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ZIRCALOY-CLAD UO_2 FUEL ROD
EVALUATION PROGRAM
QUARTERLY PROGRESS REPORT NO. 11
MAY - JULY 1970

MASTER

COMPILED BY
F. H. MEGERTH

U.S. ATOMIC ENERGY COMMISSION
CONTRACT AT(04-3)-189
PROJECT AGREEMENT 41

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GEAP-10217
AEC Research and
Development Report
August 1970

ZIRCALOY-CLAD UO_2 FUEL ROD
EVALUATION PROGRAM

Quarterly Progress Report No. 11
May-July 1970

Compiled by
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1. INTRODUCTION

At the time of the final shutdown of the Vallecitos Boiling Water Reactor (VBWR) and the termination of the AEC Fuel Cycle Program (Contract AT(04-3)-189, Project Agreement 11), the leading Zircaloy-2-clad UO_2 fuel rods of the program's Type J basic assemblies had attained rod average exposures as high as 3×10^{20} fissions/cm³ (10,000 MWd/t U).

To demonstrate at the earliest possible time the capability of Zircaloy-clad fuel to achieve average burnups of at least 6×10^{20} fissions/cm³ (20,000 MWd/t U) and to take advantage of the available lead time in achieving these burnup targets, 98 of the higher exposure Fuel Cycle Program rods were incorporated in a special assembly designated as SA-1 and inserted in the Dresden Reactor Unit No. 1 for additional irradiation. The assembly consisted of two 7-by-7 subassemblies, stacked one on top of the other in a channel of standard dimensions for the Dresden reactor.

Assembly SA-1 was operated continuously from June 1964 to September 1969, with the exception of reactor shutdowns for refueling and maintenance.

To investigate in detail the performance of the SA-1 fuel rods and to determine the isotopic composition of the highly exposed fuel, a two-phase program of destructive examination and analysis was drawn up. For the initial phase of the program, ten rods were removed from the assembly in January 1967, after 1200 days of VBWR and Dresden operation, and were shipped to the Vallecitos Nuclear Center. The remaining rods operated a total of 1850 days. The two 44-rod subassemblies were shipped last March to VNC, where the final phase of the program is now in progress.

Prior reports to the Atomic Energy Commission under Project Agreement 41 comprise the following:

- a. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 1, November 1967—January 1968," March 1968 (GEAP-5598).
- b. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 2, February 1968—April 1968," May 1968 (GEAP-5624).
- c. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 3, May 1968—July 1968," August 1968 (GEAP-5667).
- d. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 4, August 1968—October 1968," November 1968 (GEAP-5702).
- e. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 5, November 1968—January 1969," February 1969 (GEAP-5746).
- f. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report Nos. 6 and 7, February—July 1969," August 1969 (GEAP-10079).
- g. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 8, August—October 1969," November 1969 (GEAP-10121).
- h. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 9, November 1969—January 1970," February 1970 (GEAP-10160).
- i. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 10, February—April 1970," May 1970 (GEAP-10204).

2. SUMMARY

The relative rod-average burnups of 41 rods were determined by gamma scanning with a germanium (lithium) detector system and counting the Cs-137 line. The minimum-to-maximum burnup ratio among these rods was 0.82. The axial gamma profiles of the 41 rods were of similar shape. The ratios of axial peak burnup to average burnup for the different rods ranged from 1.06 to 1.17. Some data on the distributions of Ce-144, Ru-106, and Zr-95 also were obtained by gamma scanning.

Diameter profiles determined on six fuel rods before and after chemically removing the deposited crud indicated rod-average crud thicknesses that ranged from 0.0008 cm (0.0003 in.) to 0.0017 cm (0.0007 in.) and had an overall mean value of 0.0012 cm (0.0005 in.). The maximum circumferential-average crud thickness at any axial location of any rod was 0.0028 cm (0.0011 in.). When compared with pre-irradiation rod diameter measurements, the diameters measured after decruding showed an overall average increase of 0.0025 cm (0.0010 in.) and a maximum rod-average increase of 0.0046 cm (0.0018 in.). Some local increases in diameter up to approximately 0.010 cm (0.004 in.) were noted. All rods showed peaks in

their diameter profiles at the VBWR spacer locations, and quite frequently peaks also were present at the SA-1 subassembly spacer locations. Some instances of diameter peaking at pellet-length intervals were noted. The difference in diameter between these peaks and the corresponding troughs was typically 0.0012 to 0.0019 cm (0.0005 to 0.00075 inch) and had a maximum value of about 0.0025 cm (0.001 inch). The difference between the maximum diameter and the minimum diameter at a given point along the rod axis ($\text{o.d.}_{\text{max}} - \text{o.d.}_{\text{min}}$) was as large as 0.0075 or 0.010 cm (0.003 or 0.004 in.) at some locations of some rods. Decrudding resulted in decreases in $\text{o.d.}_{\text{max}} - \text{o.d.}_{\text{min}}$ in some cases and increases in others.

A surface roughness, which varied in degree from rod to rod and with axial location and with radial orientation on a given rod, was observed after the six rods had been decrudded. Portions of the rod surfaces were primarily white or light gray after the decrudding operation, while other areas had a generally darker appearance, sometimes in combination with fine, white speckling. The darker areas had a brownish color.

Four of the rods showed smooth, dark brown circumferential rings spaced at pellet-length intervals, and concentrations of small, blister-like protrusions arranged in a series of narrow parallel rings were noted on three rods. Marks where the SA-1 subassembly spacer springs had contacted the cladding surface were apparent on most of the rods. They did not appear to indicate cladding penetrations any deeper than those measured in the initial phase of the program (i.e., ~ 0.005 cm or 0.002 in.). Two rods contained small pit-like penetrations up to 0.010 or 0.012 cm (0.004 or 0.005 in.) deep.

Neutron radiographs were taken of six rods in which possible cracks, small pit-like penetrations, small blisters, or discolored areas were noted during visual examination of the external surfaces. Nothing significant was detected on the neutrographs in these particular regions. The neutrograph of one rod, however, did show a ~ 0.1 cm (0.04 in.)-long, rather faint dark gray spot on the tube wall which might have been caused by a relatively high concentration of hydrogen. Further investigation of this region is planned.

Nine of the fuel rods were punctured, and the contained free gases were collected. They have not yet been analyzed for composition or fission gas release. The gas contents ranged from 5.1 to 12.5 cm³. By comparison, rods examined in the initial phase of the program contained 4.7 to 9.5 cm³ of gas. The maximum fission gas release shown by any of these earlier rods was 0.9 percent.

Seven cladding samples from some of the fuel rods that were removed from SA-1 in January 1967 were analyzed for tritium. The tritium contents ranged from approximately 5 to 58 ppb, equivalent to about 2 to 16 percent of the tritium generated during operation. The samples contained 500 to 920 $\mu\text{L/gm}$ (45 to 82 ppm) total hydrogen and, in addition, 6 to 108 $\mu\text{L/gm}$ CO, 3 to 26 $\mu\text{L/gm}$ CO₂, 15 to 20 $\mu\text{L/gm}$ nitrogen, up to 5 $\mu\text{L/gm}$ oxygen, and up to 2 $\mu\text{L/gm}$ CH₄. Krypton-85 analyses also were performed primarily to qualitatively determine the detectability of this isotope in cladding samples.

3. EXAMINATIONS AND ANALYSES OF FUEL RODS

3.1 FUEL ROD GAMMA SCANNING (C. P. Ruiz, Technical Director, J. P. Peterson, F. R. Smith)

Procedure

Axial gamma scan data were obtained on 41 rods by use of a Ge(Li) detector system. The system used was similar to that previously described,¹ but different components were used (Figure 1). A 40cc detector was placed in front of a collimator slit which penetrated through the 91-cm (36-in.) cell wall. Thus, as the fuel rod was drawn past the appropriate end of the slit (located against the interior of the cell wall), the detector received a collimated gamma ray beam.

