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EFFECTS OF IRRADIATION
ON
HYDRIDED ZIRCONIUM-URANIUM ALLOY
NAA 120-4 EXPERIMENT
(Title Unclassified)

AEC Research and Development Report

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SNAP REACTOR —
SNAP PROGRAM
M-3679 (63rd Ed.)

EFFECTS OF IRRADIATION
ON
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NAA 120-4 EXPERIMENT
(Title Unclassified)

By:
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R. E. FORRESTER

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ABSTRACT

In the NAA 120-4 Experiment, 17 fuel specimens were irradiated simultaneously in a single facility at the Hanford K-East Reactor, to a target burnup of ~ 0.9 metal at.%. The experiment was statistically designed to evaluate the performance of SNAP fuel materials as a function of fuel composition and temperature. The specimens were contained in five separate experimental units, which provided a means of controlling the maximum fuel temperatures at different levels, ranging from 1300 to 1600°F. Design compositions included 1.70 and 1.85 H/Zr ratios. The maximum temperature of the Hastelloy X cladding and the hydrogen barrier (SCB-1 ceramic) did not exceed 1300°F for any extended periods.

Burnup analysis showed that the NAA 120-4 Experiment achieved a burnup within 10% of its target requirements.

Postirradiation examination showed that the cladding was intact on all 17 fuel specimens. Disassembly of the fuel specimens showed cracking had occurred in 8 of the 17 central fuel sleeves (test fuel specimens). Besides exhibiting a strong temperature dependence for fuel swelling, this experiment also confirmed the existence of the postulated "offset" fuel growth which saturates early in life.

This report describes the fuel element fabrication, experiment design, and hot laboratory examination. It does not include that portion of the irradiation that would disclose information on Hanford production.

I. INTRODUCTION AND BACKGROUND

The SNAP^{*} 8 fuel element concept is based on the use of a homogenous fuel-moderator system. A dispersion of metallic uranium in a zirconium hydride matrix results from hydriding Zr - 10 wt % U to $N_H = 6.0$ (6.0×10^{22} atoms H/cc fuel). Carbon is added to the alloy as a grain refiner. This alloy constitutes the fuel-moderator combination in the compact SNAP reactors. The fuel-moderator rods are clad in Hastelloy N, which has a continuous layer of ceramic on its inner surface to inhibit hydrogen permeation through the cladding at reactor operating temperatures.

Axial and radial temperature profiles exist across the fuel element in an operating SNAP reactor. The axial profile results in operation of the higher-temperature zone of the element, near the reactor outlet, in the face-centered-cubic δ -hydride phase region. The lower-temperature end of the element, by reason of the temperature gradient, operates in the face-centered-tetragonal ϵ -hydride phase region. The radial gradient superimposes a segregation of δ phase along the centerline, and ϵ to the surface in the boundary region. The compositional region of interest for a SNAP 8 reactor is presented in the zirconium-hydrogen phase diagram, Figure 1. Excessive loss of hydrogen, and entry into the $\beta + \delta$ phase regions, is considered an undesirable condition.^(1,2)

At the initiation of the SNAP 8 Fuel Element Irradiation Program, dimensional stability of the fuel-moderator was considered of key importance, based on results of a series of earlier irradiation tests.^(3,4)

Part of the program involved full-scale and sublength prototype irradiations, to determine the performance capabilities of the fuel element configuration as a function of temperatures, burnup, burnup rate, and fuel material composition and structure. The three sublength prototype series^(3,5,6) have been completed. The initial full-scale experiment, NAA 82-1⁽⁷⁾ (actually a SNAP 2 experiment, but with applicable results), has been completed; the NAA 121 series of full-scale fuel elements has been irradiated, and is undergoing postirradiation examination.

*SNAP designates a System for Nuclear Auxiliary Power. The SNAP 8 reactor is rated at 600 kwt for 10,000 hr.

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Figure 1. Zirconium-Hydrogen Phase Diagram
(Figure Unclassified)

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The basic objective of the NAA 120-4, reported here, was to obtain a better understanding of the behavior of hydrided uranium-zirconium alloy fuel during irradiation. Unlike the sublength and full-scale fuel element tests, the NAA-120-4 was designed to operate at constant fuel temperatures (1300, 1500, and 1600°F), rather than at constant cladding temperatures. Specimens having H/Zr ratios of 1.70 and 1.85 were tested at each temperature. Target burnup for the experiment was 0.9 metal at. %. Five of the 17 fuel specimens which were irradiated contained in-fuel thermocouples. These thermocouples were included for temperature control and for measurement of fuel-to-cladding ΔT , from which calculations of fuel swelling were made. The specimen test parameters are shown in Table 1.

TABLE 1
NAA 120-4 TEST PARAMETERS

Fuel Specimens	H/Zr	Irradiation Temperature (°F)
E-1*	1.85	1300
E-2	1.70	1300
E-3	1.70	1300
E-4	1.85	1300
E-5	1.85	1300
H-1*	1.70	1500
H-2	1.70	1500
H-3	1.70	1500
H-4	1.85	1500
H-5	1.85	1500
M-1*	1.70	1600
M-2	1.70	1600
M-3	1.70	1600
M-4	1.85	1600
M-5	1.85	1600
V-1*	1.70	1500
V-2*	1.70	1500

*Specimens equipped with in-fuel thermocouples and hydrogen pressure sensors.

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II. EXPERIMENT DESIGN

A. MECHANICAL

The NAA 120-4 Experiment consisted of five separate subcapsules, designed to operate at three different fuel centerline temperatures. Three of the subcapsules contained five fuel specimen assemblies each, and the other two subcapsules [the phase change experiments⁽⁸⁾] each contained one fuel specimen assembly. Each fuel specimen assembly was instrumented with three cladding thermocouples. In addition, one specimen in each subcapsule was instrumented with one in-fuel thermocouple and a hydrogen pressure probe.

The phase change fuel specimen assemblies were designed so that in-fuel temperature and fuel dissociation pressure could be continuously monitored, and hydrogen could be added or removed from the fuel during operation.

The fuel specimen assemblies were centered in 0.710-in. ID stainless steel fuel sleeves. The entire void space surrounding the fuel specimen assemblies was filled with NaK-78 containing a maximum of 14 ppm oxygen. A bellows was used to compensate for NaK expansion and contraction. The bellows was contained in a bellows compartment which was designed so that, even if the bellows failed, each fuel specimen assembly would still be submerged in NaK.

The NaK chamber of each capsule was surrounded by a precisely machined gas annulus. The temperatures of the fuel specimens were controlled by varying the mixture of helium and argon in this annulus. The gas annulus, formed by the OD of the NaK compartment and the ID of the outer shell, was essentially the same for all of the subcapsules - 0.008 in. in width. The outer shell of each subcapsule was a 1.0-in. OD, 0.065-in. wall, Type 304 stainless steel tube. Each of the five subcapsules had a separate gas temperature control system. Fuel centerline temperature of the instrumented specimen was used for control of each capsule.

The experiment was instrumented with 62 thermocouples, 5 pressure sensors, and 4 flux monitors. A total of 51 Chromel-Alumel thermocouples were banded to the fuel specimen cladding tubes. The remainder of the thermocouples were Chromel-Alumel, and were used to monitor reactor coolant temperature and peripheral experiment components. The flux monitors used in the experiment were Reuter-Stokes, Type RSN-202, Neutron Detectors, and were located

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as shown in Figure 2. The pressures associated with E-1, H-1, and M-1 specimens were monitored by null-balance sensors located near the specimen. The pressures of the V-1 and V-2 specimens were monitored, using direct communication lines and ex-reactor strain gage transducers. A neutron dosimeter was banded to each fuel specimen. The dosimeters consisted of Fe, Ni, and Co wires in a stainless steel tube with both ends brazed closed.

B. NUCLEAR

The 2A Test Facility in the HAPO KE Reactor was chosen for the NAA 120-4 Experiment, because it fulfilled the requirements of a high thermal neutron flux that was reasonably flat over a distance that exceeded the length of the experiment. A shroud tube was required, in order to obtain the specified heat generation rate in the fuel during irradiation. The shroud tube design criteria were determined from: (1) lattice cell calculations that provided relative spatial thermal neutron flux distributions, and (2) the measured unperturbed thermal neutron flux in the test facility.⁽⁹⁾ The cell calculations were performed,⁽¹⁰⁾ employing a 30 thermal group transport theory code⁽¹¹⁾ using cylindrical geometry.

C. THERMAL AND HYDRAULIC

Fuel temperatures in the experiment were controlled by varying the amounts of helium and argon in the gas annuli (helium has good thermal conduction, argon poor conduction). The design of the temperature control annulus was accomplished⁽¹²⁾ by using the one-dimensional heat transfer code, GASP.⁽¹³⁾ The nominal gas composition was fixed at 50% He - 50% A, allowing a good temperature control range.

The axial temperature profile of the fuel specimens would assume an approximate cosine shape, if the axial heat losses were not compensated for. The temperature profile was markedly improved (made essentially isothermal) by the use of heat barriers (enlarged gas annuli) at both ends of the fuel specimens, and by the use of 20 wt % fully enriched uranium end sleeves. The optimum location of these barriers was determined by an iterative process, in which the thickness and location of the gas annulus step and the length and thickness of the 20 wt % end sleeves were varied. The heat barrier optimization analyses⁽¹²⁾ were performed in an r-Z geometry, using the computer code HEATING.⁽¹⁴⁾

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Figure 2. NAA 120-4 Flux Monitor Positions
(Figure Unclassified)

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Prior to insertion of the experiment into the KE Reactor, the assembly was flow tested in a hydraulic loop at Hanford. The hydraulic tests consisted of: (1) overall experiment pressure drop, (2) capsule pressure drop, and (3) flow-split inside and outside the shroud tube. Test results showed an overall experiment pressure drop of 14 psi. The flow split on Capsules 1, 2, and 3 was 30% inside and 70% outside of the shroud tube. The corresponding ratios for Capsules 4 and 5 were 39 and 61%.

