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EVALUATION OF FUEL ELEMENTS
FROM
SNAP 2
DEVELOPMENT REACTOR CORE
(Title Unclassified)

AEC Research and Development Report

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By

PAUL S. OLSON

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K. J. Miller conducted hot cell examinations of the fuel elements and compiled the experimental data. K. R. Birney analytically established the power distribution in a SNAP 10A fuel element with hydrogen redistribution. A. H. Willis consulted on fission gas release mechanisms from the surface of this fuel moderator material.

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ABSTRACT

The fuel elements used in the SNAP 2 Development Reactor contained a hydrided 90 Zr - 10 U fuel/moderator rod and beryllium oxide reflectors, enclosed in ceramic-coated Hastelloy N cladding. The elements were in the core for 11,290 reactor operating hours, and produced a total energy release of 272,900 kwh. Peak fuel temperature at SNAP 2 conditions was 1279°F.

Eight fuel elements were inspected at the hot cell after disassembly of the reactor. The cladding of two fuel elements which had been in contact with the beryllium internal reflectors had reacted with the beryllium, for a penetration of about 0.003 in. (about a 1/8-in. diameter reaction zone). Apart from the small reaction zone on these two elements, all elements were in excellent condition. Measured burnup of the fuel/moderator material was 0.027 metal at. %.

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Figure 1. S2DR Fuel Element
(Figure Unclassified)

I. INTRODUCTION

A. SNAP 2 DEVELOPMENT REACTOR DESCRIPTION¹

The SNAP Development Reactor (S2DR) was the second* in a series of compact reactor tests. The SNAP 2 design point for this test involved operation at 55 kwT power with a NaK core coolant outlet temperature of 1200°F.

The fuel/moderator material[†] for the reactor was an alloy of 90 Zr - 10 U (93% enriched), hydrided to a core average $N_H = 6.45$.[§] The fuel was fabricated into 10-in. long by 1.212-in. diameter rods, and was sheathed in 1.250-in. OD by 0.014-in. wall Hastelloy N tubing. The interior of the cladding contained a 0.0035-in. thick hydrogen diffusion barrier of boron-free glass enamel coating (Solar Aircraft Company's S14-35A-SM2), modified to contain Sm_2O_3 to pre-poison the reactor with burnable samarium. The clad element also contained beryllium oxide slugs, 1.211 in. in diameter by 1.50 in. long, which were placed on either end of the fuel rod to form top and bottom neutron reflectors. The element was sealed by welding ceramic-coated end caps in place and heating at an elevated temperature to fuse the faying ceramic coatings. The resulting element had a 0.0015-in. diametral clearance between fuel and cladding, and no axial clearance. A sketch, giving the dimensions of the fuel element, is shown in Figure 1.

The reactor core contained 37 fuel elements, arranged on 1.260-in. centers in a triangular cell, to form a hexagonal array 8 in. across flats and 9 in. across corners. Six stationary beryllium metal reflector pieces were included in the core, to form the overall circular cylinder which was housed in the Hastelloy C core vessel. A cross-sectional sketch of the S2DR layout is presented in Figure 2. Grid plates, fabricated from Hastelloy C, laterally supported the fuel elements. The eutectic alloy of sodium and potassium (NaK) was used as the heat transfer medium for the core. The NaK entered through a plenum at the bottom of the reactor and was discharged at the top.

*The first reactor in the series was the SNAP Experimental Reactor (SER), which contained 61 fuel elements of 93 Zr - 7 U, hydrided to $N_H = 6.40$ and enclosed in 1-in. diameter Type 347 stainless steel cladding.

†There were no agents added to this fuel/moderator material for the purpose of grain size control.
§ N_H = Hydrogen atoms per cubic centimeter of fuel $\times 10^{-22}$.

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Figure 2. Core Cross Section
(Figure Unclassified)

B. REACTOR HISTORY²

The SNAP 2 Development Reactor operated from April 1961 through December 1962. The operation comprised runs at various temperatures and power levels, hydrogen loss and redistribution experiments, and various other core performance measurements (see Figure 3). The reactor performed for 11,290 hr, with a total energy release of 272,900 kwh. The history was highlighted by 2060 hr at 55 kw and 1200°F outlet temperature (SNAP 2 condition), 1544 hr at 30.5 kw and 1100°F, and 1150 hr at 30.5 kw and 950°F (SNAP 10A conditions). The peak fuel temperature at SNAP 2 operating conditions was 1279°F. The reactor was disassembled in the first half of calendar year 1963, and fuel element examinations were conducted during the second half of that year.