The pulses from the detector were passed through a solid-state preamplifier, amplifier, and base line restorer. At this point, the pulses were routed to a 1024 multichannel analyzer and to two single-channel analyzers. One of the single-channel analyzers was set to accept pulses at the energy corresponding to a gamma ray from a particular fission product nuclide of interest, while the other single-channel analyzer was set to accept pulses at a slightly higher energy for a measure of the Compton and background distribution. The output pulses from each of these analyzers were fed to scalers and also two linear count rate meters, from which the signals were fed to a strip chart recorder, such that the background signal count rate was subtracted from the total count rate to yield only a net count rate.

The energy region for the single-channel analyzer was set by operating the multichannel analyzer in the coincidence mode. The gating signal was supplied from a given single-channel analyzer. The width of the energy region was set to accept the total photopeak for the signal. A corresponding number of channels was set for the background single-channel analyzer.

Figure 1. Schematic Drawing of Ge(Li) Gamma Scanning System

A typical gamma ray spectrum is shown in Figure 2. Of primary interest was Cs-137 to provide relative burnup data. Data were obtained on all 41 rods scanned by using the signal at 0.662 mev with the background region set at 0.680 mev. To obtain some information on the relative amounts of Pu-239 fission some of the rods were scanned for Ru-106, which by virtue of its fission yield shows an 11-fold enhancement over that for U-235 fission. In addition, data were obtained for Ce-144 (285 days half-life) and Zr-95 (65 days half-life) to provide data on more recent relative rod burnups. For the Ru-106, Ce-144 and Zr-95, the signal gating regions were 0.515, 2.19 (Pr-144), and 0.765 mev (Zr + Nb-95) with the background gating regions set at 0.530, 2.30, and 0.820 mev, respectively.

In the scanning, the rods from both subassemblies were oriented axially as they had operated in Dresden so that the relative gamma activity could be recorded as a function of increasing distance from the bottom of the active fuel zone of that reactor. The rods were pulled past the slit at 6.4 cm (2-1/2 in.) per minute; the speed of the recorder was 1.3 cm (1/2 in.) per minute. The linear count rate meters had a time constant of two seconds. The number of pulses passed by each single-channel analyzer was accumulated by a scaler for the whole axial traverse. The difference in the scaler counts thus yielded the integral count rate for the data obtained on the chart recorder.

For the Ru-106, Cs-137, and Zr-95, a 0.157 cm (0.062 in.) collimator slit-height was used. Because of the relatively weak intensity of the high energy Pr-144 line, the slit was opened to 0.64 cm (0.25 in.). To limit the count rate to 5×10^4 cps, a 0.64 cm lead filter was used with 0.102 cm (0.040 in.) cadmium interposed between the lead and the detector. The width of the slits used in both cases was 1.90 cm (0.75 in.).

To obtain data on the reproducibility of the scans, rod A37 was selected as a standard rod which was scanned at least once a day throughout the scan period. In general, the reproducibility was within 3%.

Results

Gamma scan traces for each of the four fission products are shown in Figures 3 through 10 for one of the highest exposure rods from the bottom subassembly (A37) and for a rod from the top subassembly (B86). These are fairly representative of the traces of the rods scanned in each respective bundle. It is seen that significant differences appear on comparisons of the gamma scan traces obtained from the four fission products. The fission product Cs-137 is a good integrator of total exposure by virtue of its long half-life, and its fission yield is not very sensitive to the fissioning species. However, it is relatively volatile; hence, the question of migration exists. If migration occurs in the axial direction, it could distort the axial exposure profile. Although future experimental work will be directed toward answering this question by destructive burnup analyses at discrete axial locations in the highest exposure rods, based on other evidence, it is thought that axial migration is probably not extensive for this fuel. Hence, this distribution is proportional to burnup. Apparent differences between the count rates of the A and B rods are not significant because of source-detector geometry changes. The slight depressions seen in the gamma scan traces are due to the spacers.

A comparison of the Ru-106 profile with that of Ce-144 is interesting. Both nuclides have roughly the same half-lives—365 days for Ru-106 and 285 days for Ce-144. However, the thermal fission yields for U-235 and Pu-239 are very different for the two nuclides; 5.4% and 3.8%, respectively, for Ce-144, and 0.4% and 4.4% for Ru-106. Hence, the axial shape of the Ru-106 is proportional approximately to the number of Pu-239 fissions. (The fission from U-238 also contributes to this spectrum since its fission yield is similar to that of Pu-239.) The Ce-144 axial shape is roughly the sum of the fission rates of both U-235 and Pu-239 during the same effective exposure period; i.e., approximately the last two years of irradiation. Finally, the Zr-95 curve shows the axial rod power distribution during the last period of exposure by virtue of its 65-day half-life.

All data were obtained using A37 as a standard rod; hence, this rod was used for normalization. The integrals of the count rate curves for all rods in which gamma scan measurements were made are given in Figures 11 and 12. Although trends are somewhat perturbed by the varying enrichments and rod vacancies within the subassemblies, the relative burnup (based on Cs-137) is still seen to correlate with the enrichment and position of the rods within the bundles. A somewhat different pattern emerges based on the other fission products, which can be interpreted qualitatively in the same way as previously described with regard to the axial traces. The error in each of the values is estimated to be $\pm 2\%$ (1σ). The reason for the relatively low value for Ru-106 and high value of Ce-144 for rod A-47 is not yet understood.

Based on the Cs-137 axial gamma scan data, the peak to average values were obtained (Table 1). It is seen that the values are slightly higher for the top subassembly than for the bottom subassembly. Hence, these values are consistent with data obtained on the ten rods that were scanned with a NaI(Tl) detector system during the initial phase of the program; however, the magnitude of the difference between the two subassemblies is now somewhat lower.

Figure 2. Typical Gamma Ray Spectrum
Obtained with the 40 cc Ge(Li)
Detector on a 1024 Multichannel
Analyzer

Figure 3. Gamma Scan of Rod A37 for Cs-137

Figure 4. Gamma Scan of Rod B86 for Cs-137

Figure 5. Gamma Scan of Rod A37 for Ru-106

Figure 6. Gamma Scan of Rod B86 for Ru-106

Figure 7. Gamma Scan of Rod A37 for Pr-144

Figure 8. Gamma Scan of Rod B86 for Pr-144

Figure 9. Gamma Scan of Rod A37 for Zr/Nb-95

Figure 10. Gamma Scan of Rod B86 for Zr/Nb-95

ROD
CONTROL

(ROD ARRANGEMENT SHOWN IS FOR THE LAST TWO OF FOUR CYCLES OF OPERATION)

1 2.8%			11 3.5%			7 2.8%
0.95 -			1.00 -			0.97 -
- -			- -			- -
	90 3.5%					
	0.98 -					
	- -					
15 2.8%			18 3.5%			
0.88 -			0.94 -			
- -			- -			
22 2.8%				26 3.5%		27 3.5%
0.88 -				0.94 -		0.96 -
- -				- -		- -
			32 3.2%	33 3.5%	34 3.2%	35 2.8%
			0.90 -	0.86 0.98	0.87 0.94	0.87 0.95
			- -	0.94 1.03	0.95 1.04	0.99 1.01
	37 3.5%	38 3.2%	39 3.5%	40 3.2%	9 3.5%	42 2.8%
	1.00 1.00	0.88 0.97	0.96 0.96	0.88 0.96	0.97 1.00	0.93 0.93
	1.00 1.00	0.93 0.97	0.94 0.97	0.93 0.98	0.99 1.05	0.99 0.93
43 2.8%				47 2.8%	48 2.8%	49 2.8%
0.93 -				0.90 1.02	0.91 0.86	0.95 0.90
- -				0.69 0.89	0.86 0.89	1.02 0.95