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a. Typical Noninstrumented Fuel Specimen

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b. Typical Instrumented Fuel Specimen

Figure 3. Fuel Specimen Configuration
(Figure Unclassified)

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III. FUEL ELEMENT FABRICATION

The NAA 120-4 Experiment contained 17 fuel specimen assemblies. Each assembly consisted of five fuel sleeves containing fully enriched U-ZrH_x mounted on an "inert" support rod of normal U-ZrH_x , as shown in Figure 3. The end fuel sleeves contained 20 wt % of 93% enriched U-ZrH_x , while the intermediate and center sleeves contained 10 wt % fully enriched U-ZrH_x . The end and intermediate sleeves, as well as the support rod, were considered as "throw-away" pieces, merely included to provide the proper conditions for the test specimen (center fuel sleeve).

The specimen assemblies were centered in the cladding by the use of Hastelloy X fuel supports. Compressible 10% dense Type 430 stainless steel Felt-metal* was placed between the end fuel sleeves and the fuel supports, to prevent the fuel sleeves from separating. The specimens were then clad in Hastelloy X which had been coated with a ceramic hydrogen-permeation-barrier material (SCB-1).

A. CLADDING

The NAA 120-4 fuel specimen assembly cladding tube was nominally 0.480-in. OD by 0.030-in. wall, Hastelloy X-285. The end closures consisted of Hastelloy X machined from bar stock. The lower end closure consisted of an end cap and inner cup, E.B.† welded together to form a vacuum thermal barrier. The instrumented assemblies contained a Hastelloy X thermocouple well and a Type 304 stainless steel pressure tap, E.B. welded to the upper end cap. The fuel supports were also machined from Hastelloy X, and had a thin coating of tungsten vapor-deposited on them, to prevent adherence of the Hastelloy X to the SCB-1 ceramic coating. In a test, prior to assembly of the experiment, the tungsten was found to be compatible with the SCB-1 ceramic coating.⁽¹⁵⁾

B. HYDROGEN BARRIER COATING

The cladding tubes, end closures, and the outside of the thermocouple wells were coated with SCB-1 ceramic hydrogen barrier. The composition of the ceramic

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†Electron Beam

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is given in Table 2. The ceramic was applied as an aqueous slip, air dried, and fired for 5 min at 2125°F. The lower end cap assembly, which comprised the final specimen assembly closure, was coated with SCB-1 ceramic prior to welding into the tube assembly. After the lower end cap assembly had been welded into the assembly, a fusion cycle was performed. The purpose of the fusion cycle was to form a ceramic-to-ceramic bond between the lower end cap assembly and the tube assembly. The fusion bond was formed by induction heating the lower end of the assembly for 1-1/2 min at 1950°F in a vacuum. Nominal ceramic layer thickness on the cladding tubes ranged from 1.5 to 2.0 mils.⁽¹⁶⁾

TABLE 2
COMPOSITION OF SCB-1 CERAMIC
HYDROGEN BARRIER

<u>Frit</u>	
Material	wt %
SiO ₂	45.2 ± 0.18
TiO ₂	12.6 ± 0.05
Al ₂ O ₃	7.2 ± 0.03
BaO ₂	35.9 ± 0.14
<u>Slip</u>	
Material	Weight (gm)
SCB-1 (Frit)	485.0 ± 1.95
Ferro Black Label Clay	15.0 ± 0.06
Chrome Oxide (Cr ₂ O ₃)	5.0 ± 0.05
Bentonite	1.25 ± 0.05
Distilled Water	350.0 ± 1.40

C. FUEL RODS

Fuel material was prepared by melting, extruding, and hydriding. The melt stock consisted of reactor grade zirconium sponge, reactor grade uranium, and high-purity zirconium carbide powder. Alloys were prepared by arc-melting under a vacuum of ~1μ.

Extrusion was carried out at 1000°F, using a reduction in area of ~10 to 1. The fuel was then rough machined into the various required configurations.

Hydriding was accomplished by placing the fuel in an evacuated retort held at 1650°F, and slowly introducing a known volume of hydrogen gas into the retort.⁽¹⁷⁾ Following hydriding, the sleeves and support rods were machined to the required dimensions. Tables 3 and 4 present compositions of the various finished specimen assembly parts.

TABLE 3
CHEMICAL COMPOSITION OF ZIRCONIUM-URANIUM
HYDRIDE FUEL SPECIMENS
(Center Fuel Sleeves)

Fuel Specimen	Fuel Rod Number	Zirconium (wt %)	Uranium		Carbon (wt %)	Hydrogen	
			Total (wt %)	Enrichment (%)		(wt %)	H/Zr
E-1	707-D-1-B	Balance	10.52	93.05	0.13	1.78	1.85
E-2	692-B-2-B	Balance	10.08	93.12	0.14	1.66	1.72
E-3	692-B-1-A	Balance	10.08	93.12	0.14	1.65	1.72
E-4	692-A-1-B	Balance	10.10	93.13	0.14	1.78	1.86
E-5	707-D-1-C	Balance	10.52	93.05	0.13	1.78	1.86
H-1	692-C-1-C	Balance	10.05	93.14	0.12	1.65	1.70
H-2	707-C-3-C	Balance	10.56	93.05	0.12	1.65	1.71
H-3	707-D-2-C	Balance	10.52	93.05	0.13	1.63	1.70
H-4	692-D-1-B	Balance	9.98	93.11	0.15	1.79	1.85
H-5	692-D-1-C	Balance	9.98	93.11	0.15	1.78	1.85
M-1	707-D-2-B	Balance	10.52	93.05	0.13	1.64	1.71
M-2	692-B-2-A	Balance	10.08	93.12	0.14	1.65	1.72
M-3	692-C-2-A	Balance	10.05	93.14	0.12	1.64	1.70
M-4	692-C-3-A	Balance	10.05	93.14	0.12	1.78	1.85
M-5	692-D-1-A	Balance	9.98	93.11	0.15	1.77	1.84
V-1	707-D-2-A	Balance	10.52	93.05	0.13	1.64	1.71
V-2	692-B-2-C	Balance	10.08	93.12	0.14	1.65	1.71

Dimensional measurements, taken on the support rods and fuel sleeves prior to assembly, are listed in Table 5. Figure 4 shows a typical fuel assembly, prior to insertion into the cladding tube assembly.

TABLE 4

CHEMICAL COMPOSITIONS OF ZIRCONIUM-URANIUM HYDRIDE
SUPPORT RODS AND INTERMEDIATE AND END FUEL SLEEVES

Description	Fuel Rod Number	Zirconium (wt %)	Uranium		Carbon (wt %)	Hydrogen	
			Total (wt %)	Enrichment (%)		(wt %)	H/Zr
E-1							
Support Rod	392-A-2-A	Balance	10.62	Normal	0.12	1.78	1.85
Intermediate Sleeve	707-D-1-B	Balance	10.52	93.05	0.13	1.78	1.85
End Sleeve	630-A-5	Balance	18.77	92.5	0.09	1.62	1.85
E-2							
Support Rod	391-C-3-B	Balance	10.64	Normal	0.17	1.62	1.69
Intermediate Sleeve	692-B-2-B	Balance	10.08	93.12	0.14	1.66	1.72
End Sleeve	630-A-7-A	Balance	18.50	92.5	0.11	1.49	1.70
E-3							
Support Rod	391-A-3-C	Balance	10.63	Normal	0.16	1.64	1.71
Intermediate Sleeve	692-B-1-A	Balance	10.08	93.12	0.14	1.65	1.72
End Sleeve	630-A-3-B	Balance	18.30	92.5	0.11	1.50	1.71
E-4							
Support Rod	391-D-2-B	Balance	10.64	Normal	0.17	1.76	1.85
Intermediate Sleeve	692-A-1-B	Balance	10.10	93.13	0.14	1.78	1.86
End Sleeve	630-A-8	Balance	18.55	92.5	0.11	1.49	1.70
E-5							
Support Rod	392-A-3-B	Balance	10.62	Normal	0.14	1.78	1.85
Intermediate Sleeve	707-D-1-C	Balance	10.52	93.05	0.13	1.78	1.86
End Sleeve	630-A-8	Balance	18.55	92.5	0.11	1.49	1.70
H-1							
Support Rod	391-C-2-A	Balance	10.64	Normal	0.17	1.62	1.70
Intermediate Sleeve	692-C-1-C	Balance	10.05	93.14	0.12	1.65	1.70
End Sleeve	630-A-4-B	Balance	18.40	92.5	0.08	1.50	1.70
H-2							
Support Rod	391-C-3-A	Balance	10.64	Normal	0.17	1.62	1.70
Intermediate Sleeve	707-C-3-C	Balance	10.56	93.05	0.12	1.65	1.71
End Sleeve	630-A-4	Balance	18.40	92.5	0.08	1.50	1.70
H-3							
Support Rod	264-D-4-A	Balance	10.46	Normal	0.13	1.65	1.71
Intermediate Sleeve	707-D-2-C	Balance	10.52	93.05	0.13	1.63	1.70
End Sleeve	630-A-6-A	Balance	18.37	92.5	0.09	1.50	1.71

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TABLE 4 (Continued)