C. FUEL ELEMENT EVALUATION SCOPE

The examination was directed at inspecting the following eight S2DR fuel elements, as illustrated in the numbering scheme given in Figure 2.

<u>Ring Number</u>	<u>Element Numbers</u>
II	2, 3, 4, 5
III	6
IV	6, 8, 15

The eight fuel elements were evaluated for obvious visual effects, hydrogen permeation characteristics, and gamma activity profiles. A detailed examination of one fuel element was planned to include fission gas release, visual inspection of the cladding-coating and fuel rod, burnup analysis, hydrogen analysis, dimensions, density, and metallography.

Figure 3. S2DR Operating History
(Figure Unclassified)

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II. COMPOSITE ELEMENT EXAMINATIONS

A. VISUAL INSPECTION

The eight fuel elements were visually examined at the CDHC,* using a Kollmorgan periscope and stereomicroscope. During core disassembly, test facility people noted that two fuel elements (IV-6 and IV-15) had small reaction zones, approximately 1/8 in. in diameter, at the bottoms of the elements. These elements were located in the outer ring of the core, and were adjacent to the internal beryllium metal reflectors. Except for these two small reaction areas, all elements appeared to be in excellent condition, with no indications of erosion, corrosion, or heat-affected zones. The eight elements are shown, in Figure 4, as they appeared when viewed through the Kollmorgan periscope. The surface condition of the cladding was typical of its appearance after the chromizing† and liquid honing phases of the production process. Photographs of the reaction zones of Elements IV-6 and IV-15 were taken through the stereomicroscope, and are presented in Figures 5 and 6. The circumferential marks at the bottom of the cladding, immediately below the reaction product, resulted from machining of the weld bead during manufacture of the end-cap-to-tubing joint.

B. REACTION ZONE OF ELEMENTS IV-6 AND IV-15

1. Metallography and Analysis

Microsections were made directly through the center of the reaction zone on Element IV-15. Photomicrographs of the reaction zone are given in Figures 7 and 8, and show the reaction area. A large portion of the reaction product was dislodged from the metal during the mounting and polishing of the sample, as can be seen in Figure 7. The measured penetration was 0.003 in.

The reaction product found on Element IV-6 was filed off for analysis. X-ray diffraction and fluorescence analyses failed to identify the reaction product; however, by emission spectroscopy relative to a standard Hastelloy N sample, the reaction product was found to have a high concentration of beryllium.

2. Discussion

From the brittle nature of the reaction product and the fact that it was high in beryllium, it is believed that this product is probably a beryllide.

*CDHC - Component Development Hot Cell at Atomics International

†Solar Aircraft's F3-78 chromizing process

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Figure 4. Fuel Elements Examined at the CDHC
(Figure Unclassified)

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Figure 5. External Cladding Reaction Area of Element IV-15, Located $\frac{3}{4}$ in. From Bottom of Element

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Figure 6. External Cladding Reaction Area of Element IV-6, Located $\frac{5}{8}$ in. From Bottom of Element

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Figure 7. Cross Section of Cladding From Element IV-15 at the Reaction Zone

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Figure 8. Reaction Zone of the Element IV-15 Cladding

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Previous experimental results on the compatibility of SNAP fuel and cladding materials showed that beryllium metal and nickel-based alloys, such as Hastelloy B and Inconel-X readily formed diffusion couples, when in intimate contact at 1300°F.³ The chromized coating along the length of the element prevented intimate contact and diffusion coupling, except for the area at the bottom of the element, where the weld-cleaning operation was performed and resulted in the removal of this coating.