ROD NO.
W/O ²³⁵ U
¹³⁷ Cs ¹⁴⁴ Ce
¹⁰⁶ Ru ⁹⁵ Zr

Figure 11. Relative Fission Product Activity Values Obtained From Rods From Subassembly A-1 (Bottom)

**CONTROL
ROD**

(ROD ARRANGEMENT SHOWN IS FOR THE LAST TWO OF FOUR CYCLES OF OPERATION)

[illegible]

ROD NO. 235 U
W/O
137Cs 144Ce
106Ru 95Zr

Figure 12. Relative Fission Product Activity Values Obtained From Rods From Subassembly B-1 (Top)

TABLE 1

PEAK-TO-AVERAGE BURNUP RATIOS FOR SA-1 FUEL RODS BASED ON Cs-137

Subassembly A-1		Subassembly B-1	
Rod No.	Burnup Ratio (Peak/Average)	Rod No.	Burnup Ratio (Peak/Average)
A1	1.06	B50	1.13
A7	1.11	B52	1.10
A9	1.09	B56	1.12
A11	1.10	B57	1.11
A15	1.11	B58	1.16
A18	1.12	B60	1.16
A22	1.09	B61	1.10
A26	1.09	B62	1.14
A27	1.11	B83	1.08
A32	1.11	B85	1.12
A33	1.07	B86	1.13
A34	1.07	B89	1.06
A35	1.07	B91	1.15
A37	1.10	B92	1.16
A38	1.08	B93	1.15
A39	1.08	B95	1.11
A40	1.09	B96	1.15
A42	1.08	B98	1.17
A43	1.06		
A47	1.11		
A48	1.09		
A49	1.08		
B90	1.13		

3.2 ROD DIAMETER MEASUREMENTS (G. P. Ferguson)

Procedure

A hot cell profilometer was used to measure diametral variations along the lengths of six fuel rods. Three scans were made along the entire length of each rod in the as-received condition. One scan was made with the rod oriented to measure the diameter in the 0–180° plane; a second was made to measure the diameter in the 90–270° plane; and a third scan was made while the rod was rotating. The part of the rod that was pointed toward the "identification mark side" of the assembly during the last two operating cycles was taken as the 0° orientation in all cases. All scans were made from the short end plugs to the long end plugs of the rods.* The above procedure was repeated on the same six rods after the crud had been removed by dissolving it with hydrochloric acid.

The rods selected for diameter measurements are listed in Table 2. In general, these rods had comparatively high combinations of exposure and fuel density, and thus would tend to demonstrate relatively high degrees of fission product-induced UO₂ swelling.

* The SA-1 rods are sealed with a 0.952 cm (0.375 in.) long plug at one end and a 3.492 cm (1.375 in.) long plug at the other. A 4.42 cm (1.74 in.) long plenum is adjacent to the longer plug, while a standard pellet lies next to the shorter plug. The longer plugs and plenums were pointed upward in the top subassembly (B-1) and downward in the bottom subassembly (A-1). In Sections 3.2, 3.3, and 3.4 of this report, the axial positions of points of interest on the surface of either an "A" rod or a "B" rod are defined in terms of distance from the shoulder of the shorter end plug.

TABLE 2
RODS SELECTED FOR DIAMETER PROFILOMETRY

Rod	Relative Rod-Average Burnup	Peak Burnup Average Burnup	As-Fabricated Fuel Density (Percent of Theoretical)	
			Smeared*	Pellet** Average-Range
A7	0.97	1.11	91.6	95.7 (95.0–96.4)
A9	0.97	1.09	91.6	94.7 (94.2–95.3)
A22	0.88	1.09	91.3	95.1 (95.1–96.0)
A37	1.00	1.10	90.8	94.1 (92.3–95.3)
A49	0.95	1.08	91.5	95.3 (94.4–96.2)
B50	0.91	1.13	92.7	96.0 (95.5–96.6)

*Smeared density (percent of theoretical) is defined as:

$$\frac{\text{Weight UO}_2 \text{ in rod (gm)} \times 100}{\pi/4 \times (\text{cladding ID in cm})^2 \times (\text{fuel column length in cm}) \times 10.97}$$

**Water immersion density on sampling of approximately 2 percent.

Based on measurements made on samples from the various tubing lots, the average and median pre-irradiation inside diameters of the cladding were 0.9665 cm (0.3805 in.). The measurements were within a range of 0.965 to 0.968 cm (0.380 to 0.381 in.). The nominal pre-irradiation pellet diameter was 0.955 ± 0.000 cm (0.376 ± 0.000 in.). The diameter of each pellet in rod A22 was measured before it was loaded into the tube. These pellet diameters ranged from 0.945 to 0.952 cm (0.372 to 0.375 in.). Thirty-nine of the 61 pellets were 0.950 cm (0.374 in.) and 18 were 0.952 cm.

Results

The diameter profiles of the six rods are shown in Figures 13 through 18. The general thickness of the crud on the rods was determined by comparing the rotational traces before and after the rods were decruded. As Table 3 indicates, the rod-average thicknesses ranged from 0.00084 cm (0.00033 in.) for rod A37 to 0.00173 cm (0.00068 in.) for rod B50. Rod A7 showed a local maximum crud thickness of 0.0028 cm (0.0011 in.), which was the highest value among the rods. The thickest crud on most rods was located between the short end plug and midlength; i.e., on the higher power half of the rod (rod A22 represented an exception to this general condition). These six rods had operated about 1500 days in Dresden. Similar measurements on six rods that had operated in Dresden 840 days showed an average crud thickness of about 0.0009 cm (0.00035 in.). The maximum thickness averaged over the length of any of the previously measured rods was 0.0012 cm (0.0005 in.).²

Prior to any irradiation, the diameter of each rod was measured to the nearest ~0.001 cm (0.0005 in.) with a micrometer at the top, middle, and bottom; the exact locations of these measurements are not known.

Figure 13. Diameter Profiles of Fuel Rod A7

DISTANCE FROM SHORT END PLUG SHOULDER

INCHES

PRE-IRRADIATION ROD DIAMETERS AT THREE AXIAL LOCATIONS (cm [in.])

1.080(0.425)

1.082(0.426)

1.080(0.425)

Figure 14. Diameter Profiles of Fuel Rod A9

Figure 15. Diameter Profiles of Fuel Rod A22

PRE-IRRADIATION ROD DIAMETERS AT THREE AXIAL LOCATIONS (cm [in.])

1.078(0.4245)

1.080(0.425)

1.077(0.424)

Figure 16. Diameter Profiles of Fuel Rod A37

PRE-IRRADIATION ROD DIAMETERS AT THREE AXIAL LOCATIONS (cm [in.])

1.078(0.4245)

1.080(0.425)

1.080(0.425)

Figure 17. Diameter Profiles of Fuel Rod A49

PRE-IRRADIATION ROD DIAMETERS AT THREE AXIAL LOCATIONS (cm [in.])

