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POST-IRRADIATION EXAMINATION
OF CVTR FUEL ASSEMBLIES

FINAL REPORT

M. G. Balfour and J. F. Mellor

August 1969

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ABSTRACT

A program of post-irradiation examinations has been conducted upon failed CVTR standard core and experimental fuel rods to determine the nature and the causes of the defects. The examinations have shown a single rod failure in each of two standard assemblies and in three of four special high power test assemblies. The failures generally are characterized by multiple clad defects, with localized internal hydriding, but no correlation can be found in terms of design or irradiation performance to account for them. The evidence suggests that the failures were caused by local hydriding and consequent embrittlement of the Zircaloy clad combined with clad stresses. The source of internal hydriding is related to the fabrication history peculiar to each rod.

SECTION 1

INTRODUCTION

1.1 PROGRAM DESCRIPTION

A series of fuel element defects occurred in both standard and experimental assemblies operating in the Carolinas-Virginia Tube Reactor (CVTR), Parr, S. C. A post-irradiation program to determine the cause of the defects was subsequently conducted. This program was sponsored jointly by the USAEC and Westinghouse Electric Corporation under Contract AT(30-1)-3850, with the following objectives:

1. Locating and identifying the individual cladding defects;
2. Determining the nature of the cladding defects;
3. Determining the mechanism(s) producing the defects; and
4. Identifying, based on the results of the above investigations, the necessary revisions to the fuel design, fuel and cladding specifications, fabrication procedures, or fuel operating conditions, which, if utilized, would have prevented these fuel cladding defects.

Four special test assemblies and two standard core assemblies were shipped to the Westinghouse Post-Irradiation Facility for examination. The Post-Irradiation Examination of CVTR Fuel Assemblies began in May 1967 and consisted of nondestructive and destructive examinations of selected failed and nonfailed fuel rods.

This contract was assigned to the Nuclear Fuel Division (NFD) of the Westinghouse Nuclear Energy Systems, with overall project management conducted by J. V. Haley, Project Manager, Fuel Projects. Technical direction of the work was the responsibility of R. S. Miller, Manager of Irradiation Technology, and K. R. Jordan, Manager of Fuel Materials Technology. The principal investigators were M. G. Balfour (NFD) and J. F. Mellor (NFD).

The nondestructive and destructive examinations were administered by D. T. Galm, Manager of Technical Service Laboratories (TSL) and E. S. Schwartz, Manager of the Post-Irradiation Facility (WPIF). Fuel assembly shipping and licensing was directed by H. E. Walchli, Manager of Fuel Service (NFD).

This is the last of a series of technical progress reports on the Post-Irradiation Examination of CVTR Fuel Assemblies. The previous reports in this series are:

WCAP-3850-1	"Quarterly Progress Report for the Period Ending September 30, 1967"
WCAP-3850-2	"Quarterly Progress Report for the Period Ending December 31, 1967"
WCAP-3850-3	"Semi-Annual Progress Report for the Period Ending June 30, 1968"

1.2 SPECIAL TEST ASSEMBLIES

In May 1966, four special test assemblies began irradiation in CVTR.* The test assemblies contained UO_2 fuel rods designed to provide experience in the following areas:

1. High specific power;
2. High burnup;
3. Thin-walled clad; and
4. Variations in fuel cladding material processing.

The experimental design for this program is described in detail in Reference 1. Each test assembly, comprising a total of 19 rods, contained six UO_2 fuel rods, located in the inner ring of the assembly configuration (Figure 1). Fuel rods of two enrichments were used, designed for peak power levels of 20.9 and 25.1 kw/ft, respectively. The remaining rods were dummies, containing Al_2O_3 pellets. Each UO_2 rod contained pellets sintered to

* The CVTR Post-Construction Research and Development Program was conducted jointly by the USAEC, Westinghouse Electric Corporation, and Carolinas-Virginia Nuclear Power Associates under Contract AT(30-1)-2289.