Description	Fuel Rod Number	Zirconium (wt %)	Uranium		Carbon (wt %)	Hydrogen	
			Total (wt %)	Enrichment (%)		(wt %)	H/ Zr
H-4							
Support Rod	392-A-4-A	Balance	10.62	Normal	0.12	1.78	1.85
Intermediate Sleeve	692-D-1-B	Balance	9.98	93.11	0.15	1.79	1.85
End Sleeve	630-A-6-A	Balance	17.45	92.5	0.08	1.65	1.85
H-5							
Support Rod	232-B-1-A	Balance	10.14	Normal	0.18	1.75	1.83
Intermediate Sleeve	692-D-1-C	Balance	9.98	93.11	0.15	1.78	1.85
End Sleeve	630-A-9-B	Balance	17.45	92.5	0.08	1.65	1.85
M-1							
Support Rod	391-C-3-C	Balance	10.64	Normal	0.17	1.62	1.70
Intermediate Sleeve	707-D-2-B	Balance	10.52	93.05	0.13	1.64	1.71
End Sleeve	630-A-6-C	Balance	18.37	92.5	0.09	1.50	1.71
M-2							
Support Rod	391-B-3-B	Balance	10.65	Normal	0.15	1.64	1.71
Intermediate Sleeve	692-B-2-A	Balance	10.08	93.12	0.14	1.65	1.72
End Sleeve	630-A-6-A	Balance	18.37	92.5	0.09	1.50	1.71
M-3							
Support Rod	310-A-1-B	Balance	10.46	Normal	0.18	1.64	1.71
Intermediate Sleeve	692-C-2-A	Balance	10.05	93.14	0.12	1.64	1.70
End Sleeve	630-A-6	Balance	18.37	92.5	0.09	1.50	1.71
M-4							
Support Rod	392-A-4-B	Balance	10.62	Normal	0.14	1.78	1.86
Intermediate Sleeve	692-C-3-A	Balance	10.05	93.14	0.12	1.78	1.85
End Sleeve	630-A-5	Balance	18.77	92.5	0.09	1.62	1.85
M-5							
Support Rod	392-A-1-B	Balance	10.62	Normal	0.15	1.77	1.85
Intermediate Sleeve	692-D-1-A	Balance	9.98	93.11	0.15	1.77	1.84
End Sleeve	630-A-8	Balance	18.55	92.5	0.11	1.50	1.70
V-1							
Support Rod	391-B-4-A	Balance	10.65	Normal	0.15	1.64	1.72
Intermediate Sleeve	707-D-2-A	Balance	10.52	93.05	0.13	1.64	1.71
End Sleeve	630-A-6	Balance	18.37	92.5	0.09	1.50	1.71
V-2							
Support Rod	391-B-4-B	Balance	10.65	Normal	0.15	1.64	1.71
Intermediate Sleeve	692-B-2-C	Balance	10.08	93.12	0.14	1.65	1.71
End Sleeve	630-A-3	Balance	18.30	92.5	0.11	1.50	1.71

Figure 4. Typical NAA 120-4 Fuel Specimen Assembly
(Figure Unclassified)

Sub-capsule Number	Specimen Number	Support Rod (in.)			End Fuel Sleeve (in.)			Intermediate Fuel Sleeve (in.)			Center Fuel Sleeve (in.)			Intermediate Fuel Sleeve (in.)			End Fuel Sleeve (in.)			Fuel-Cladding As-Assembled Clearance (mils)	
		Length	OD	ID	Length	OD	ID	Length	OD	ID	Length	OD	ID	Length	OD	ID	Length	OD	ID	Radial	Axial
1	E-1	3.747	0.2502	0.098	0.1987	0.3352	0.2502	0.1991	0.4092	0.2503	2.251	0.4093	0.2506	0.1993	0.4093	0.2503	0.1993	0.3369	0.2501	4.2 ± 1.4	80
	E-2	3.753	0.2501	-	0.1995	0.3319	0.2501	0.1985	0.4091	0.2501	2.250	0.4092	0.2504	0.1986	0.4090	0.2500	0.1988	0.3354	0.2500	4.2 ± 0.8	50
	E-3	3.752	0.2501	-	0.1914	0.3359	0.2500	0.1988	0.4090	0.2502	2.250	0.4090	0.2503	0.1988	0.4088	0.2501	0.1988	0.3357	0.2499	4.4 ± 1.3	90
	E-4	3.752	0.2502	-	0.1988	0.3363	0.2501	0.1998	0.4087	0.2502	2.249	0.4093	0.2503	0.1998	0.4092	0.2503	0.1998	0.3363	0.2503	4.1 ± 1.2	37
	E-5	3.754	0.2503	-	0.1995	0.3373	0.2508	0.1999	0.4090	0.2502	2.250	0.4091	0.2505	0.2003	0.4092	0.2503	0.1993	0.3350	0.2502	4.2 ± 0.8	88
2	H-1	3.755	0.2500	0.098	0.1991	0.3312	0.2499	0.198	0.4011	0.2500	2.250	0.4011	0.2500	0.198	0.4012	0.2500	0.1986	0.3310	0.2499	8.1 ± 1.5	129
	H-2	3.752	0.2502	-	0.1979	0.3303	0.2502	0.1992	0.4010	0.2500	2.250	0.4007	0.2502	0.1993	0.4009	0.2502	0.1923	0.3303	0.2501	8.2 ± 1.6	44
	H-3	3.752	0.2501	-	0.2083	0.3307	0.2505	0.2002	0.4010	0.2500	2.249	0.4011	0.2501	0.2003	0.4011	0.2502	0.1998	0.3307	0.2499	8.4 ± 1.3	75
	H-4	3.752	0.2502	-	0.1991	0.3305	0.2501	0.1992	0.4011	0.2502	2.248	0.4010	0.2505	0.1991	0.4011	0.2502	0.1998	0.3306	0.2501	8.4 ± 1.3	54
	H-5	3.752	0.2502	-	0.1991	0.3302	0.2500	0.1992	0.4011	0.2501	2.254	0.4011	0.2503	0.1978	0.4009	0.2501	0.1972	0.3303	0.2501	8.4 ± 1.6	28
3	M-1	3.750	0.2501	0.098	0.1991	0.3311	0.2499	0.2013	0.3920	0.2499	2.251	0.3921	0.2502	0.1994	0.3924	0.2499	0.1996	0.3307	0.2501	12.6 ± 0.8	110
	M-2	3.752	0.2501	-	0.1990	0.3303	0.2499	0.1993	0.3920	0.2502	2.249	0.3921	0.2503	0.1991	0.3921	0.2501	0.1993	0.3307	0.2499	12.9 ± 1.8	76
	M-3	3.752	0.2500	-	0.1998	0.3307	0.2499	0.1999	0.3917	0.2500	2.254	0.3921	0.2501	0.1996	0.3920	0.2501	0.1998	0.3307	0.2499	13.0 ± 1.9	93
	M-4	3.751	0.2503	-	0.1989	0.3305	0.2501	0.2001	0.3919	0.2501	2.251	0.3921	0.2503	0.1991	0.3922	0.2503	0.1991	0.3305	0.2501	12.8 ± 1.8	85
	M-5	3.751	0.2502	-	0.1962	0.3305	0.2508	0.1984	0.3922	0.2501	2.249	0.3922	0.2502	0.1994	0.3922	0.2501	0.1967	0.3302	0.2502	12.9 ± 2.4	43
4	V-1	3.754	0.2500	0.098	0.1994	0.3308	0.2499	0.1998	0.4014	0.2500	2.249	0.4012	0.2501	0.2002	0.4011	0.2499	0.1989	0.3309	0.2499	8.0 ± 1.2	89
5	V-2	3.751	0.2501	0.098	0.2003	0.3307	0.2499	0.1998	0.4007	0.2500	2.251	0.4011	0.2501	0.2002	0.4007	0.2500	0.1983	0.3307	0.2499	8.0 ± 1.1	76

Table 5. As-Built Dimensions of Test Specimen Support Rods and Fuel Sleeves

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D. FUEL SPECIMEN ASSEMBLY

Final fabrication of the fuel specimen assemblies consisted of: (1) insertion of the fuel assembly into the cladding tube assembly, (2) assembly and welding of the lower end cap into the tube, and (3) performance of the ceramic-to-ceramic fusion of the lower end cap to the tube.

Permeation measurements were made by placing the completed assembly in a system evacuated to 10^{-4} torr. A mass spectrometer was used to measure hydrogen loss rates at appropriate isothermal test temperatures. Table 6 shows the results of these tests. Those specimens which did not meet the acceptance criteria were subsequently permeation scanned, with the exception of Specimen E-1. This specimen, as well as the other instrumented specimens, had an uncoated pressure sensor tube which, under in-pile conditions, would operate at a much lower temperature, and hence lower permeation rate. When the calculated bare metal permeation loss through the probe was subtracted from the 1250°F isothermal permeation rate, the specimen was found to be acceptable.

Ten fuel specimen assemblies exhibited isothermal permeation rates that were higher than acceptable. These specimens were subjected to permeation scan tests. The scanning tests showed that the hydrogen permeating the center 4-1/4 in. of the specimen tube amounted to a maximum of 10% of the total. Since the ends were to operate at a lower temperature during in-pile testing, resulting in lower hydrogen loss rates, the specimens were considered acceptable.⁽¹⁸⁾ The results of the hydrogen permeation scanning tests are also shown in Table 6.

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TABLE 6
FUEL SPECIMEN PERMEATION RATES AND
SCANNING TEST RESULTS

Assembly Number	Hydrogen Permeation Rate [cc(STP)/hr]				Permeation Scan Tests (% hydrogen loss at)		
	Isothermal Test Temperature (°F)			Maximum Allowable 1250°F Rate	Blind End†	Center§	Closure End†
	1100	1200	1250				
E-1	0.19	0.61	1.6*	1.03	-	-	-
E-2	0.03	0.04	0.09	0.45	-	-	-
E-3	0.03	0.08	0.11	0.45	-	-	-
E-4	0.11	0.24	0.44	1.03	-	-	-
E-5	0.08	0.26	0.33	1.03	-	-	-
H-1	0.11	0.33	0.67	0.10	52	1	48
H-2	0.05	0.13	0.71	0.10	50	1	50
H-3	0.01	0.01	0.04	0.10	-	-	-
H-4	0.02	0.05	0.23	0.26	-	-	-
H-5	0.02	0.14	0.28	0.26	82	1	18
M-1	0.13	0.54	1.04	0.05	52	1	48
M-2	0.06	0.24	0.47	0.05	54	1	46
M-3	0.06	0.24	0.47	0.05	47	1	53
M-4	0.04	0.14	0.26	0.10	52	4	44
M-5	0.03	0.12	0.26	0.10	46	11	43
V-1	0.14	0.57	1.02	0.10	54	1	46
V-2	0.20	0.77	1.17	0.10	58	1	42

*E-1 considered acceptable, based on calculation of hydrogen loss attributable to ends and thermocouple well.