1.080(0.425)

1.080(0.425)

1.080(0.425)

Figure 18. Diameter Profiles of Fuel Rod B50

TABLE 3
CRUD THICKNESS BY PROFILOMETRY

Rod	Circumferential-Average Crud Thickness (cm [in.])		Location(s) of Thickest Crud (cm [in.] from Short End Plug)	Remarks
	Rod-Average	Maximum*		
A7	0.00140 (0.00055)	0.0028 (0.0011)	41 to 51 (16 to 20) 23 to 30 (9 to 12)	Only 0.0008 to 0.001 cm (0.0003 to 0.0004 in.) over most of that part of rod between mid-length and long end plug.
A9	0.00099 (0.00039)	0.0020 (0.0008)	13 to 20 (5 to 8)	Decreases toward long end plug. Only 0.0005 to 0.0008 cm (0.0002 to 0.0003 in.) over most of that part of rod between mid-length and long plug.
A22	0.00127 (0.00050)	0.0018 (0.0007)	58 to 76 (23 to 30)	Fairly uniform thickness over full length with most values between 0.0010 and 0.0015 cm (0.0004 and 0.0006 in.); however, crud is slightly thicker at indicated location than elsewhere on rod.
A37	0.00084 (0.00033)	0.00165 (0.00065)	8 to 18 (3 to 7)	Gradually decreases to 0.00025 to 0.0005 cm (0.0001 to 0.0002 in.) toward long end plug.
A49	0.00112 (0.00044)	0.0020 (0.0008)	—	Ranges from about 0.0013 to 0.0020 cm (0.0005 to 0.0008 in.) in fairly random manner between short plug and mid-length. Drops off to about 0.0005 cm (0.0002 in.) over rest of rod.
350	0.00173 (0.00068)	0.0024 (0.00095)	10 to 23 (4 to 9)	Tends to decrease somewhat toward long end plug, but thickness is still 0.00165 to 0.0019 cm (0.00065 to 0.00075 in.) in some places near long plug.

* Largest mean value for any 2.5 cm (1 in.) increment along rod.

As Table 4 indicates, all rods showed some diameter increase with respect to the pre-irradiation measurements. The rod-average increases ranged from 0.0008 cm (0.0003 in.) on rod A37, the highest exposure rod, to 0.0046 cm (0.0018 in.) on rod B50. The greatest increases in diameter, up to about 0.0075 cm (0.003 in.) on rods A7, A49, and B50, occurred near the short end plugs on most rods between the ~2.5 cm (1 in.) location and the 5 cm (2 in.) or 7.5 cm (3 in.) locations. However, if one considers full thirds of the rod length over the fuel column, all rods except A22 showed the greatest average diameter increase over the section that includes the long end plug and plenum and which operated at a power somewhat below the average for a given rod. The lowest diameter increases took place over the middle thirds of the rods, again with the exception of rod A22. During the initial phase of the program, six rods were similarly decrudded and measured. These earlier rods operated 650 fewer days than the present ones and had an average burnup of 6×10^{20} fissions/cm³ (22,000 MWd/tU), which is estimated to be 2×10^{20} fissions/cm³ less than the present rods. The largest average diameter increase above pre-irradiation measurements shown by any of these earlier rods was only 0.0005 cm (0.0002 in.) lower than that shown by rod B50, and the average increase for all six rods was 0.0025 cm (0.0010 in.); i.e., the same as that of the present group.²

TABLE 4
ROD DIAMETER AFTER DECRUDDING

Rod	Distance From Short End Plug Shoulder (cm [in.])	Rod Diameter (cm [in.])		
		Post-Irradiation (Average)	Pre-Irradiation	Change
A7	0.95–30.5 (0.375–12)	1.0752 (0.4233)	1.0732 (0.4225)	+0.0020 (0.0008)
	30.5–63.5 (12–25)	1.0726 (0.4223)	1.0719 (0.422)	+0.0007 (0.0003)
	63.5–94.0 (25–37)	1.0744 (0.4230)	1.0706 (0.4215)	+0.0038 (0.0015)
	Average			+0.0023 (0.0009)
A9	0.95–30.5 (0.375–12)	1.0825 (0.4262)	1.0795 (0.425)	+0.0030 (0.0012)
	30.5–63.5 (12–25)	1.0825 (0.4262)	1.0820 (0.426)	+0.0005 (0.0002)
	63.5–94.0 (25–37)	1.0828 (0.4263)	1.0795 (0.425)	+0.0033 (0.0013)
	Average			+0.0023 (0.0009)
A22	0.95–30.5 (0.375–12)	1.0795 (0.4250)	1.0770 (0.424)	+0.0025 (0.0010)
	30.5–63.5 (12–25)	1.0787 (0.4247)	1.0770 (0.424)	+0.0017 (0.0007)
	63.5–94.0 (25–37)	1.0772 (0.4241)	1.0770 (0.424)	+0.0002 (0.0001)
	Average			+0.0015 (0.0006)
A37	0.95–30.5 (0.375–12)	1.0792 (0.4249)	1.0782 (0.4245)	+0.0010 (0.0004)
	30.5–63.5 (12–25)	1.0795 (0.4250)	1.0795 (0.425)	0.0000 (0.0000)
	63.5–94.0 (25–37)	1.0790 (0.4248)	1.0770 (0.424)	+0.0020 (0.0008)
	Average			+0.0008 (0.0003)
A49	0.95–30.5 (0.375–12)	1.0828 (0.4263)	1.0795 (0.425)	+0.0033 (0.0013)
	30.5–63.5 (12–25)	1.0803 (0.4253)	1.0782 (0.4245)	+0.0021 (0.0008)
	63.5–94.0 (25–37)	1.0851 (0.4272)	1.0795 (0.425)	+0.0056 (0.0022)
	Average			+0.0036 (0.0014)
B50	0.95–30.5 (0.375–12)	1.0846 (0.4270)	1.0795 (0.425)	+0.0051 (0.0020)
	30.5–63.5 (12–25)	1.0823 (0.4261)	1.0795 (0.425)	+0.0028 (0.0011)
	63.5–94.0 (25–37)	1.0851 (0.4272)	1.0795 (0.425)	+0.0056 (0.0022)
	Average			+0.0046 (0.0018)

The difference between the maximum diameter and the minimum diameter at a given point along the rod axis ($\text{o.d.}_{\text{max}} - \text{o.d.}_{\text{min}}$) was as large as 0.0075 or 0.010 cm (0.003 or 0.004 in.) at some locations of some rods. The general magnitude of this difference varied considerably among different rods (for example, compare rod B50 in Figure 18 with rod A9 in Figure 14); and the portion of a given rod where $\text{o.d.}_{\text{max}} - \text{o.d.}_{\text{min}}$ was greatest varied considerably from rod to rod. For instance, this difference was largest over the short end plug, high power, half of rod A22 (Figure 15), but on rod A49 $\text{o.d.}_{\text{max}} - \text{o.d.}_{\text{min}}$ was low over the maximum power region in comparison with almost any other location.

After the rods were decrudded, some portions of rods B50 and A22 between the plenum region and midlength showed decreases in $\text{o.d.}_{\text{max}} - \text{o.d.}_{\text{min}}$; up to approximately 0.0025 cm (0.001 in.) on the former rod and about 0.0012 cm (0.0005 in.) on the latter. In some cases, however, decrudding resulted in increases in $\text{o.d.}_{\text{max}} - \text{o.d.}_{\text{min}}$; up to nearly 0.0025 cm over portions of rod A7. In general, rods A9 and A37 probably showed the least changes in $\text{o.d.}_{\text{max}} - \text{o.d.}_{\text{min}}$.

In addition to the general increase in diameter that occurred during operation, local peaks in rod diameter often were present near spacer locations. The SA-1 subassembly spacer springs contacted the rods at approximately 3.8 cm (1-1/2 in.), 35.7 cm (14-1/16 in.), 67.6 cm (26-5/8 in.), and 99.7 cm (39-1/4 in.) from the short end plug shoulder. The VBWR spacer wires were located about 51.4 cm (20-1/4 in.) from the short plug.