C. GAMMA SCANS

The axial gamma activity was measured on Elements II-2, II-3, III-6, and IV-8 by mounting each element on a motor-driven stage which moved back and forth in front of an 1/8 in. collimator. A NaI (Tl) crystal and photomultiplier supplied the signal to a single channel analyzer. The analyzer was adjusted, using a Zr⁹⁵ (0.72 Mev photopeak) standard, and the relative activity was recorded on a strip chart with a 2 to 1 multiplication factor.

1. Axial Profiles

From the gamma scans, the area under the profile was measured with a planimeter. The average height was then calculated by dividing the area under the curve by the length of the fuel rod. The peak height was measured on the basis of a smooth curve through the gamma scan. Table 1 lists these measurements and the Peak/Average values for the four fuel elements. These values are in excellent agreement with the axial Peak/Average power prediction of 1.26.⁴

TABLE 1
SUMMARY OF FUEL ELEMENT GAMMA
SCAN MEASUREMENTS

Element Number	Average Height	Peak Height	Peak/Average
II-3	2.41	3.14	1.30
II-2	2.43	3.13	1.29
III-6	2.09	2.68	1.28
IV-8	1.71	2.18	1.28

2. Hydrogen Redistribution and Power Shift

The axial gamma scan of Element II-3, as shown in Figure 9, is typical of the activity profiles. The predicted axial power profile, using the one-dimensional neutron diffusion code, ZOOM,⁵ has been superimposed on the gamma scan for comparative purposes. Generally speaking, gamma scans were begun at a location two collimator diameters (1/4 in.) from the ends of the rod. The rapid drop off in activity at the ends of the rod is believed to be the result of experimental end effects. The end effects result from the low activity of the rod relative to background and geometric factors where scattering contributions from the fuel to the detector are less at the ends of the body. In every case, the area under the curve over the bottom half of the fuel rod was greater than the area under the curve over the top half. The average ratio of activities for the four fuel rods, comparing bottom to top, is 52.5 to 47.5. Hydrogen redistribution due to axial temperature gradients was not taken into account in the earlier S2DR calculations using ZOOM (Figure 9). However, recent analytical predictions on a SNAP 10A core,* using the HYTRAN⁶ computer program to redistribute the hydrogen and the DTK⁷ neutron transport code to evaluate axial power density, showed a spatial shift effect in the same direction and of the same magnitude.

3. Radial Profile

The activities of the individual elements, as represented by the average heights, were normalized to Element II-3. These data were then compared to the radial power profile for the core, as predicted using ZOOM. The results are shown in Figure 10. Experimental results are in excellent agreement with the predicted profile.

D. DIMENSIONS

The diameter and straightness of Fuel Elements II-2, II-4, III-6, and IV-8 were measured. Each element was supported in chucks on a lathe bed; and, using linear transducers, a strip chart record of the dimensional profile was obtained. Profiles of each element were taken at two orientations. Length measurements were then made from end cap to end cap, using vernier calipers. The diameter and straightness measurement precision was ± 0.0005 in. and length was ± 0.001 in. All dimensions were compared with measurements made

*The SNAP 10A fuel element does not contain reflectors.

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Figure 9. Axial Gamma Activity Profile for Fuel Element II-3 (greater activity at bottom of element is result of hydrogen redistribution)

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Figure 10. Radial Power Density in S2DR Normalized to 1 at Center of Core

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before insertion in the reactor core. It can be seen, from Table 2, that, within the limits of measurement, there were no changes in the diameter, straightness, or length of the elements.

TABLE 2
DIMENSIONS OF FUEL ELEMENTS

Element Number	Diameters (in.)		Straightness (in.)		Length (in.)	
	Before	After	Before	After	Before	After
II-2	1.248-1.251	1.248-1.251	0.005	0.005	13.222	13.223
II-4	1.248-1.251	1.248-1.252	0.000	0.000	13.223	13.225
III-6	1.249-1.251	1.248-1.252	0.005	0.004	13.224	13.224
IV-8	1.248-1.251	1.249-1.251	0.007	0.007	13.226	13.227

E. PERMEATION MEASUREMENTS

1. Analytical Predictions

To compare experimental permeation measurements among fuel elements which have been at elevated temperatures for some period of time, it is necessary to correct the preoperational values to reflect the reduced permeation rates that would be expected from normal loss of hydrogen. The computer code HYPAT⁸ is used for this purpose. This code computes the permeation rate and hydrogen loss for a given set of operating conditions as follows:

For the initial atom ratio (H/Zr), and average axial fuel rod operating temperature,

$$T(1)_{\text{fuel}} = A_1 + A_2 l + A_3 l^2, \quad \dots(1)$$

where A_1 , A_2 , and A_3 are constants, and l is the fuel rod length, the HYPAT model computes a single equilibrium pressure,

$$P_{\text{Eq}} = \exp \left[B_1 + B_2 (H/Zr) + B_3 (H/Zr)^2 \right], \quad \dots(2)$$

where B_1 , B_2 , and B_3 are constants. This equilibrium pressure is then used to compute permeation losses over arbitrary increments of time. The total fuel

element permeation rate (L_{FE}) is governed by transmission of hydrogen through the ceramic diffusion barrier (L_{C1}) and defects (L_{BLD}) in the coating,

$$L_{FE} = L_{BLD} + L_{C1} \quad \dots(3)$$

The permeation through the defects in the coating is characteristic of the permeation through metal, and is given by

$$L_{BLD} = K_{BLD} \sqrt{P_{Eq}} \exp(-K_1/T_c), \quad \dots(4)$$

where K_{BLD} and K_1 are constants, and T_c is the temperature. The permeation through the ceramic diffusion barrier is given by

$$L_{C1} = \frac{K_{Aging} P_{Eq} T_c^{5/2} \exp(-K_2/T_c)}{T_c^{5/2} + K_3 P_{Eq} \exp(-K_2/T_c)}, \quad \dots(5)$$

where K_{Aging} , K_2 , and K_3 are constants. The permeation is then summed for the end caps and tubular portion of the elements by setting T_c equal to the appropriate temperatures. This total element permeation is used for an arbitrary increment of time to compute the loss in hydrogen and reduced equilibrium pressure. A new permeation rate is then computed for each following incremental time step, thus providing a stepwise function of permeation rate vs time.

2. Experimental Methods

Hydrogen permeation measurements were made at the hot cell by placing each fuel element in a Vycor retort inside a clamshell furnace and heating to 1200°F. The system was evacuated through a mass spectrometer, modified to be sensitive only to hydrogen. Each element was kept at temperature until a constant hydrogen leak rate was obtained over at least a 6-hr period of time.

A SNAP 10A element (N154), which had experienced a series of environmental and qualification tests, was used as a reference in this series of permeation tests. This element had an established permeation rate of 0.26 cc(STP)/hr at 1200°F through its history of testing. Before the series of tests at the hot

cell, it recorded 0.26 cc(STP)/hr at 1200°F; and, after the series of tests, 0.21 cc(STP)/hr. On the basis of this standard, the equipment was assigned a 20% measuring tolerance. This tolerance was also assigned to preoperational results, since the measurements made on the S2DR elements represented some of the first permeation tests conducted at AI, when permeation testing techniques were under development.

3. Comparison of Predicted and Experimental Results

The preoperational hydrogen content, fuel element permeation measurements, and postoperational calculated and measured permeation rates are presented in Table 3. The calculated values were obtained by computing K_{BLD} and K_{Aging} constants for the initial permeation rates and dissociation pressures.

TABLE 3

FUEL ELEMENT HYDROGEN PERMEATION MEASUREMENTS

Element Number	Preoperational Measurements		Postoperation Results L[cc(STP)/hr] at 1200°F	
	N_H	L[cc(STP)/hr] at 1200°F	Calculated	Measured
II-2	6.59	1.60	1.34	1.42
II-4	6.54	0.84	0.76	1.17
IV-8	6.28	0.40	0.38	0.79
III-6	6.47	0.29	0.28	0.73

The reactor operating history, at the various power levels and corresponding temperature profiles, were then programmed in 100-hr increments to obtain the reduced permeation rates, using HYPAT. These rates were then corrected to reflect an isothermal condition. Within the assigned limits of testing error, there is no detectable change in Element II-2. The changes in permeation rates of Elements II-4, III-6, and IV-8 are considered to be significant; however, the change was not of a magnitude sufficient to limit operation of the S2DR.