X - VIBRA-TOOL MARK ON UPPER END PLUG AS SHOWN TO FIX ORIENTATION
 ⊕ - TO NOTCHES IN UPPER GUIDE RING

Figure 1. Location of Special Test Assembly Fuel Rods

93 or 96 percent of theoretical density. Table 1 gives the individual design and fuel loading data for the special rods. The upper grids on each assembly were modified so that fuel rods could be removed individually. Following the standard CVTR design, each rod was spirally wrapped with a Zircaloy wire spacer, 0.112 inch diameter. Each test assembly was located in the inlet leg of a pressure tube, with a fresh standard (2.0 w/o U^{235}) CVTR core assembly occupying the outlet leg.

1.3 STANDARD CORE ASSEMBLIES

Each standard assembly contained 19 fuel rods with 93 percent dense, 2.0 w/o enriched UO_2 . Complete design data may be found in References 2 and 3. A comparison between the standard and special test assembly fuel rod designs is presented in Table 2. The fuel rod number identification and location for standard assemblies 502-1-18 and 502-2-31 is shown in Figure 2.

TABLE 1

ENRICHED FUEL ROD DESIGN DATA - CVTR HIGH-POWER FUEL ASSEMBLIES^a

Fuel Assembly Group	Rod Identity Number	Cladding Material	Temper	Peak kw/ft	UO ₂ Fuel Enrichment	UO ₂ Density (% T.D.)	UO ₂ Pellet Diameter (Inches)	Length of Fuel (Inches)	Weight of Fuel (Grams)	Clad ID (Inches)	Minimum Wall (Inches)	Cold Gap Diameter Clearance (Inches)
1	33.832	Zr-4	H-14 ^c	20.90	3.8%	93%	.4329	81.0625	1967.9	.4395	.0222	.0066
	W13.831	Zr-4	H-33 ^b	20.90	3.8%	93%	.4329	80.250	1963.7	.4395	.0222	.0066
	44.731	Zr-4	H-14 ^c	25.08	4.7%	93%	.4317	77.8125	1894.3	.4395	.0222	.0078
	44.732	Zr-4	H-14 ^c	25.08	4.7%	93%	.4317	77.8125	1892.4	.4395	.0222	.0078
	24.732	Zr-4	H-33	25.08	4.7%	93%	.4317	77.8125	1893.7	.4395	.0222	.0078
	33.833	Zr-4	H-14 ^c	20.90	3.8%	93%	.4329	81.125	1932.3	.4395	.0222	.0066
2	13.836	Zr-4	H-33	20.90	3.8%	93%	.4329	81.125	1974.2	.4395	.0222	.0066
	13.834	Zr-4	H-33	20.90	3.8%	93%	.4329	80.969	1968.1	.4395	.0222	.0066
	13.833	Zr-4	H-33	20.90	3.8%	93%	.4329	81.0625	1971.9	.4395	.0222	.0066
	13.831	Zr-4	H-33	20.90	3.8%	93%	.4329	81.156	1968.5	.4395	.0222	.0066
	13.835	Zr-4	H-33	20.90	3.8%	93%	.4329	81.0625	1933.9	.4395	.0222	.0066
	13.832	Zr-4	H-33	20.90	3.8%	93%	.4329	81.125	1965.5	.4395	.0222	.0066
3	53.861	Zr-4	H-33	20.90	3.8%	96%	.4329	81.0625	2046.1	.4395	.0222	.0066
	13.837	Zr-4	H-33	20.90	3.8%	93%	.4329	81.0625	1951.1	.4395	.0222	.0066
	64.761	Zr-4	H-33	25.08	4.7%	96%	.4317	77.8125	1912.8	.4395	.0222	.0078
	64.762	Zr-4	H-33	25.08	4.7%	96%	.4317	77.938	1920.9	.4395	.0222	.0078
	24.734	Zr-4	H-33	25.08	4.7%	93%	.4317	77.875	1897.0	.4395	.0222	.0078
	53.862	Zr-4	H-33	20.90	3.8%	96%	.4329	81.0625	2022.1	.4395	.0222	.0066
4	83.832	Zr-4	H-33	20.90	3.8%	93%	.4475	81.156	2135.2	.4535	.0160	.0060
	73.832	SS	b	20.90	3.8%	93%	.4560	81.0625	2167.5	.4625	.0120	.0065
	13.838	Zr-4	H-33	20.90	3.8%	93%	.4329	81.0625	1991.4	.4395	.0222	.0066
	13.839	Zr-4	H-33	20.90	3.8%	93%	.4329	81.0625	1986.0	.4395	.0222	.0066
	73.833	SS	b	20.90	3.8%	93%	.4560	81.250	2171.2	.4625	.0120	.0065
	83.831	Zr-4	H-33	20.90	3.8%	93%	.4475	81.125	2172.4	.4535	.0160	.0060