All of the rods exhibited local increases in diameter at the 3.8 cm subassembly spacer location and less frequently at the other subassembly locations. These peaks were generally 0.0025 to 0.005 cm (0.001 to 0.002 in.) above the typical rod diameter in the vicinity. Those near the 99.7-cm location were 0.005 to 0.0075 cm (0.002 to 0.003 in.), but a force fit between the long end plug and tube may be responsible for at least some of the diameter peaking noted near this location. Chemical removal of the crud from the six rods decreased the rod diameter at the spacer peaks in almost all cases. The average and maximum amounts of decrease at the 3.8-cm, 35.7-cm, and 67.6-cm locations taken as a group were 0.0030 cm (0.0012 in.) and about 0.0051 cm (0.002 in.), respectively. Thus, the maximum thickness of crud on some part of the rod circumference might have been as much as 0.0051 cm or as little as one-half of this value depending upon the evenness of the crud layer around the rod. A comparison of the diameters at the SA-1 spacer locations with those of the adjoining regions indicated that the crud was relatively thicker at the 35.7-cm and 67.6-cm peaks (possibly by as much as 0.0025 cm [0.001 in.]) but that at the 3.8-cm peaks the crud was usually thinner than that over the adjoining regions. Hence, on the diameter profiles produced after the rods had been decrudded, local increases in diameter could still be seen at most of the 3.8-cm spacer locations and at some of the other subassembly spacer locations as well. The difference between diameter at the spacer peaks and those of the corresponding adjoining regions on these profiles were at least 0.0025 cm (0.001 in.) at seven spacer locations and showed a maximum of 0.0051 cm (0.002 in.) on rod A7 at 3.8 cm (Figure 13).

All rods showed peaks in their diameter profiles at the VBWR spacer locations. Such peaks were 0.007 to 0.008 cm (~0.003 in.) high on rods A7, A49, and B50 (Figures 13, 17, and 18). Decrudding the six rods resulted in an average diameter decrease of 0.0025 cm and a maximum decrease (on rod A7) of 0.0056 cm (0.0022 in.) at these locations. Thus, the maximum thickness of crud on some part of the circumference was between 0.0028 cm and 0.0056 cm. In one or two instances the crud at the VBWR spacer location was possibly 0.001 cm (0.0004 in.) thicker than the crud on either side of the peak; however, the crud was relatively thinner at the spacer location more often than not. Hence, peaks were apparent at this location on all profiles produced after the rods were decrudded. The average and maximum height of these peaks were 0.0061 cm (0.0024 in.) and 0.0079 cm (0.0031 in.), respectively.

During the initial phase of the program, one rod was sectioned through both a subassembly spacer location and through a VBWR spacer wire location. The diameter profile of this rod run after it was decrudded showed peaks at both locations similar to those shown by profiles of the present rods. Metallographic examination of these sections showed increased cladding corrosion near spacer spring contact points and proximate to a groove worn in the tube by the spacer wire.³ Hence, at least some of the peaks in the current profiles might have been caused by a relatively higher degree of corrosion in these spacer regions.

The stationary, "decrudded" diameter profiles of rod A37 showed peaks at pellet-length intervals (~2.2 cm) [7/8 in.] between 28 and 38 cm (11 and 15 in.) from the short end plug shoulder and between the 56- and 61-cm (22- and 24-in.) locations (Figure 16). Diameter peaking of this type might have been caused by mechanical interaction between the pellets and cladding. The difference in diameter between these peaks and the corresponding troughs was typically 0.0012 cm (0.0005 in.) to 0.0019 cm (0.00075 in.) and had a maximum value of about 0.0025 cm (0.001 in.). The rod A22 profiles showed some similar peaks at pellet-length intervals of about 1.5 cm (0.6 in.) between the short end plug and midlength and also between the ~69- and 74-cm (27- and 29-in.) locations (Figure 15). The average height of such peaks on this rod was

somewhat greater than 0.0012 cm, and the maximum was 0.0019 cm. A few scattered and rather ill-defined peaks were shown by the profiles of rod A9 over the short end plug half of the rod (Figure 14), and a few small peaks were barely discernible within the same region of rod A7 (Figure 13). The length of the pellets in the former rod is ~ 2.2 cm and in the latter ~ 1.5 cm. The 0-180° and 90-270° profiles of rods A49 and B50 were so jagged that it was difficult to tell whether or not there was any particular diameter peaking at pellet-length intervals on these rods. The rod A49 traces, however, did show some sharp and easily discernible troughs at pellet-length intervals, particularly between the 84- and 91-cm (33- and 36-in.) locations (Figure 17). Some of these troughs were as deep as 0.004 to 0.005 cm (~ 0.0015 to 0.002 in.). Other troughs that were somewhat less well-defined were observed between the 74- and 84-cm (29- and 33-in.) locations and between the 14- and 24-cm (5-1/2- and 9-1/2-in.) locations. The cause of these troughs is not certain. They may be the result of a possible decrease in cladding corrosion at pellet interfaces.

Rather pronounced local decreases in diameter were noted on the as-received profile traces of rods A7 and A49 near some of the points of contact with the subassembly corner angle recesses. These recesses were centered at 3.8, 11.7, 19.7, 27.6, 35.8, 43.7, 51.6, 59.5, 67.7, 75.6, 83.5, and 91.4 cm (1-1/2, 4-5/8, 7-3/4, 10-7/8, 14-3/32, 17-3/16, 20-5/16, 23-7/16, 26-21/32, 29-3/4, 32-7/8, and 36 in.) from the short end plug shoulders. On the profile traces run after the rods were decrudded, no depressions of this sort could be detected on rod A7, and only a few of relatively lower prominence could be seen on rod A49.

3.3 VISUAL EXAMINATION OF DECRUDDED RODS (F. H. Megerth)

In the preceding quarterly progress report it was mentioned that a surface roughness was noted to varying degrees on portions of nearly all fuel rods and that it was probably a consequence of an uneven, pitting-type ZrO_2 corrosion layer.⁴ After carrying out the chemical decrudding operation on six rods, this roughness had the appearance of small, closely spaced blisters, as shown in Figure 19.

Figure 19. Rough Surface on Rod B50 After Crud Had Been Chemically Removed

The roughness on a given rod generally varied somewhat with radial orientation. Although there was usually some variation along the length of a rod, this variation did not appear to be related to the power profiles; i.e., the surfaces of most of the six rods were about as rough at the low power ends as they were at the high power ends. There were noticeable differences in the extent and degree of roughness among different rods, although the burnups or average power ratings of most of the rods were similar. Corner rods A49 and B50 showed the roughest surfaces of the six rods; the surface of corner rod A7 was about average with respect to roughness. Rods A49 and B50 were fabricated from the same tubing lot (CA-2), but rod A9, which was relatively smooth, also was fabricated from this lot. Portions of the rod surfaces were primarily white or light gray after the decrudding operation (Figures 19 and 20), while other areas had a generally darker appearance (Figure 21), sometimes in combination with fine, white speckling. The darker areas had a brownish color. This coloration suggests that a thin film of crud might have been retained over these areas at the end of the decrudding operation.

Four of the rods (A7, A22, A49, and B50) showed smooth, dark brown circumferential rings spaced at pellet length intervals. The surface usually had a speckled light gray appearance between these rings (Figure 22). These rings were located between the 69-cm (27-inch) locations and the plenums; i.e., in relatively low power regions, on the "A" rods, but they were located on the higher power half of rod B50. The locations of the rings on rod A49 correspond with the locations of some of the sharp troughs noted on the diameter profiles of this rod, but A49 was the only rod of the four that showed these deep, well-defined troughs at pellet-length intervals.