III. DETAILED EXAMINATION OF ELEMENT II-3

A. FISSION GAS RELEASE

Fuel Element II-3 was punctured, to collect the fission gas which had been released into the annulus between the fuel rod and cladding. This was done by supporting the element in a closed chamber and drilling into the end of the element to release the gas. The system was then evacuated through activated charcoal traps, where the gas was collected. The traps were then analyzed for Kr^{85} activity.

1. Surface Release and Recoil Calculations

Fission gas is believed to be released from hydrided Zr-U primarily by surface release and recoil mechanisms. In-pile experiments have shown the diffusion of fission gases to be unimportant at S2DR temperatures.⁹

The major contributor to fission gas in the annulus of the fuel element is believed to be through surface release. This is the process where the surface uranium which is dispersed in the zirconium hydride matrix undergoes hydriding and dehydriding phase changes, resulting in the release of the fission gas from the particles. Assuming that the uranium particles in the matrix are spheres of equal diameter, and the area fraction of particles seen on a plane is equal to the volume fraction, the number of fission product atoms released is:

$$N_i = FDSY_i, \quad \dots (6)$$

where:

N_i = number of atoms of the isotope released

F = total fissions per unit volume

D = average uranium particle diameter

S = geometric surface area of the body

Y_i = fission yield of the isotope

For a uranium particle diameter of $0.9 \times 10^{-3} \mu$,¹⁰ a burnup of 0.027% of the metal atoms available (see Section III-C-4), and a Kr^{85} fission yield of 0.3 atoms per 100 fissions, the number of Kr^{85} atoms released is 0.55×10^{15} , or 0.013% of Kr^{85} produced.

The fraction of an isotope released by recoil of the total produced is

$$\frac{SL_f}{4V} , \quad \dots (7)$$

where

S = geometric surface area of fuel-moderator rod

L_f = recoil length of the isotope in the fuel

V = volume of the rod.

The recoil range in the S2DR fuel moderator material was estimated at 9.6×10^{-4} cm, based on the density of the material relative to the density and recoil length of zirconia.¹¹ For this S2DR fuel-moderator rod geometry, the recoil yield of Kr^{85} would then be 0.034%. However, for Fuel Element II-3, with a diametral gap of 0.001 in., the actual quantity expected to be found in the gap would be less than 1% of this recoil yield; since most recoil nuclides would be imbedded in the cladding, due to the long recoil length (~ 1 in.) for the fission products in the annulus environment.

2. Experimental Results

The analysis for Kr^{85} in the fission gas sample from Element II-3 yielded an activity of 2.58×10^8 dis/min. When corrected for decay, this gives a Kr^{85} atom release of 2.2×10^{15} , or 0.052% of that produced. Although this release is somewhat higher than that calculated, it is felt that the magnitude is representative of the process involved. Factors such as uranium particle diameter uncertainties, scattering of fission products, return of fission products to the annulus, and thermal spikes are influential and unaccounted for in the calculations, and may account for the predicted value being lower than the measured value.

B. CLADDING AND HYDROGEN BARRIER

The end caps were removed from Element II-3 with a cutoff wheel, and two longitudinal cuts of the cladding were made with an electric discharge Elox machine. Since one of the longitudinal cuts did not have complete penetration, it was necessary to pry the cladding open with the hot cell manipulators, to free the fuel rod and beryllium oxide reflector pieces.

1. Visual Inspection

The interior ceramic coating was viewed through a stereomicroscope. Directly in line with the bottom beryllium reflector, the coating color was green, as is typical after the production firing operation. This color changed along the length of the cladding, approaching a blackened color at the top of the element. A transition in color was apparent near the top of the cladding, between the fuel rod and top beryllium oxide reflector. The darkening followed a pattern, along the length of the fuel element, typical of the temperature profile in the reactor. This appearance was characteristic of the discoloration of the coating from aging at elevated temperatures, observed in out-of-pile tests.

Prying the cladding to remove the fuel rod resulted in chipping of the coating from the cladding. Figure 11 shows the discoloration and chipped coating.