†Includes end-cap and 1/2-in. of cladding.

§Center 4-1/4 in. of cladding.

IV. OPERATION

As stated previously, the NAA 120-4 was irradiated in the 2A Test Facility in the HAPO KE Reactor. The temperatures of each of the five subassemblies were independently controlled, through the use of a binary (helium and argon) gas control system. A variable mixture of helium and argon was sent from ex-reactor gas supplies, through the control system, into annuli concentric with the outside of the fuel specimens. Each experiment subassembly was controlled, using the fuel centerline thermocouple in the instrumented fuel specimen. The fuel temperature was maintained at design temperature conditions, whenever possible, by altering the binary control gas mixture. Cladding temperatures were not maintained at any specified level, but were not allowed to exceed 1300°F for any long periods, due to cladding strength considerations.

The data recording and monitoring system⁽¹⁹⁾ provided a punched paper tape record of all the thermocouple readings, with a strip chart record on most of the thermocouples for direct readout purposes. Each of the thermocouples had high- and low-alarm limit input values, which provided an automatic helium purge of the capsule gas annulus to prevent over-temperature operation.

Startup and operation of the experiment proceeded as planned.⁽²⁰⁾ After ~1500 hr of experiment operation, Fuel Specimen M-1 could not be maintained at design operating conditions (1600°F fuel centerline temperature), even with pure argon in the gas control annulus. This was due to a decrease in its fuel-to-cladding ΔT , which had occurred much more rapidly than had been anticipated. The other specimens instrumented with in-fuel thermocouples showed similar decrease in fuel-to-cladding ΔT , but proportionally smaller, due to their lower design fuel temperatures. Power-normalized fuel-to-cladding ΔT 's as a function of time at temperature are shown in Figure 5.

After ~2100 hr at operating temperature, the instrumentation indicated that a significant amount of fission gas had been released into the fuel-to-cladding hydrogen gap of the 1500 and 1600°F fuel specimens (pressure rise and fuel-to-cladding ΔT increase). Similar indications were seen in the 1300°F fuel specimen (E-1) after ~6000 hr. In all cases, the fission gas was released while the experiment was at helium operating conditions, and was probably due to the 20 wt % end fuel sleeves changing phase.

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Figure 5. Power-Normalized Fuel-to-Cladding ΔT vs Time
(Average data)

[REDACTED]

Although the phase change experiments⁽⁸⁾ had been planned for Specimens V-1 and V-2, only Specimen V-2 was intentionally driven into β phase. This was due to a failure of a weld joint on Specimen V-1, and subsequent NaK in-leakage into the cladding, which plugged the pressure line to the specimen.

Time-averaged operating temperatures are listed in Table 7.⁽²¹⁾ During the final shutdown of the experiment, temperatures were decreased as rapidly as possible from design operating conditions, so that the hydrogen profile in the fuel would reflect the operating hydrogen profile.

Upon reaching a calculated burnup ranging from 0.82 to 1.03 metal at.%, the experiment was discharged from the reactor. At the time of discharge, all but two cladding thermocouples, and all in-fuel thermocouples, were in operating condition.

TABLE 7
TIME-AVERAGED OPERATING TEMPERATURES

Subcapsule Number	Fuel Specimen Number	Thermocouple Number	Temperature at Design Conditions (°F)		Temperature at Helium Conditions (°F)	
			Range	Average	Range	Average
1	E-1	{ 1	1275-1325	1290	1100-1250	1138
		{ 2	900-1050	965	550-700	650
		{ 3	925-1050	990	575-725	675
		{ 4	900-1025	975	550-675	625
	E-2	{ 5	900-125	975	550-675	625
		{ 6	875-1000	935	525-675	625
		{ 7	850-1025	960	550-650	610
	E-3	{ 8	875-975	925	525-675	620
		{ 9	850-1000	940	550-650	610
		{ 10	825-950	885	475-600	540
	E-4	{ 11	900-1000	960	550-650	610
		{ 12	900-1000	960	550-675	625
		{ 13	875-975	935	550-650	620
	E-5	{ 14	850-975	935	525-650	590
		{ 15	850-950	915	525-650	600
		{ 16	850-950	910	500-650	580
2	H-1	{ 17 [†]	1450-1525	1470	1375-1450	1415
		{ 18	800-1075	975	500-650	575
		{ 19	875-1100	1010	600-700	640
		{ 20	825-1100	1010	525-725	590
	H-2	{ 21	825-1050	990	500-650	575
		{ 22	950-1125	1040	525-675	570
		{ 23	950-1125	1040	575-700	650
	H-3	{ 24	900-1025	975	500-650	560
		{ 25	900-1125	1025	550-650	590
		{ 26	925-1050	990	500-625	560
	H-4	{ 27	875-1125	1010	550-675	600
		{ 28	850-1125	1040	550-625	610
		{ 29	825-1075	980	525-625	575
	H-5	{ 30	850-1100	1000	500-650	580
		{ 31	825-1050	935	500-625	575
		{ 32	850-1075	990	525-675	590
3	M-1	{ 33 [†]	1425-1600	1510	875-1425	1360 [§]
		{ 34	1075-1225	1175	500-600	910 [~]
		{ 35	1100-1225	1200	500-600	550
		{ 36	1100-1250	1220	475-600	540
	M-2	{ 37	1100-1250	1190	475-600	540
		{ 38	1100-1200	1160	450-575	520
		{ 39	1125-1250	1210	500-575	535
	M-3	{ 40	1100-1225	1180	500-600	550
		{ 41	1150-1275	1210	500-625	565
		{ 42	1100-1250	1190	500-575	530
	M-4	{ 43	1175-1300	1250	500-625	560
		{ 44	1150-1300	1240	500-625	560
		{ 45	1150-1250	1210	500-575	530
	M-5	{ 46	1150-1250	1210	475-575	525
		{ 47	1075-1200	1140	475-575	520
		{ 48	1150-1250	1200	525-600	560
4	V-1	{ 49 [†]	1450-1525	1490	1225-1400	1350 [§]
		{ 50	1025-1225	1160	500-600	535
		{ 51	1025-1225	1160	500-625	560
		{ 52	1075-1250	1180	500-625	560
5	V-2	{ 53 [†]	1450-1525	1490	1275-1375	1335
		{ 54	1025-1200	1130	475-600	540
		{ 55	1075-1225	1140	500-600	560
		{ 56	1050-1225	1140	475-600	540

Range is considered the temperature domain in which the thermocouple spent the majority of the time

[†]In-fuel thermocouple

[§]Early in life

[~]Late in life

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V. POSTIRRADIATION EXAMINATION

A. EXPERIMENT DISASSEMBLY AND EXAMINATION

All five subcapsules were visually examined, and appeared in excellent condition after the test.⁽²²⁾ The subcapsules were then cut open, and the fuel assemblies removed and cleaned. Examination of the fuel assemblies, using both Kollmorgen periscope and stereomicroscope, showed that all thermocouples, bands, dosimeters, and all related components were in excellent condition.

B. FUEL SPECIMEN ASSEMBLIES

Disassembly of the fuel element assemblies showed NaK weeping from the gas line in Specimen V-1. Subsequent examination showed two small pinholes in the gas line weld area at the upper end cap of the element, which permitted NaK to enter the element cladding.

Neutron radiography proved to be a successful method to examine irradiated fuel elements nondestructively. In most cases, it indicated the fuel specimen condition and a relative hydrogen distribution. In some cases, because of fuel condition, dimensional measurements of the fuel were deduced from the radiographs. This type of examination aided subsequent disassembly procedures and sample selection.

As can be seen in Figure 6, the neutron radiographs show that a good number of the support rods (normal U) were broken off, and remained inside the Hastelloy X fuel supports. No absolute value was assigned to the hydrogen content, although a relative hydrogen distribution within a specimen is clearly indicated, by comparison of the radiographs and the hydrogen analysis reported later in Table 12. The radiographs also indicated that the support rods inside the Hastelloy X fuel supports had a higher hydrogen content than did other portions of the fuel specimen (note the darker color in Figure 6).

A series of diameter and length measurements were made on each of the clad fuel elements, with dial indicators and vernier calipers. The accuracy of the dial indicators, when used remotely, is ± 0.001 in., while the accuracy of the vernier calipers is ± 0.0015 in. The results of these measurements are shown in Table 8. The length measurements showed no significant changes. The

Figure 6. Neutron Radiographs of the NAA 120-4 Experiment

TABLE 8

NAA 120-4 FUEL SPECIMEN CLADDING DIMENSIONAL MEASUREMENTS

Element Number	Average Pre-Irradiation Diameter Measurements (in.)	Average Postirradiation Diameter Measurements (in.)	Average ΔD (in.)	Average Pre-Irradiation Length Measurements (in.)	Average Postirradiation Length Measurements (in.)	Average ΔL (in.)
E-1	0.4813	0.4811	-0.0002	5.246	5.245	-0.001
E-2	0.4817	0.4815	-0.0002	5.243	5.242	-0.001
E-3	0.4817	0.4814	-0.0003	5.246	5.245	-0.001
E-4	0.4818	0.4831	+0.0013	5.246	5.247	+0.001
E-5	0.4813	0.4812	-0.001	5.247	5.246	-0.001
H-1	0.4816	0.4810	-0.0006	5.245	5.245	0.000
H-2	0.4813	0.4809	-0.0004	5.245	5.245	0.000
H-3	0.4813	0.4807	-0.0006	5.246	5.245	-0.001
H-4	0.4805	0.4803	-0.0002	5.246	5.244	-0.002
H-5	0.4809	0.4808	-0.0001	5.247	5.245	-0.002
M-1	0.4814	0.4817	+0.0003	5.250	5.249	-0.001
M-2	0.4813	0.4814	+0.0001	5.238	5.239	+0.001
M-3	0.4815	0.4823	+0.0008	5.246	5.248	+0.002
M-4	0.4812	0.4810	-0.0002	5.245	5.246	+0.001
M-5	0.4813	0.4810	-0.0003	5.245	5.246	+0.001
V-1	0.4812	0.4809	-0.0003	5.246	5.245	-0.001
V-2	0.4811	0.4812	+0.0001	5.236	5.236	0.000

cladding diameter measurements generally indicated little or no change between pre- and post-average diameters, although some postirradiation measurements did indicate fuel-cladding interference. Also, the postirradiation diameter cladding measurements of H-2, H-3, H-4, and H-5 showed smaller diameters at the top half of the specimen than at the lower half. The difference amounted to ~ 0.001 in.