Concentrations of small, blister-like protrusions arranged in a series of narrow parallel rings were noted on rod A9 at about 46 cm (18-1/8 in.) from the short end plug shoulder, on rod A37 at 45 cm (17-3/4 in.), and on rod A7 at about 55 cm (21-1/2 in.). Such a series consisted of up to approximately 5 rings over 0.6 to 1.0 cm (1/4 to 3/8 in.) of rod length. Although similar appearing "rough rings" were seen on a number of other rods in the as-received condition,⁴ the rings on the above three rods apparently were not observed until after the crud had been removed from them. As was pointed out in the preceding progress report, these rings of blisters were found to be regions where the amount of pitting-type cladding corrosion was greater than typical of the general area.

Figure 23 shows a ~0.25-cm (0.1-in.) long oval-shaped mark on rod B50 near a subassembly spacer spring location. What appears to be an area of increased corrosion borders this mark (note the light-colored ring around it). However, it was impossible to determine conclusively just from visual examination of the rod surface if any significant cladding penetration had occurred through either mechanical wear or corrosion. Similar, but less prominent, markings were noted near other spring-rod contact locations, both on rod B50 and on four additional rods, and near a few areas of contact with corner angle recesses on rods B50 and A49. Two shallow 0.16 by 0.16 cm (1/16 by 1/16 in.) gouges were observed at spacer band locations on rod A22. During the initial phase of the examination program, a relative increase in the thickness of the external ZrO_2 layer was found near some spring-rod contact points.³ These later markings did not appear to indicate cladding penetrations any deeper than those measured previously.

Two of the rods (A9 and A22) showed localized increases in surface roughness near those points where the wire-type spacers used during VBWR operation contacted the rods. With two other rods (A49 and B50) the ZrO_2 layers were fairly smooth near the VBWR spacer locations, but they contained fine surface cracks around the wire-rod contact points and appeared to be thicker than normal over areas that extended ~1 cm or 1/2 inch on either side of these contact points (Figure 24).

All of the rods with the possible exception of A22 had linear markings, primarily of longitudinal orientation, on their decrudded surfaces. They varied in length from 0.2 to 0.3 cm (~1/8 in.) to 10 to 20 cm (~4 to 8 in.). Most, if not all, of these markings appeared to be scratches that resulted from either fabrication or post-irradiation handling. Examples of this type of marking are shown in Figure 25.

Rod A9 contained a ~0.05 by 0.025 cm (~0.02 by 0.01 in.) oval-shaped pit 0.010 to 0.012 cm (0.004 to 0.005-in.) deep located 71.4 cm (28-1/8 in.) from the short end plug shoulder (Figure 26), and rod A7 contained a sharp-edged ~0.075 by 0.025 cm (~0.03 by 0.01 in.) penetration ~0.005 cm (0.002 in.) deep at 57.5 cm (22-5/8 in.).

3.4 NEUTRON RADIOGRAPHY OF FUEL RODS (G. P. Ferguson)

One of the principal values of neutron radiography in the examination of irradiated fuel is the ability to detect regions of high hydride concentration in the cladding, which may be indicative of a defective fuel rod. These regions show up as dark areas on positive prints produced from the exposed film.

Figure 20. Peak Burnup Region of Rod A37

Figure 21. Surface Appearance of Rod A37 84 to 87 cm (33 to 34 in.) From Short End Plug Shoulder

Figure 22. Smooth Rings at Pellet-Length Intervals on Rod A7

Figure 23. Subassembly Spacer Marking on Rod B50

Figure 24. VBWR Spacer Region on Rod B50

Figure 25. Fine, Linear Penetrations in Surface of Rod B50

Figure 26. Pit in Surface of Fuel Rod A9 71.4 cm (28 1/8 in.) From Short End Plug Shoulder

Neutron radiographs were taken of six rods in which possible cracks, small pit-like penetrations, small blisters, or discolored areas were noted during visual examination of the external surfaces (rods A1, A9, A11, A18, A43, and B52). Nothing that appeared significant was detected on the neutrographs in these particular regions. The neutrograph of rod A9, however, did show a ~ 0.1 -cm (0.04-in.)-long rather faint dark gray spot on the tube wall 29.8 cm (11-3/4-in.) from the short end plug shoulder, which might have been caused by a relatively high concentration of hydrogen (Figure 27). Further investigation of this region is planned.

Gaps in the fuel columns as large as 0.1 to 0.3 cm were noted in about six instances (Figure 28). As had been found in similar examinations conducted during the initial phase of the program, approximately 5 percent of the pellets were fractured or showed marked cracking.¹ However, whereas the previously detected pellet cracking occurred quite at random, ~ 50 percent of the cracked pellets in the present rods were located within 10 cm (4 in.) of the short end plug shoulder and ~ 85 percent were located within 40 cm (16 in.) of it.

3.5 GAS COLLECTION (G. P. Ferguson)

Nine fuel rods with average burnups of approximately 8×10^{20} fissions/cm³ (30,000 MWd/tU) were punctured, and the contained free gases were collected. They have not yet been analyzed for composition or fission gas release. The gas contents ranged from 5.1 to 12.5 cm³ (0.31 to 0.76 in.³). By comparison, rods examined in the initial phase of the program, which had burnups between 5.5×10^{20} and 7×10^{20} fissions/cm³ (20,000 and 26,000 MWd/tU) contained 4.7 to 9.5 cm³ (0.28 to 0.58 in.³) of gas. The maximum fission gas release shown by any of these earlier rods was 0.9 percent.⁵

Six of the punctured rods contained what were believed to be possible defects based on visual examination of their external surfaces. The fact that no difficulties were encountered in maintaining a vacuum on these rods during gas collection tends to indicate that they contained no full-wall cladding penetrations.

3.6 TRITIUM ANALYSIS OF CLADDING SAMPLES (D. E. Rey, L. N. Grossman, F. H. Megerth)

Earlier in the program, seven pairs of short, transverse fuel rod sections were cut from five of the rods that had been removed from Assembly SA-1 in January 1967 (Rods B70, A13, A41, A46, and B97), and the cladding from one of each pair of these rod sections was analyzed for total hydrogen content and other gaseous species.⁶ In the current work, a cladding sample from the second of each pair of sections was analyzed primarily to determine the tritium content. These samples were analyzed also for total hydrogen content and other gaseous species.

Procedure

Sample Preparation. The previous sectioning of the fuel rods was performed in a hot cell of the Radioactive Materials Laboratory. The bulk fuel was removed, and the cladding segments were leached with nitric acid and then rinsed thoroughly with distilled water. After drying, two pieces about 0.95 cm (3/8 in.) square were cut from each cladding segment and placed in appropriately labeled screw-cap glass vials. Seven such vials were shipped to Chemistry in a 110-kg (250-lb) cask.

In the lead fort of the Physical Chemistry Laboratory, one piece of cladding was removed from each vial, visually inspected with mirrors, and then weighed. Each weighed piece of cladding was carried in a lead pig to the shielded vacuum outgassing apparatus and inserted into the loading arm. (Although the previously analyzed samples went through degreasing steps with an organic solvent both immediately before the weighing and after the distilled water rinse in the hot cell, the current samples did not go through these steps.)

Outgassing. The vacuum system was pumped down overnight. The tungsten crucible then was degassed at a temperature between 1800°C and 1900°C. (The crucible was heated by induction with a 10 KVA LEPEL RF generator.) With the system at a final base pressure of less than 10^{-7} torr, each cladding piece was dropped into the crucible and heated for a period of 30 minutes at 1300°C. Evolved gases were moved rapidly from the furnace area by two mercury diffusion pumps working in tandem. The transfer of gases to the measuring section was completed by an automatic toepler pump. At the end of a run, the sample was dumped into the spent sample reservoir by lifting the crucible bail with a magnet. Blanks were run by heating the empty crucible for 30 minutes at 1300°C.