2. Metallography

Cladding samples were taken for metallographic inspection at axial locations corresponding to the top and bottom of the fuel rod; Figure 12 shows the typical structure observed. The thickness of the cladding was found to be in the range of 0.0129 to 0.0134 in. Measurements made on the tubing used in this element during the assembly stage were in the range of 0.0129 to 0.0136 in. No deterioration of the ceramic coating, chromized layer, or metal cladding was observed.

C. FUEL MODERATOR ROD

The fuel rod from Element II-3 was inspected and found to be intact, free of visible defects, heat affected zones, and signs of mass transfer. The fuel rod was broken into several pieces when it accidentally fell from the jaws of the hot cell manipulator during a routine transfer. A photograph of the fuel rod and beryllium oxide end reflectors is presented in Figure 13.

1. Dimensions and Density

Diameter measurements of the fuel rod were taken with vernier calipers, and densities were measured by liquid displacement in n-octyl alcohol. The results of the dimension and density measurements are given in Tables 4 and 5. No detectable change is evident between these results and measurements taken during the assembly of the fuel element. This is in agreement with results of

7635-5171

Figure 11. View of the Inside of the Cladding of Element II-3

Etchant: Marbles
with CuSO_4

7635-5172

Figure 12. Typical Structure of the Cladding From Element II-3

7635-5173

Figure 13. Element No. II-3 Fuel Rod and BeO Reflectors (Left reflector cut during element disassembly.) (Photo taken after rod dropped.)

TABLE 4

FUEL ROD DIAMETER — ELEMENT II-3
(Preirradiation Diameter = 1.2120 to 1.2122 in.)

Distance From Bottom of Fuel Rod (in.)	Diameter (in.)	
	0°	90°
0.50	1.213 ± 0.001	1.212 ± 0.001
2.50	1.213	1.212
3.50	1.212	1.212
5.50	1.213	1.213
7.00	1.212	1.213
8.25	1.212	1.213
9.75	1.220*	1.212

*The one abnormal value at 9.75 in. and 0° orientation was probably due to something clinging to the fuel at that point. The vernier calipers were checked on a standard, and found to be precise to ±0.001 in.

TABLE 5

FUEL DENSITY MEASUREMENTS — ELEMENT II-3

Sample Location	Density (g/cc)*
Top	6.025
Top	6.022
Top	6.022
Middle	6.035
Middle	6.032
Middle	6.028
Bottom	6.035
Bottom	6.034
Bottom	6.033
Average	6.030 ± 0.005

*Pre-irradiation Density = 6.033 g/cc

Etchant: HF,
HNO₃, and Water

7635-5174

Figure 14. Outside Edge of Fuel
Showing Columnar Structure

Etchant: HF,
HNO₃, and Water

7635-5175

Figure 15. Typical ϵ Structure
Found Throughout Fuel

7635-5176

Figure 16. Diametral Activity Profile of Fuel From Element No. II-3

experiments on this type of fuel, where the swelling of the fuel from burnup has been characterized by the relationship⁹

$$\frac{\Delta V}{V} = 3770 b^{1.5} \exp(-12,000/T) \quad \dots (8)$$

where

$\Delta V/V$ = volume increase (%)

b = fuel burnup (metal at. %)

T = average peak fuel temperature (°R).

In the S2DR, this would result in a maximum volume increase of this fuel rod of 0.029%. This small change is beyond the capabilities of detection, for the precision of measurements in these tests.

2. Metallography

A wafer, 1/2 in. thick, was cut from the center of the fuel rod, and was mounted and polished for metallographic inspection. The axial center of the element was chosen, since this represented the location of peak burnup in the reactor core.

Inspection of the sample revealed columnar structure (Figure 14) at the surface, which is characteristic of the fuel composition in Element II-3, and banding throughout the body, typical of the ϵ phase of zirconium hydride (Figure 15). There was no apparent change in the appearance or structure of the fuel, after its history of operation in the S2DR. The metallographic mount was then gamma scanned across the diameter of the fuel, in the same manner as that used for the elements. The resultant gamma scan is illustrated in Figure 16. Gamma scans of the fuel elements (see Section II-C) pointed out the necessity of entering the field two collimator diameters from the edge of the sample, to avoid end effects. Lines corresponding to the two collimator diameter positions are drawn on the fuel wafer gamma scan. The center of the fuel shows the power to be uniform in the area where confidence can be assigned to the experimental technique. Predictions of the power distribution across the diameter of the fuel, using the DTK code, give a peak-to-minimum ratio of 1.03. The gamma scan has been extrapolated to the edge of the fuel wafer to show this flat profile.