Examination after fuel specimen disassembly, summarized later in Table 14, showed cracking in Fuel Specimens E-4, H-2, H-5, M-1, M-2, M-3, V-1, and V-2.⁽²²⁾ The examination further showed that the 20 wt % end fuel sleeves were badly swollen, cracked, and/or broken, in all but one specimen assembly (M-4). Seven of the 17 specimen assemblies were damaged to some degree in the throw-away 10 wt % end sleeves. The damaged condition of the 20 wt %, and some of the 10 wt %, end sleeves was expected.

Nine of the support rods could not be removed from their fuel sleeves. All of the in-fuel thermocouples and their respective thermocouple wells were easily removed from the support rods, except for Specimen V-1. The in-fuel thermocouple and its container in V-1 snapped off just below the fuel support during handling, and appeared very brittle. The portion that was in the support rod remained stuck. Reaction was observed on all support rods under the 20 wt % fuel sleeves; and, in some cases, at the interface between the 10 and 20 wt % end fuel sleeves. The reaction on the support and sleeves is thought to be a result of a eutectic which formed between the Type 430 stainless steel Feltmetal and the 20 wt % end fuel sleeves. The temperature needed to form a eutectic between these two alloys is $\sim 1340^{\circ}\text{F}$, and the 20 wt % fuel sleeves operated above this temperature.

In most cases, the support rod was broken off in the fuel supports; and, in no case, could the support rod be removed from the fuel support. This is thought to be due to local diameter increases in the cold ends, caused by hydrogen redistribution.

Figures 7, 8, and 9 show photographs of representative fuel assemblies, taken during examination of the experiment. Specimen E-1 (Figure 7) is representative of specimens which were intact following the test. Specimens M-1 and E-4 (Figures 8 and 9) are representative of cracked and broken fuel assemblies,

a. Before Disassembly

120-156, -157

c. Inside Fuel Support

120-303

b. After Disassembly

120-296, -297, -298

d. Typical Area of Center Fuel Sleeve

120-302

Figure 7. Specimen E-1
(Figure Unclassified)

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120-238, -239

a. Before Disassembly

120-427, -428

b. After Disassembly

Figure 8. Specimen M-1

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120-175, -176

a. Before Disassembly

120-353, -354

b. After Disassembly

Figure 9. Specimen E-4

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respectively. The fuel sleeve surface was very rough on Specimens E-1, E-4, E-5, H-4, and H-5 [all of these specimens contained a high H/Zr (1.85), and operated at 1300 and 1500°F respectively].

C. FUEL MEASUREMENTS

1. Dimensional Measurements

The dimensional measurements were obtained with a profilometer measuring device and/or dial indicator, with backup measurements using Vernier calipers. In some cases, the dimensional measurements were deduced from neutron radiographs. Profilometer diameter measurements could only be obtained on the unbroken fuel specimens.

The tabulated postirradiation vs pre-irradiation dimensional measurements, and the hydrogen-loss-corrected postirradiation vs the pre-irradiation measurements, are shown in Table 9; included in the measurements are the diameters taken on the support rod (normal U), directly under the center fuel sleeve.

Hydrogen redistribution and loss during irradiation causes fuel shrinkage, leading to density, diameter, and length changes, in addition to those caused by radiation-induced swelling. The correction for density values was taken directly from the plot of density vs hydrogen content.⁽⁶⁾ The hydrogen analysis performed on the fuel specimens is described in a later section. Correction factors for the dimensions were calculated from the corrected densities in the following manner:

$$(\% \Delta D/D)_{\text{hydrogen correction}} = (\% \Delta \rho/\rho)_{\text{hydrogen correction}} / 2.5 \quad \dots (1)$$

$$(\% \Delta L/L)_{\text{hydrogen correction}} = (\% \Delta \rho/\rho)_{\text{hydrogen correction}} / 5.0 \quad \dots (2)$$

Addition of these corrections to the measured values yielded the corrected values shown in Table 10. The corrections were obtained from hydriding data that produced the ratio $(\% \Delta D/D)/(\% \Delta L/L) = 2.0$.⁽³⁾

2. Fuel Density Measurements

The fuel specimen densities were obtained, using the liquid displacement technique, with monobromobenzene as the liquid medium. The accuracy of

TABLE 9
FUEL DIMENSIONAL MEASUREMENTS

Test Specimen	Initial Diametral Gap (in)	Pre-Irradiation OD (in)	Diameter Increase (in)				Average % $\Delta D/D$ (hydrogen corrected)	Initial Axial Gap (in)	Length Increase (in)	% $\Delta L/L$ (hydrogen corrected)	Average Support Rod Diameter (in)			Support Rod Average % $\Delta D/D$ (hydrogen corrected)
			Top	Center	Bottom	Average ΔD (hydrogen corrected)					Pre-Irradiation	Post Irradiation	ΔD	
E 1 *	0 0084 $\pm$ 0 0028	0 4092	0 0081	0 0081	0 0076	0 0083	2 03	0 080	0 051	2 27	0 2502	0 2480	-0 0022	0 8
E-2 *	0 0084 $\pm$ 0 0016	0 4092	0 0065	0 0069	0 0068	0 0069	1 69	0 050	0 031	1 38	0 2501	0 2516	0 0015	1 6
E 3 *	0 0088 $\pm$ 0 0026	0 4091	0 0066	0 0063	-	0 0065	1 59	0 090	0 025	1 11	0 2501	0 2518	0 0017	1 9
E-4	0 0082 $\pm$ 0 0024	0 4093	-	-	-	0 0097 [†]	2 37	0 037	0 001	0 4	0 2502	0 2460	-0 0042	4 0
E-5 *	0 0084 $\pm$ 0 0016	0 4091	0 0073	0 0074	0 0072	0 0073	1 78	0 088	0 053	2 36	0 2503	0 2504	0 0001	0 2
H-1 *	0 0162 $\pm$ 0 0030	0 4011	0 0121	0 0120	0 0120	0 0121	3 02	0 129	0 055	2 44	0 2500	0 2504	0 0004	2 0
H-2	0 0164 $\pm$ 0 0032	0 4007	0 0162	0 0156	0 0165	0 0178	4 44	0 044	0 060	2 67	0 2502	0 2485	-0 0017	0 6
H-3 *	0 0168 $\pm$ 0 0026	0 4011	0 0138	0 0132	0 0134	0 0135	3 37	0 075	0 064	2 85	0 2501	0 2504	0 0003	0 7
H-4 *	0 0168 $\pm$ 0 0026	0 4011	0 0106	0 0106	0 0104	0 0109	2 72	0 054	0 025	1 11	0 2502	0 2515	0 0013	2 0
H-5	0 0168 $\pm$ 0 0032	0 4011	0 0117	0 0126	0 0146	0 0131	3 27	0 028	0 001	0 04	0 2502	0 2480	-0 0022	1 8
M-1	0 0252 $\pm$ 0 0016	0 3921	0 0267	0 0260	0 0264	0 0284	7 24	0 110	0 099 [§]	4 40	0 2501	0 2450	-0 0051	1 2
M-2	0 0258 $\pm$ 0 0036	0 3921	0 0270	0 0262	0 0250	0 0293	7 47	0 076	0 116 [§]	5 16	0 2501	0 2480	-0 0021	2 3
M-3	0 0260 $\pm$ 0 0038	0 3921	-	-	-	0 0283 [†]	7 22	0 093	0 115 [§]	5 10	0 2500	No meas	-	-
M-4 *	0 0256 $\pm$ 0 0036	0 3921	0 0194	0 0202	0 0205	0 0206	5 25	0 085	0 083	3 69	0 2503	0 2515	0 0012	2 7
M-5 *	0 0258 $\pm$ 0 0048	0 3923	0 0211	0 0203	0 0193	0 0211	5 38	0 043	0 100	4 45	0 2502	0 2525	0 0023	3 1
V-1	0 0160 $\pm$ 0 0024	0 4012	0 0080	0 0102	0 0070	0 0117	2 92	0 089	0 051 [§]	2 27	0 2500	0 2490	-0 0010	1 5
V-2	0 0160 $\pm$ 0 0022	0 4011	-	-	-	0 0175 [†]	4 36	0 076	0 084 [§]	3 73	0 2501	0 2470	0 0031	2 7