Analysis. The total gas release from each sample of cladding was measured by means of a multirange McLeod gage. After this measurement, the collected gas was aliquoted as follows:

1. About two-thirds of the gas was used for constituent analysis by gas chromatography. The gas chromatographic scans were obtained by temperature programming a 6.1-m (20-ft) Porapak QS column from -60°C to +75°C, using helium as carrier gas. This provided complete resolution and determination of all gas components.
2. The remaining one-third was first scanned for Kr-85 by gamma spectrometry and then sent to the Metallurgy Mass Spectrometry Laboratory for conversion of hydrogen/tritium to water products (H_2O , HTO , T_2O) and residual gas analysis. The conversion was done by addition of a carefully measured amount of oxygen and subsequent combustion with a heated filament. The products of combustion were frozen out with a dry ice/acetone bath and then taken up with scintillator fluid for tritium counting.

Results

The analyses are presented in Table 5, and data on the operation of the different rods in the two reactors are given in Table 6. The tritium content of the cladding ranged from approximately 5 to 58 parts per billion, equivalent to about 2 to 16 percent of the tritium generated during operation. The burnup of the fuel corresponding to the cladding samples analyzed for tritium ranged from 5×10^{20} to 8×10^{20} fissions/cm³ (17,000 to 30,000 MWd/t), and the time-averaged power values ranged from about 30 to 50 W/cm² (100,000 to 160,000 Btu/h-ft²). The maximum power of any rod at a sample location while in Dresden is estimated to have been 80 W/cm² or 250,000 Btu/h-ft². This power was attained relatively early in Dresden operation. The cladding tritium content correlates quite well with the fuel burnup or average power rating over the total operating time. It will be noted that the ratio of cladding tritium content to burnup and time-averaged power increased rather markedly with increasing burnup and power both among samples from different rods and between two samples from an individual rod.

The tritium contents were decay-corrected to the time of discharge, but the krypton-85 values, tabulated in terms of disintegrations per minute, were not corrected for either decay or counting efficiency. The krypton-85 analyses were performed primarily to qualitatively determine the detectability of this isotope in cladding samples.

Figure 27. Neutron Radiographs of Short End Plug Portions of Rods A43, B52, and A9. Dark Gray Spot on Cladding of Rod A9 Is Indicated.

Figure 28. Neutron Radiographs of Short End Plug Portions of Six SA-1 Rods. Fuel Column Gaps and Pellet Cracking Are Indicated.

TABLE 5
TRITIUM CONTENT OF CLADDING SAMPLES

Sample Number and Location (approximate cm [inches] up from Bottom of SA-1 Fuel Column) ^a	Gas Volume (Microliters per Gram at STP)	Tritium Content ^c			Approximate gm ³ H ₂ gm UO ₂ generated (X 10 ⁹)	Percent Tritium in Cladding	Krypton-85 $\frac{\text{dpm}}{\text{gm}} \times 10^{-6}$ ^b
		$\frac{\mu\text{Ci}}{\text{gm Cladding}}$	ppb Cladding	$\frac{\text{gm in Cladding}}{\text{gm UO}_2}$ (X 10 ⁹)			
B70E							
131.76 cm (51-7/8 in.)	1025.2	242.4	24.9	3.99	50	8	12
A13G							
64.14 cm (25-1/4 in.)	844.0	561.4	57.8	9.25	61	15	16
A41E							
68.90 cm (27-1/8 in.)	528.5	553.4	56.9	9.14	57	16	NA
A46U							
3.81 cm (1-1/2 in.)	836.9	43.3	4.5	0.71	33	2	0.75
A46G							
70.80 cm (27-7/8 in.)	870.2	246.4	25.4	4.02	53	8	6.9
B97D							
123.51 cm (48-5/8 in.)	924.4	311.6	32.1	5.14	57	9	8.9
B97K							
173.67 cm (68-3/8 in.)	969.6	92.7	9.5	1.53	36	4	1.1

NA = Not Analyzed

(a) The bottom of the SA-1 fuel column was 58.6 cm (23-1/16 in.) above the bottom of the Dresden Reactor active fuel zone. The height of the SA-1 fuel column was approximately 192 cm (75-5/8 in.), which included a ~3 cm (1-1/4 in.) non-fueled gap between the top and bottom subassemblies.

(b) Not corrected for either counting efficiency or decay. Analyzed in July 1970.

(c) Corrected to date of discharge (January 13, 1967).

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TABLE 6
ROD OPERATING DATA

		Estimated Heat Flux in W/cm ² (Btu/h-ft ² X 10 ⁻³) ^b										
Rod or Sample	Initial Enrichment (weight percent U-235)	Burnup		Days of Operation ^a	Time-Averaged VBWR + Dresden	VBWR Operation	Dresden Operation					
		Rod-Average, or Sample				At Maximum Reactor Power Rod-Average or Sample	Time-Averaged Cycles 3 and 4 Rod-Average or Sample	At Maximum Reactor Power Average for Rod or Sample				
		Fissions/cm ³ X 10 ⁻²⁰	MWD/tU					BOC 3 ^c	EOC 3 ^c	BOC 4	EOC 4	
B70	2.76	5.5	20,200	1,217	35.1 (111)	74 (230)	34 (110)	58 (180)	—	—	—	
B70F		6.6	23,900		41.5 (132)	—	40 (130)	69 (220)	45 (140)	53 (170)	40 (130)	
A13	3.49	7.0	25,600	1,204	44.9 (142)	88 (280)	41 (130)	70 (220)	—	—	—	
A13G		8.1	29,700		52.1 (165)	132 (420)	47 (150)	80 (250)	52 (160)	63 (200)	47 (150)	
A41	3.49	6.6	24,000	1,204	42.0 (133)	88 (280)	36 (110)	63 (200)	—	—	—	
A41E		7.8	28,500		49.8 (158)	132 (420)	43 (140)	74 (230)	48 (150)	57 (180)	43 (140)	
A46	2.76	6.0	22,000	1,218	38.7 (123)	63 (200)	40 (130)	69 (220)	—	—	—	
A46U		4.6	16,900		29.7 (94)	—	30 (100)	52 (160)	34 (110)	41 (130)	31 (100)	
A46G		6.9	25,300		44.5 (141)	95 (300)	46 (150)	79 (250)	51 (160)	61 (190)	46 (150)	
B97	2.76	6.1	22,200	1,218	38.6 (122)	80 (250)	36 (110)	62 (200)	—	—	—	
B97D		7.3	26,600		46.3 (147)	121 (380)	43 (140)	75 (240)	48 (150)	58 (180)	44 (140)	
B97K		5.1	18,500		32.2 (102)	—	30 (100)	51 (160)	34 (110)	41 (130)	30 (100)	

- (a) Total days of operation, 840 days of which were in Dresden in all cases.
Days of operation in the VBWR is defined as hours at temperature (reactor thermal power > 5 MW) divided by 24.
Days of operation in Dresden is defined as hours of turbine-generator operation divided by 24.
- (b) To obtain linear power in kW/ft of rod length, multiply W/cm² by 0.103.
- (c) BOC = Beginning of Cycle. EOC = End of Cycle.

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The values of tritium generation were approximated as follows: The distribution of fissions among U-235, Pu-239, Pu-241, and U-238 were obtained from chemical isotopic analyses. This information was directly obtainable for samples A13G, A41E, and A46G, inasmuch as adjacent fuel samples already had been analyzed.⁵ Fission distributions for the other four samples were estimated from curves of atom percent fission versus burnup generated from the SA-1 data and other chemically determined isotopic data. Yield values of 1.3×10^{-4} for U-235 and U-238 and 2.3×10^{-4} for Pu-239 and Pu-241 were then applied to estimate the production of tritium.

