to 2790 C, the nonirradiated oxide melted at the point of contact with the filament and rapidly vaporized. The irradiated material did not melt at that temperature and vaporized relatively slowly. About 70 per cent of the nonirradiated UO_2 vaporized during 20 seconds at 2790 C, while the irradiated oxide was less than 30 per cent vaporized. When the temperature was raised to 2880 C, the irradiated UO_2 melted and both specimens vaporized rapidly. After five minutes at 2880 C, all of the nonirradiated and about 90 per cent of the irradiated UO_2 had vaporized. A small amount of solid material remained at the point of contact between the irradiated oxide and the tungsten until the filament melted at 3400 C. Photomicrographs of the described sequence of events are shown in Figure 5.20.

Observations of an irradiation-induced increase in the melting point of UO_2 have led to the proposal of a hypothetical liquidus for the uranium-oxygen system. Melting behavior, X-ray diffraction studies, metallographic examination, and thermodynamic calculation predict the existence of a stable UO_{2-x} solid solution above 1800 C. The kinetics of dissociation reactions liberating oxygen in a nonequilibrium system indicate a high probability that O/U ratios less than 2.00 are characteristically reached before stoichiometric UO_2 can be heated to its melting point. The increase in melting point during irradiation can then be explained in terms of stabilization of higher O/U ratios by fission fragment cations having valence less than +4. Such cations can form an anion deficient defect lattice in which hyperequilibrium oxygen concentrations are stable. Increasing the O/U ratio in such a manner is, in effect, equivalent to moving along the liquidus to higher oxygen concentration and higher temperatures. Probably the maximum in the liquidus occurs at an O/U ratio greater than 2.00 because of the dissociation reactions which occur and of the greater solubility of oxygen, as compared with uranium, in $\text{UO}_{2.00}$. Possible additions to the uranium-oxygen phase diagram are shown by dotted lines in the accompanying Figure 5.21.

The concentration of fission fragments required to cause significant melting point increase is much less than the concentration of impurities usually needed to form a defect lattice. Investigators at BAPL have found, however, that the fission fragments in UO_2 migrate to positions described by an exponential concentration distribution along the mass gradient of each individual grain with the highest concentration in the grain boundaries. Grain boundary regions thus have a much higher bulk fission fragment concentration and may form a more refractory shell that supports the body even after the interior of the grain has melted.

SPECIMEN GRID-HEATER ELEMENT
200 MESH .001 INCH THICK TUNGSTEN

FIGURE 5.14

Electron Microscope High Temperature Stage

Ambient Temperature

1300°C

1400°C

1500°C

1700°C

2000°C

FIGURE 5. 15

Transmission Electron Micrographs
Evaporated UO_2 and Carbon Film During Heating
(9000 X)

FIGURE 5.16.

Thermal Conductivity of Nonirradiated UO₂
(Corrected to 100 Per Cent Theoretical Density)

FIGURE 5.17

Effect of Irradiation on Thermal Conductivity of UO_2
(Corrected to 100 Per Cent Theoretical Density)

FIGURE 5.18

Electrical Conductivity of Nonirradiated UO_2
(First Heating Cycle)

FIGURE 5.19
Electrical Conductivity of Irradiated UO₂
(First Heating Cycle)

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5.33

HW-70557

FIGURE 5.20

Melting and Volatility Comparison - 25 X
Left: Unirradiated Right: 0.20 Per Cent Burnup

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FIGURE 5.21

Uranium-Oxygen System, Hypothetical Phase Diagram

Three fuel rod specimens were tested under simulated in-reactor thermal-hydraulic conditions. Four-inch segments of swaged PRTR fuel rods were centrally heated to obtain surface heat fluxes between 0 and 310,000 Btu/hr ft². The water coolant flowed through a 1/8-annulus at rates up to 1 gal/min.

Two of the specimens were heated at a uniform rate until the melting point of the axial tungsten heater was reached, at which time the heat transfer rate at the surface of the cladding was 200,000 Btu/hr ft². UO₂ sintering and grain growth comparable to that observed in some irradiated fuel cores was revealed by examination of cross sections of the specimens. The extent of grain growth indicated an average center temperature of about 2500 C. That some portions of the sample were hotter than this was indicated by evidence of UO₂ melting at several places adjacent to the tungsten heater.

A third capsule was heated to produce a surface heat flux of 180,000 Btu/hr ft² for 30 minutes after an initial abrupt increase to 190,000 Btu/hr ft² which corresponded to a maximum tungsten temperature of 3400 C. The heat input was then increased until the tungsten again reached 3400 C. The second temperature maximum occurred at a surface heat flux of 310,000 Btu/hr ft², indicating a significant increase in thermal and/or electrical conductivity of the swaged fuel. Such a change reflects the sintering and grain growth that occurred during 30 minutes at the surface heat flux of 180,000 Btu/hr ft². An increase in thermal conductivity of approximately 60 per cent would account for the enhanced heat dissipating capacity.

Voids in Irradiated UO₂ - J. L. Bates

Post-irradiation studies on a variety of cylindrical UO₂ fuel elements irradiated under conditions that produce high central core temperatures and large radial temperature gradients revealed five types of voids and pores. Each type of void is located in a region of characteristic UO₂ microstructure. The types of voids, Figure 5. 22, and their related microstructures in order of their occurrence from the cladding to the thermal center of the fuel are:

1. Anisometric Voids - Small "irregular" voids, 10⁻²-10⁻³ mm in diameter are observed predominantly at the boundaries between the sintered UO₂ grains.
2. Lenticular Voids - "Plate-like" or "lenticular" voids, 0. 01 to 0. 15 mm in diameter and as much as 0. 02 mm thick are found in the transition region between the large equiaxed grains of sintered UO₂ and the small columnar grains. These voids are oriented with their long axis perpendicular to the radius of the fuel rod. Each void represents the inner, high temperature end of a small radially oriented columnar grain.
3. Large Irregular, Tubular and Spherical Voids - A band of tubular, spherical and irregular voids with a well defined outer edge occurs in the large columnar grains, and extends to the central cavity. The outer edge is ordinarily located approximately one-third of the distance from the outer edge of the large columnar grains to their inner termini.

4. Macroscopic Central Cavity - A large central cavity at the thermal center of the fuel is surrounded by columnar grains. This cavity contains a spongy deposit of UO_2 when center melting has occurred.
5. Large Spherical Voids in Sponge-Like UO_2 - Large spherical voids, as much as 0.3 mm in diameter are observed in the sponge-like UO_2 that is found in the partially-filled central cavity in fuel elements in which center melting has occurred. Temperature, irradiation exposure, stress and fuel composition probably influence the formation and movement of the voids. How they affect physical properties of the fuel, thermal conductivity, and fission gas retention may be better understood when more specimens still being irradiated are examined.

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FIGURE 5.22

The Relationship of Voids to UO_2 Microstructures
in Irradiated Fuel Cores

DATE

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7/15/94

END