*Intact specimen

[†]Deduced from peak cladding measurements

[§]Deduced from neutron radiographs

TABLE 10
HYDROGEN-CORRECTED FUEL DENSITY MEASUREMENTS

Specimen Number	Center Fuel Sleeve										Support Rod				
	Pre- ρ_1 (gm/cc)	Post- ρ_2 (gm/cc)			$\Delta\rho$ (gm/cc)			Volume Change ($\Delta\rho/\rho_2$) (%)			Pre- ρ_1 (gm/cc)	Post- ρ_2 (gm/cc)	$\Delta\rho$ (gm/cc)	Change ($\Delta\rho/\rho_1$) (%)	Volume Change ($\Delta\rho/\rho_2$) (%)
		Top*	Bulk	Bottom	Top	Bulk	Bottom	Top	Bulk	Bottom					
E-1	6.024		5.607			0.417			7.44		6.059	6.019	0.040	0.66	0.66
E-2	6.048	5.802	5.777	5.759	0.246	0.271	0.289	4.23	4.69	5.01	6.077	5.990	0.087	1.43	1.45
E-3	6.046		5.801			0.245			4.22		6.080	5.974	0.106	1.74	1.77
E-4	6.028		5.720			0.308			5.38		6.062	6.057	0.005	0.08	0.08
E-5	6.052	5.679	5.674	5.734	0.373	0.378	0.318	6.56	6.66	5.54	6.060	6.046	0.014	0.23	0.23
H-1	6.047		5.519			0.528			9.57		6.076	6.009	0.067	1.10	1.11
H-2	6.069		5.359			0.710			13.25		6.078	6.058	0.020	0.33	0.33
H-3	6.071	5.564	5.516	5.555	0.507	0.555	0.516	9.11	10.06	9.28	6.083	6.069	0.014	0.23	0.23
H-4	6.050	5.608	5.581	5.601	0.442	0.469	0.449	7.88	8.40	8.01	6.057	6.019	0.038	0.63	0.63
H-5	6.052	5.570	5.577	5.627	0.482	0.475	0.426	8.65	8.52	8.56	6.048	5.994	0.054	0.89	0.90
M-1	6.070		4.751			1.319			27.76		6.078	6.042	0.036	0.59	0.60
M-2	6.045		4.902			1.143			23.32		6.077	5.963	0.114	1.88	1.91
M-3	6.044		4.812			1.232			25.60		6.070	5.838	0.232	3.82	3.97
M-4	6.025	5.337	5.300	5.299	0.688	0.725	0.726	12.89	13.68	13.70	6.057	5.949	0.108	1.78	1.82
M-5	6.022	5.170	5.178	5.221	0.852	0.844	0.801	16.47	16.30	15.34	6.057	5.895	0.162	2.67	2.75
V-1	6.060		5.682			0.378			6.65		6.076	6.012	0.064	1.05	1.06
V-2	6.047		5.170			0.877			16.96		6.074	6.074	0.000	0.00	0.00

*"Top" indicates end of specimen toward capsule lead tube.

TABLE 11
NAA 120-4 BURNUP ANALYSIS

Element Number	Burnup (metal at. %) Measured By			Gamma Scan Results (Zr ⁹⁵ Photopeak)		
	Mass Spectrometry	Dosimeter	Thermal Calculation	Peak Gamma Activity (counts/5 sec)	Average Gamma Activity (counts/4 sec)	Peak-to-Average Activity Ratio
E-1	0.83	0.87	1.03	5484	5324	1.03
E-2	0.82	0.81	1.03	5522	5321	1.04
E-3	0.83	0.79	1.03	5379	5256	1.02
E-4	0.86	0.85	1.03	6144	5865	1.05 [†]
E-5	0.85	*	1.03	6240	6063	1.03
H-1	0.83	0.75	0.96	5692	5403	1.05
H-2	0.86	0.69	0.96	5551	5405	1.03 [†]
H-3	0.84	*	0.96	5414	5191	1.04
H-4	0.87	0.89	0.96	5938	5761	1.03
H-5	0.89	0.93	0.96	6553	6196	1.06 [†]
M-1	0.82	0.52	0.90	3188	3021	1.06 [†]
M-2	0.81	0.69	0.90	3278	2978	1.10 [†]
M-3	0.82	0.71	0.90	3483	3157	1.10 [†]
M-4	0.85	0.72	0.90	3305	3204	1.03
M-5	0.85	0.74	0.90	3349	3238	1.03
V-1	0.82	1.00	0.89	3856	3756	1.04 [†]
V-2	0.85	0.82	0.82	3575	3423	1.04 [†]

*Data not available

†Fuel specimen badly cracked and broken

these measurements has been determined, from previous measurements, to be within 0.1% of the cited values. Table 10 shows the hydrogen-corrected density measurements. The hydrogen-corrected densities are based on the relationship:

$$(\Delta\rho/\rho)_{\text{swelling}} = (\Delta\rho/\rho)_{\text{measured}} + (\Delta\rho/\rho)_{\text{hydrogen correction}} \quad \dots(3)$$

The hydrogen correction was obtained from published curves of ρ vs H/Zr .⁽⁶⁾ The density measurements showed volume changes from 4.22 to 27.76%, after correcting for hydrogen loss.

3. Fuel Sampling

Each specimen was sectioned across the center fuel sleeve and support rod, to provide samples for burnup determinations (center fuel sleeve only), hydrogen analysis, and metallographic examination. All of the samples were taken at approximately the same location, as shown in the following sketch, except when the broken and/or cracked condition of the fuel dictated the sample location.

4. Fuel Burnup Measurements

A representative section of fuel from the middle portion of the center fuel sleeves was analyzed for uranium isotopic content by mass spectrometry. This analysis was used to calculate the metal atom percent burnup for the 17 fuel specimens from the NAA 120 Experiment. The dosimeter analysis for burnup was calculated from three foils (Ni, Fe, and Co) which formed a single dosimeter, banded to each of the 17 elements during the test.

The mass spectrographic and dosimeter burnup values are shown in Table 11, along with the value calculated from thermal data. As can be seen in Table 11,

TABLE 12
NAA 120-4 HYDROGEN ANALYSIS*

Specimen Number	Center Sleeve					Center 2-1/4 in. Support Rod				
	Pre-H/Zr	Post-H/Zr			Average H/Zr	Pre-H/Zr	Post-H/Zr			Average H/Zr
		Top	Center	Bottom			Top	Center	Bottom	
E-1	1.85	1.77 [†]	1.75 [†]	1.71 [†]	1.74	1.85	1.77 [†]	1.75 [†]	1.71 [†]	1.74
E-2	1.72	1.74	1.74	1.72	1.73	1.69	1.65	1.67	1.70	1.67
E-3	1.72	1.72	1.68	1.72	1.71	1.71	1.66	1.65	1.67	1.66
E-4	1.86	Sample lost	1.82	1.81	1.81	1.85	1.78	1.78	1.77	1.78
E-5	1.86	1.83	1.86	1.86	1.85	1.85	1.84	1.86	1.88	1.86
H-1	1.71	1.66	1.66	1.67	1.66	1.70	1.51	1.53	1.57	1.54
H-2	1.71	1.74	1.58	1.44	1.59	1.70	1.68	1.57	1.50	1.58
H-3	1.70	1.69	1.70	1.70	1.70	1.71	1.62	1.63	1.63	1.63
H-4	1.85	1.76	1.74	1.72	1.74	1.85	1.68	1.68	1.69	1.68
H-5	1.85	1.70	1.76	1.91 [†]	1.79	1.83	1.68	1.72	1.91 [†]	1.77
M-1	1.71	1.58 [†]	1.30 [†]	1.60	1.49	1.70	1.58 [†]	1.30 [†]	1.34	1.41
M-2	1.72	1.62	1.59	1.12 [†]	1.44	1.71	1.54	1.52	1.12 [†]	1.39
M-3	1.70	1.24	1.63 [†]	0.90 [†]	1.26	1.71	0.99	1.63 [†]	0.90 [†]	1.17
M-4	1.85	1.69	1.69	1.69	1.69	1.86	1.52	1.66	1.64	1.61
M-5	1.84	1.71	1.72	1.70	1.71	1.85	1.66	1.66	1.65	1.66
V-1	1.71	1.51 [†]	1.49	1.47 [†]	1.49	1.72	1.51 [†]	1.46	1.47 [†]	1.48
V-2	1.71	1.63 [†]	1.61 [†]	1.61 [†]	1.62	1.71	1.63 [†]	1.61 [†]	1.61 [†]	1.62

*Accuracy ± 0.02 H/Zr

[†]Support rod and fuel sleeve analyzed together

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the burnup analyses failed to agree with each other. In past experiments of this nature, the analysis by mass spectrometry was considered more reliable, because it has the least degree of uncertainty. The mass spectrometry data on past experiments would tend to follow the gamma activity profile (also shown in Table 11) within $\sim 10\%$. However, in the case of the mass spectrometry burnup for the NAA 120 Experiment, it was found, after normalizing the data, that correlation with the gamma activity existed only in the E and H elements. The reason for the disagreement in the data for the M and V elements is probably related to different thickness of the fuel sleeves and to fuel condition.

5. Hydrogen Measurements

Samples for hydrogen analysis were obtained as 1/8-in. thick wafers, across both the fuel sleeve and support rod. Three samples from each specimen were analyzed for hydrogen content, after being cleaned in ethyl alcohol and weighed.

The hydrogen analysis was performed at the Atomics International Hot Laboratory (AIHL), using a vacuum extraction analysis unit which involves induction heating the fuel samples in a vacuum at temperatures high enough to release the hydrogen ($\sim 2200^\circ\text{F}$). The released gas is collected in a calibrated volume, and the pressure of the hydrogen is measured. The accuracy associated with this type of measurement is considered to be ± 0.02 wt % of that measured.

A summary of the hydrogen analyses is presented in Table 12. Some of the specimens showed steep hydrogen gradients, over the length of the specimens; in general, most of the support rods showed a slightly lower hydrogen content than the fuel sleeves.

D. FUEL METALLOGRAPHIC EXAMINATION

Metallographic samples of the fuel were taken from each of the fuel specimens, to determine what structural changes had occurred.

The samples were mounted with Hysol C-9 epoxy resin, and were ground on 240-, 320-, and 600-grit silicon carbide paper, using water as a lubricant. Attack polishing was performed on the fuel samples, using rayon cloth with alternate application of a Linde-B slurry, a HNO_3 -HF- H_2O mixture, and a 10% oxalic acid solution. Electro-etching was performed in a 10% oxalic acid solution at 20 to 300-v dc. Many of the samples were reground, repolished, and etched again to confirm observations.