The total volumes of gas evolved from the samples ranged from approximately 500 to 1000 microliters per gram of cladding (Table 7). The chromatographic analyses showed total hydrogen contents of 84 to 95 percent, carbon monoxide contents of 1 to 12 percent, and carbon dioxide contents of 0.5 to 3 percent. The samples also contained 1.6 to 3.4 percent nitrogen, up to 0.6 percent oxygen, and up to 0.2 percent methane. In terms of parts per million, the total hydrogen contents ranged from 45 to 82.

The earlier analyses of samples from rod locations next to those of the above samples are given in Table 8 from comparison. The order in which the samples are listed corresponds to that of Table 7; i.e., sample B70E came from a location next to sample B70F, etc.

For both sets of samples, the total volumes of gas collected ranged from approximately 500 to 1100 microliters per gram of sample. For a given pair of samples, these volumes generally were within $\sim 100 \mu\text{L/gm}$ of each other, and the total volumes collected from the current samples were neither predominately larger nor smaller than the volumes collected from the corresponding earlier samples. Sample B70F, however, contained 250 to 275 $\mu\text{L/gm}$, or approximately one-third, more gas than sample B70E.

The corresponding percentages of total hydrogen were within ~ 5 points of each other, with the exception of the sample pair B97C and B97D. The combined differences in total gas volumes and percentages of hydrogen then resulted in differences in cladding hydrogen contents that ranged from about 3 to 11 parts per million in six of the seven pairs of samples. The total hydrogen content of B70F was 21 to 22 ppm greater than that of B70E.

With most sample pairs, the CO and CO₂ contents of the current samples were lower in terms of microliters per gram than the corresponding earlier samples. However, samples B97D and A13G were exceptions in the case of CO, and B70F and again A13G were exceptions in the case of CO₂.

The presence of oxygen in the current samples, where none was detected previously, indicates air in-leakage. This probably accounts for the higher nitrogen values. Since neither oxygen nor nitrogen could be detected in the blank runs, it is suspected that air contamination occurred during the extra manipulations required in taking aliquots for the tritium determinations.

The determination of tritium should be included as an integral part of the analytical scheme for cladding outgas. With this in mind, the section of the outgassing apparatus for aliquoting and GC entry is going to be modified. Switching valves (in particular) that seal-off and trap pockets of air as they are turned, will be either eliminated or replaced with other devices.

TABLE 7
ANALYSES OF GASES EVOLVED FROM CURRENT CLADDING SAMPLES

Sample Number and Location (approximate cm [inches] up from Bottom of SA-1 Fuel Column) ^a	Total Gas Volume (Microliters per Gram at STP)	Composition (percent by volume)						Volumes of Constituents (microliters per gram at STP)						Total Hydrogen Content (ppm)
		H ₂	O ₂	N ₂	CH ₄	CO	CO ₂	H ₂	O ₂	N ₂	CH ₄	CO	CO ₂	
B70F 131.76 cm (51-7/8 in.)	1025.2	89.99	0.40	1.68	0.03	6.36	1.54	922.6	4.1	17.2	0.3	65.2	15.8	82.3
A13G 64.14 cm (25-1/4 in.)	844.0	84.92	0.13	2.16	ND	10.32	2.47	716.7	1.1	18.2	<0.1	87.1	20.8	64.0
A41E 68.90 cm (27-1/8 in.)	528.5	94.81	0.02	3.44	0.01	1.21	0.51	501.1	0.1	18.2	0.1	6.4	2.7	44.7
A46U 3.81 cm (1-1/2 in.)	836.9	86.86	0.42	1.79	0.19	7.66	3.06	726.9	3.5	15.0	1.6	64.1	25.6	64.9
A46G 70.80 cm (27-7/8 in.)	870.2	85.60	0.58	2.30	ND	10.03	1.49	744.9	5.0	20.0	<0.1	87.3	13.0	66.5
B97D 123.51 cm (48-5/8 in.)	924.4	83.73	0.48	1.88	0.04	11.69	2.19	774.0	4.4	17.4	0.4	108.1	20.2	69.1
B97K 173.67 cm (68-3/8 in.)	969.6	89.73	0.33	1.59	0.12	5.69	2.54	870.0	3.2	15.4	1.2	55.2	24.6	77.6

ND = Not Detected (<0.01%)

(a) The bottom of the SA-1 fuel column was 58.6 cm (23-1/16 in.) above the bottom of the Dresden Reactor active fuel zone. The height of the SA-1 fuel column was approximately 192 cm (75-5/8 in.), which included a ~3 cm (1-1/4 in.) non-fueled gap between the top and bottom subassemblies.

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TABLE 8
PREVIOUSLY DETERMINED GAS CONTENTS OF SAMPLES LOCATED
ADJACENT TO THE CURRENT CLADDING SAMPLES

Sample Number	Total Gas Volume (Microfilters per Gram at STP)	Composition (percent by volume)						Volumes of Constituents (microliters per gram at STP)						Total Hydrogen Content (ppm)
		H ₂	O ₂	N ₂	CH ₄	CO	CO ₂	H ₂	O ₂	N ₂	CH ₄	CO	CO ₂	
B70E (a)	750	88.3	ND	0.9	ND	10.8	—	662	≤1	7	≤1	81	—	60
(b)	775	—	—	—	—	—	—							
(c)	760	90.0	ND	0.7	ND	9.0	0.3	684	≤1	5	≤1	68	2	61
A13F (a)	703	89.7	ND	ND	T	10.3	—	631	≤1	≤1	<1	72	—	57
(b)	730	—	—	—	—	—	—							
(c)	698	87.2	ND	ND	T	10.6	2.2	609	≤1	≤1	<1	74	15	54
A41D (a)	622	89.9	ND	ND	T	10.1	—	559	≤1	≤1	<1	63	—	50
(b)	615	—	—	—	—	—	—							
(c)	650	89.6	ND	ND	T	5.2	3.4	582	≤1	≤1	<1	34	22	52
A46T (a)	745	88.0	ND	ND	T	12.0	—	656	≤1	≤1	<1	89	—	59
(b)	749	—	—	—	—	—	—							
(c)	784	85.7	ND	ND	T	10.3	5.7	672	≤1	≤1	<1	81	45	60
A46F (a)	960	87.4	ND	1.0	T	11.6	—	839	≤1	10	<1	111	—	75
(b)	986	—	—	—	—	—	—							
(c)	886	88.2	ND	0.2	T	7.4	4.2	781	≤1	2	<1	66	37	70
B97C (a)	894	90.5	ND	ND	T	9.5	—	809	≤1	≤1	<1	85	—	73
(b)	890	—	—	—	—	—	—							
(c)	933	89.4	ND	ND	T	7.3	3.3	834	≤1	≤1	<1	68	31	74
B97J (a)	1095	90.0	ND	0.8	T	9.2	—	986	≤1	9	<1	101	—	89
(b)	1095	—	—	—	—	—	—							
(c)	1061	87.1	ND	1.5	T	7.0	4.4	924	≤1	16	<1	74	47	82

T = Trace (<0.03%)
ND = Not Detected (<0.01%)
— = Not Analyzed

REFERENCES

1. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 2, February 1968–April 1968," May 1968 (GEAP-5624).
2. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 1, November 1967–January 1968," March 1968 (GEAP-5598).
3. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 4, August 1968–October 1968," November 1968 (GEAP-5702).
4. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 10, February–April 1970," May 1970 (GEAP-10204).
5. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 3, May 1968–July 1968," August 1968 (GEAP-5667).
6. "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 5, November 1968–January 1969," February 1969 (GEAP-5746).

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