3. Hydrogen Content

Hydrogen content of Fuel Rod II-3 was determined by heating samples in an induction furnace and analyzing the quantity of hydrogen driven off in an NRC* vacuum extraction unit. This unit consists of calibrated volumes and gauges for measuring the pressure in the volumes. Various traps are provided, to remove fission and condensable gases.

The results of hydrogen analyses on samples from the top, middle, and bottom of the fuel moderator rod are given in Table 6. The difference in hydrogen content from the top to the bottom, 0.09 wt %, is identical with the difference in content implied by the density measurements. This change is the result of the operating conditions of the S2DR, immediately prior to shutdown, when externally heated NaK was pumped through the reactor. Since the reactor was not critical, the hot NaK at 1150°F entered the bottom of the reactor; and, through normal heat loss, was passed out through the top of the reactor at a somewhat lower temperature. The result was a redistribution of hydrogen in the fuel-moderator material, with the greater concentration going to the top.

TABLE 6

HYDROGEN ANALYSIS OF FUEL ROD - ELEMENT II-3

Sample Number	Distance of Sample From Top of Fuel Rod (in.)	Hydrogen Content (wt %)
1	1	1.81
2	1	1.79
3	5	1.73
4	9	1.74
5	9	1.70
6	9	1.70

This fuel rod had an initial hydrogen content of 1.83 wt %, based on weight gain and chemical analysis. Hydrogen loss through permeation of the element was computed, using HYPAT (see Section II-E). The predicted hydrogen level was 1.81 wt % at the end of operation of the S2DR. From variations

*NRC-National Research Corporation, Palo Alto, California

in the data, the 95% confidence limits about the average are 1.78 and 1.73 wt %. The assumption was made that a linear relationship exists between hydrogen content and the length of rod. The data show the hydrogen loss to be greater than that predicted; however, the total change in hydrogen is small, and uncertainties in the initial content or distribution within the rod could account for the difference between that predicted and what was measured.

4. Burnup Determination

A sample of fuel from the center of the fuel rod was analyzed for Ce^{144} . This sample was a complete cross section wafer, 1/8 in. thick. Assuming a 5.3% fission yield of Ce^{144} , the burnup was 0.027 % of the total metal atoms in the fuel moderator material. The burnup determination is considered to have a tolerance of 10%, due to the uncertainty in the fission yield of Ce^{144} . Therefore, this result is in agreement with the calculated burnup of 0.025 metal at. %, based on energy release of the core.

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IV. CONCLUSIONS

The successful operation of the S2DR represents, in part, the result of outstanding performance of the fuel elements used in the core. The following conclusions are based on the examination of a selected number of fuel elements from the disassembled core.

Gamma scan profiles confirmed the analytical predictions of spatial power distributions in the core, and showed the effects of axial hydrogen redistribution resulting from the temperature gradients.

A beryllium reaction zone occurred in the Hastelloy N cladding, on two elements, in areas directly adjacent to the internal core reflector. The reaction occurred only at the bottom end cap area, where weld cleanup machining marks were apparent.

Three of the four fuel elements which were permeation tested showed hydrogen leak rates slightly in excess of those predicted. While the magnitude of change was not severe enough to limit overall core performance, analytical models are under development to account for the various effects responsible for hydrogen losses from similar types of ceramic-coated fuel elements.

Fission gas release from the fuel rod in Element II-3 was typical of the quantity expected from surface effects, and was not of a sufficient magnitude to limit the fuel element design.

The cladding and coating of Element II-3 appeared to be in excellent condition, with no visible evidence of deterioration from the NaK or irradiation. Discoloration of the coating from thermal aging was typical of that observed in out-of-pile tests.

There were no significant changes in the dimensions or density of the fuel moderator material from Element II-3. Hydrogen loss from the material, observed in vacuum extraction analysis, was greater than that expected; however, uncertainties in the initial concentration of hydrogen and distribution within the rod could account for the discrepancy.

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