TABLE 13
FILAR EYE DIAMETER FUEL MEASUREMENTS vs PRE-IRRADIATION
FUEL MEASUREMENTS

Specimen Number	Fuel Sleeve Thickness (in.)			Gap Between Support Rod and Fuel Sleeve (in.)				Thermocouple Hole Diameter (in.)		
	Pre-	Post-	Δt	Average			Gap Ranges (mils)	Pre-	Post-	ΔD
				Pre- [†]	Post-	ΔG				
E-1	0.0793	0.0797	0.0004	0.0004	0.0069	0.0065	4.2-8.5	0.098	0.090	0.098
E-2	0.0793	0.0800	0.007	0.0003	0.0055	0.0052	4.4-6.9			
E-3	0.0795	0.0811	0.0016	0.0002	0.0041	0.0039	1.7-5.4			
E-4	0.0793	0.0836	0.0043	0.0001	0.0012	0.0011	0.5-1.2			
E-5	0.0793	0.0811	0.0018	0.0002	0.0047	0.0045	3.8-6.3			
H-1	0.0755	0.0762	0.0007	0.0000	0.0098	0.0096	4.1-16.6	0.098	0.103	0.005
H-2	0.0753	0.0755	0.0002	0.0000	*	*	*			
H-3	0.0755	0.0778	0.0023	0.0000	0.0085	0.0085	2.9-14.1			
H-4	0.0752	0.0770	0.0018	0.0003	0.0058	0.0055	4.2-8.5			
H-5	0.0754	0.0790	0.0036	0.0001	*	*	*			
M-1	0.0709	*	*	0.0002	*	*	*	*	*	*
M-2	0.0709	0.0772	0.0063	0.0002	0.0046	0.0044	3.5-7.2			
M-3	0.0710	0.0833	0.0123	0.0001	0.0022	0.0021	1.1-2.8			
M-4	0.0709	0.0719	0.0010	0.0000	0.0179	0.0178	11.9-28.5			
M-5	0.0709	0.0734	0.0025	0.0000	0.0145	0.0145	7.3-20.0			
V-1	0.0755	0.0773	0.0018	0.0001	0.0042	0.0041	NG-6.1 [§]	0.098	0.098	0.000
V-2	0.0755	0.0797	0.0042	0.0000	0.0012	0.0012	NG-1.3 [§]	0.098	0.098	0.000

*Data not valid or not taken, due to fuel condition

[†]Pre-Gap measurements, calculated from published pre-irradiation data package⁽²⁾

[§]NG - No gap

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It should be stressed that all postirradiation metallographic observations were based on phases and structures existing at room temperature, and at the final hydrogen level. During elevated-temperature reactor operations, and before loss of major amounts of hydrogen, completely different phases were present, as can easily be seen by consulting the zirconium-hydrogen phase diagram (Figure 1). Filar eye measurements on the fuel sleeve thickness, gap between sleeve and support rod, and thermocouple holes were also obtained from the metallographic samples. These data are shown in Table 13. Steep hydrogen gradients were observed in a number of cases, particularly in Specimens M-1, M-2, and M-3 (1600°F specimens).

Earlier observation on Specimen H-5 indicated that this specimen was probably in two phases. The upper half of the specimen surface was smooth, while the lower half was rough, pitted, and broken. The hydrogen analysis supported this; the analysis showed 1.70 H/Zr (δ) at the top, and a 1.91 H/Zr (ϵ) at the bottom of the specimen. The hydrogen analysis also indicated that Specimen V-2 was not in β phase at the end of the test; however, the general appearance and condition of the fuel structure suggests β -phase operation.⁽²²⁾ The NAA 120 fuel structure evaluation is tabulated in Table 14. Typical photomicrographs of the fuel sleeves, obtained during the metallographic examination, are shown in Figure 10.

TABLE 14
NAA-120-4 FUEL MICROSTRUCTURE EVALUATION
(Metallographic and Hydrogen Analyzer)

Specimen Number	Central Fuel Sleeve and Adjacent Support Rod										
	Phases Present			Sample Structure Condition			Postirradiation H/Zr			Indicated β -Phase Operation	
	Top	Center	Bottom	Porosity	Voids	Cracks	Top	Center	Bottom	Hydrogen Analysis	Metallographic
E-1 sleeve	ϵ †	$\epsilon (\delta + \epsilon)$		Yes	No	Micro-edge§	1 77 ^{**}	1 75 ^{**}	1 71 ^{**}	No	No
F-1 support		ϵ		No	No	No	1 77 ^{**}	1 75 ^{**}	1 71 ^{**}	No	No
E-2 sleeve		$\delta + \epsilon$		No	Yes	No	1 74	1 74	1 72	No	No
E-2 support		$\delta (\delta + \epsilon)$		No	No	No	1 65	1 67	1 70	No	No
E-3 sleeve		$\delta + \epsilon$		No	No	Micro-edge	1 72	1 68	1 72	No	No
E-3 support		$\delta + \epsilon$		No	No	Micro-edge	1 66	1 65	1 67	No	No
E-4 sleeve		$\epsilon (\delta + \epsilon)$		Yes	Yes	Yes	Sample lost	1 82	1 81	No	No
E-4 support		$\delta + \epsilon$		No	Yes	Micro-	1 78	1 78	1 77	No	No
E-5 sleeve		$\epsilon (\delta + \epsilon)$		Yes	No	Micro-	1 83	1 86	1 86	No	No
E-5 support		ϵ		No	No	No	1 84	1 86	1 88	No	No
H-1 sleeve		$\delta + \epsilon$		Yes	No	Micro-edge	1 66	1 66	1 67	No	No
H-1 support		$\delta + \gamma'$		Yes	No	Micro-edge	1 51	1 53	1 57	Yes	Yes
H-2 sleeve		$\delta + \gamma'$	$\delta + \gamma'$	Yes	No	Yes	1 74	1 58	1 44	Yes	Yes
H-2 support		$\delta + \gamma'$	$\delta + \gamma'$	Yes	No	Micro-edge	1 68	1 57	1 50	Yes	Yes
H-3 sleeve		$\delta + \epsilon$		Yes	Yes	Micro-edge	1 69	1 70	1 70	No	No
H-3 support		δ		Yes	Yes	Micro-edge	1 62	1 63	1 63	No	No
H-4 sleeve		$\delta + \epsilon$		No	No	Micro-	1 76	1 74	1 72	No	No
H-4 support		$\delta + \epsilon$		No	No	No	1 68	1 68	1 60	No	No
H-5 sleeve	δ, ϵ traces	$\delta + \epsilon$	$\delta + \epsilon$	Yes	No	Yes	1 70	1 76	1 91 ^{**}	No	No
H-5 support		δ		Yes	No	Micro-	1 68	1 72	1 91 ^{**}	No	No
M-1 sleeve		$\delta + \gamma'$		Yes	Yes	Yes	1 58 ^{**}	1 30 ^{**}	1 60	Yes	Yes††
M-1 support		$\delta + \gamma'$		Yes	Yes	Micro-	1 58 ^{**}	1 30 ^{**}	1 34	Yes	Yes††
M-2 sleeve		$\delta + \gamma'$	$\delta + \gamma'$	Yes	Yes	Yes	1 62	1 59	1 12 ^{**}	Yes	Yes††
M-2 support		$\delta + \gamma'$	$\delta + \gamma'$	Yes	Yes	Micro-	1 54	1 52	1 12 ^{**}	Yes	Yes
M-3 sleeve		$\delta + \gamma'$		Yes	Yes	Yes	1 24	1 63 ^{**}	0 90 [*]	Yes	Yes††
M-3 support		$\delta + \gamma'$		Yes	Yes	Micro	0 99	1 63 [*]	0 90 ^{**}	Yes	Yes
M-4 sleeve		$\delta + \epsilon$		Yes	No	Micro-edge	1 69	1 69	1 69	No	No
M-4 support		δ		Yes	No	Micro-edge	1 52	1 66	1 64	Yes	No
M-5 sleeve		$\delta + \epsilon$		No	No	No	1 71	1 72	1 70	No	No
M-5 support		δ		No	No	No	1 66	1 66	1 65	No	No
V-1 sleeve		$\delta + \gamma'$		Yes	Yes	Yes	1 51	1 49	1 47 ^{**}	Yes	Yes
V-1 support		$\delta + \gamma'$		Yes	Yes	Micro-	1 51	1 46	1 47 ^{**}	Yes	Yes
V-2 sleeve		δ		Yes	Yes	Yes	1 63 ^{**}	1 61 ^{**}	1 61 ^{**}	No	Yes††
V-2 support		δ		Yes	Yes	Yes	1 63 [*]	1 61 ^{**}	1 61 ^{**}	No	Yes††

*Epsilon one side only (180°)

†Support rod inside fuel support

§Surface edge of sample

**Support rod and fuel sleeve analyzed together

††Metallographic interpretation

a. E-3 Fuel Sleeve ($\delta + \epsilon$)

120-10-2

120-5-1

b. H-5 Fuel Sleeve (δ) c. V-1 Fuel Sleeve ($\delta + \epsilon'$)

121-6-1

Figure 10. Typical Photomicrographs of the NAA 120-4 Fuel
(Figure Unclassified)

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CONFIDENTIAL RESTRICTED DATA (GP-1)

Figure 11. Radial Fuel Growth vs Time (Based on fuel-cladding ΔT)

CONFIDENTIAL RESTRICTED DATA

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Figure 12. Volume Increase vs Burnup for the 1300°F Subcapsules (Based on fuel-cladding ΔT)

CONFIDENTIAL RESTRICTED DATA (GP-1)

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VI. DISCUSSION

The temperature history of the NAA 120-4 is of particular significance to the swelling model for U-ZrH_x fuel material.⁽²³⁾ Briefly, the present model, based on burnup and temperature, is represented by the following relationship:

$$\% \frac{\Delta V}{V} = \alpha B + \beta \exp (-\lambda/T) \quad \dots (4)$$

$$\% \frac{\Delta V}{V} = \% \text{ Volumetric fuel growth}$$

$$\left(\% \frac{\Delta V}{V} = 2 \frac{\Delta D}{D} + \frac{\Delta L}{L} \right) \quad \dots (5)$$

B = metal at. % burnup

T = bulk average fuel temperature (°F)

α , β , λ = constants

Although derived statistically from irradiation test data, this equation was found to contain two basic mechanisms thought to be controlling fuel swelling at burnup levels less than ~1 metal at. %:

- 1) The first mechanism is the accommodation of solid fission products resulting from fission of U²³⁵, and is represented by the αB term in the model.
- 2) The second mechanism, represented by the $\beta \exp (-\lambda/T)$ term, is attributed to a saturable cavity nucleation phenomenon. This term first appeared as a non-zero intercept, when irradiation data were plotted against burnup.

The NAA 120-4 is the first experiment in which an apparent saturation of fuel growth has been seen to occur in the first 1000 to 2000 hr of irradiation. This saturation of growth, termed "offset" swelling, was deduced from the rapid decrease in fuel-to-cladding ΔT , experienced during the early part of the irradiation (see Figure 5). Calculations of fuel growth, based on the fuel-to-cladding ΔT as a function of time, are shown in Figure 11. These curves indicate that

Figure 13. Volume Increase vs Burnup for the 1500°F Subcapsules
(Based on fuel-cladding ΔT)

Figure 14. Volume Increase vs Burnup for the 1600°F Subcapsules
(Based on fuel-cladding ΔT)

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saturation occurs at ~1500 hr, for all three experimental temperature levels. These calculations were terminated at the 2100-hr point, due to indicated fission gas release on at least two of the instrumented specimens. The released fission gas is thought to have come from the 20 wt % end pieces, which operated at ~80°F above the center sleeve temperature, and probably entered β phase early in life. Further calculations were performed⁽²⁴⁾ at points in time in which data had been taken at two or more different temperature levels. These calculations involved the simultaneous solution of the two or more sets of data for fission gas concentration in the fuel-to-cladding gap and for gap width. The values of swelling computed by this method are shown in Figures 12, 13, and 14, along with the corresponding curves from Figure 11. The values are shown as $\% \Delta V/V$, so that they can be compared directly with final measured volume changes; and are converted from diameter calculations, assuming $\% \frac{\Delta V}{V} = 3 \left(\% \frac{\Delta D}{D} \right)$. The aforementioned calculations were hampered somewhat by the fact that, due to the test reactor's control rod position, there was a 5 to 10% flux gradient from the top to the bottom of the fuel specimens. This flux gradient resulted in cladding thermocouples reading 40 to 55°F higher on top than at the bottom of the cladding. It will be noted, in Figures 12, 13, and 14, that the spread in the calculated values seems to increase as burnup increases. The reason for the scatter is not known, but is thought to be due to any one or a combination of items, such as instrumentation error, poor assumptions in thermal model, and/or poorly known thermo-physical properties of irradiated materials.

In the present fuel swelling model,⁽²³⁾ once "offset" fuel swelling has been completed, the solid fission product accommodation term becomes dominant. This term is defined as a constant x burnup, and is relatively temperature insensitive. The value of the constant, which appears to be theoretically defensible, has been determined to be ~2.76. This term has been added to the calculated offset swelling, resulting in the curves shown in Figures 12, 13, and 14.

Although these curves do not appear to match the instrumented specimens in any of the subcapsules, they do appear to correlate with some of the uninstrumented specimens. In the case of Specimen M-1 (1600°F instrumented specimen), it is known that it, as well as M-2 and M-3, entered the β -phase domain late in the irradiation, because of excessive loss of hydrogen. As has been seen in the past, excessive fuel swelling occurs when an irradiated fuel specimen enters

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Figure 15. Fuel Volume Changes vs Fuel Diameter Changes
(Corrected for hydrogen loss)

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β phase. This has been attributed to the β -phase material's lower strength and/or phase boundary sweeping of fission gases. Specimens M-4 and M-5, which did not enter β phase, appear to correlate well with the model.

Specimen H-1 also appears to be somewhat higher than the analysis would indicate, but not by a very wide margin. This, again, is probably due to the instrumentation or calculational errors described previously. It is interesting to note the consistency of Specimens H-1 and H-3 (pre-irradiation H/Zr = 1.70), and of Specimens H-4 and H-5 (pre-irradiation H/Zr = 1.85). According to the phase diagram, these specimens should all have been in δ phase while at operating temperature. The difference may have been that, during the helium mode of operation, Specimens H-4 and H-5 probably crossed the δ - ϵ phase boundary. There is, however, some uncertainty in the location of this δ - ϵ phase boundary for irradiated U-ZrH, and it is therefore impossible to state that E-4 and E-5 did not cross this phase boundary. Specimens E-1, E-4, and E-5 did swell significantly more than E-2 and E-3, apparently due to the presence of ϵ -phase material.

A plot of diameter changes versus the volume changes based on density measurements is shown in Figure 15. The diameters of some of the specimens were measured at more than one location. The plot shows that most of the data tend to follow the isotropic swelling line. The density of M-1 was probably biased by its broken and cracked condition.

In correlating U-ZrH fuel swelling data from different irradiation experiments, the greatest success has been achieved by using the "offset," or equilibrium, bulk average fuel temperature.⁽²³⁾ In the case of a constant cladding temperature experiment, the equilibrium fuel temperature is that temperature which the fuel reaches after offset swelling has been completed. The NAA 120-4 was essentially a constant-fuel-temperature experiment; therefore, the equilibrium temperature was derived differently. Since the operating data indicated that ~ 1500 hr is needed to reach equilibrium, the highest temperature at which the instrumented fuel specimens operated for 1500 hr or more was determined. The temperatures of the uninstrumented specimens were then obtained, using the temperature of the instrumented specimen and correcting for relative heat-generation rate (burnup rate), using the mass-spectrographic burnup analyses.

TABLE 15
SUMMARY OF NAA 120-4 EXPERIMENT DATA

Specimen Number	Fuel Temperature (°F)		Beginning- of-Life		End-of-Life		Mass- Spectrographic Burnup (metal at. %)	Average % $\frac{\Delta V}{V}$ (hydrogen corrected)
	Design	Calculated Equilibrium Bulk Average	H/Zr	Phase*	H/Zr	Phase†		
E-1	1300	1293	1.85	ε	1.74	ε (δ + ε)	0.832	7.44
E-4	1300	1336	1.86	ε	1.81	ε (δ + ε)	0.859	5.38
E-5	1300	1317	1.86	ε	1.85	ε (δ + ε)	0.847	6.66
E-2	1300	1268	1.72	δ	1.73	δ + ε	0.816	4.69
E-3	1300	1293	1.72	δ	1.71	δ + ε	0.832	4.22
H-1	1500	1476	1.71	δ	1.66	δ + ε	0.832	9.57
H-3	1500	1489	1.70	δ	1.70	δ + ε	0.839	10.06
H-2	1500	1518	1.71	δ	1.59	δ + γ'	0.855	13.25
H-4	1500	1551	1.85	δ	1.74	δ + ε	0.874	8.40
H-5	1500	1585	1.85	δ	1.79	δ + ε	0.893	8.52
M-1	1600	1560	1.71	δ	1.49	δ + γ'	0.816	27.76
M-2	1600	1539	1.72	δ	1.44	δ + γ'	0.805	23.32
M-3	1600	1568	1.70	δ	1.26	δ + γ'	0.820	25.60
M-4	1600	1620	1.85	δ	1.69	δ + ε	0.847	13.68
M-5	1600	1620	1.84	δ	1.71	δ + ε	0.847	16.30
V-1	1500	1485	1.71	δ	1.49	δ + γ'	0.816	6.65
V-2	1500	1491	1.71	δ	1.62	δ [§]	0.851	16.96

*From phase diagram at operating temperature

†From postirradiation metallography (room temperature)

§Metallographic interpretation indicates V-2 entered β phase

[REDACTED]

These temperatures were then corrected for fuel ΔT , to obtain bulk average fuel temperatures. These temperatures, along with the more pertinent experiment data, are summarized in Table 15. The specimens have been separated according to temperature and H/Zr, to show the definite influence of hydrogen content on fuel performance.

A plot of reciprocal absolute temperature vs burnup-corrected $\Delta V/V$ shows that the δ -phase specimens in the experiment correlate fairly well (see Figure 16). The figure also shows the ϵ -phase, as well as β -phase, material swelled more than the δ -phase material.

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Figure 16. Burnup-Corrected Volume Change vs Reciprocal Calculated Fuel Temperature

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VII. CONCLUSIONS

All five subcapsules from the NAA 120-4 appeared in excellent condition after irradiation. Postirradiation examination showed that all cladding tubes were intact following the test. Only one specimen suffered NaK in-leakage during the test (Specimen V-1), which was attributed to a faulty weld. Burnup levels achieved were within 10% of the target 0.9 metal at. %.

The NAA 120-4 confirmed the existence of the postulated "offset" fuel growth which saturates early in life. Since the NAA 120-4 was designed to operate at a much higher burnup rate than typical SNAP reactor fuel, the time to saturation growth for SNAP fuel is uncertain. The temperature dependence on time to saturation was not clearly defined by this experiment.

As expected, fuel specimens which entered the β -phase domain showed volume changes larger than those specimens which operated in the δ -phase domain. Specimens which operated in the ϵ -phase domain also showed larger volume growth than comparable specimens in the δ -phase domain, although these specimens could have been affected by the δ - ϵ phase boundary.

The test results are generally consistent with earlier irradiation data, although exhibiting a somewhat higher temperature dependence than indicated in previous correlations. The data are being correlated with previous irradiation data, under a different program.

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