- a. Each test assembly contains six UO₂ rods with peak burnup goals of 40,000 MWD/MTU (25.08 kw/ft) and 33,333 MWD/MTU (20.90 kw/ft). The remaining thirteen rods in each assembly are dummy rods containing Al₂O₃ pellets, clad with Zr-4.
- b. Welded tubing. The other fuel rods have seamless tubing.
- c. Higher cold-worked clad, temper H-14.

TABLE 2

COMPARATIVE DESIGN DATA OF CVTR FUEL RODS

	<u>Assemblies G1,2,3</u>		<u>Assembly G-4</u>			<u>Standard CVTR Assembly (65 MW)</u>
No. rods per assembly (UO_2 , Al_2O_3)	19		19			19
No. UO_2 rods per assembly	6		6			19
Clad Material	Zr-4		Zr-4	Zr-4	#304S/S	Zr-4
Clad Type	Freestanding		Thin-wall	Freestanding	Thin-wall	Freestanding
Clad outside diameter, in. (± 0.0015)	0.4875		0.4875	0.4875	0.4875	0.4875
Clad thickness (nominal), in.	0.024		0.017	0.024	0.0125	0.02575
Sp. power density, kw/ft (peak)	20.90	25.08	20.90	20.90	20.90	17.00
Nominal cold pellet-clad gap (dia.) in.	0.0066	0.0078	0.0060	0.0066	0.0065	0.0060
Pellet diameter, in. (± 0.0005)	0.4329	0.4317	0.4475	0.4329	0.4560	0.4300
Cold fuel column length, in. (± 0.35)	81.25	78.00	81.25	81.25	81.25	95.40
UO_2 enrichment, w/o U-235	3.80	4.70	3.80	3.80	3.80	1.5, 2.0
UO_2 density, % TD ($\pm 1.5\%$)	93, 96	93, 96	93	93	93	93

LOCATION OF FUEL RODS, STANDARD ASSEMBLY 502-2-31

LOCATION OF FUEL RODS, STANDARD ASSEMBLY 502-1-18

Figure 2. Location of Standard Assembly Fuel Rods

SECTION 2

IRRADIATION RESULTS

2.1 IRRADIATION HISTORY, FAILED ASSEMBLIES

Detailed irradiation histories of the failed fuel assemblies are presented in References 1 and 4, based upon CVTR core measurements and calculations made prior to the post-irradiation examinations. The following paragraphs summarize the irradiation history of the relevant assemblies.

Core I of the CVTR first went critical on March 31, 1963, went to power in December 1963, and operated at power levels up to the maximum of 44.3 MWt until September 1965. At this time, the maximum power level was increased to 65 MWt. Core I operation was completed in March 1966, at which time refueling operations began for Core II. This included the insertion of four special test assemblies in peripheral core positions (Figure 3).

During the first four months irradiation of CVTR Core II, beginning in mid-May 1966, all test assemblies operated satisfactorily with the reactor operating at 80 to 92 percent of full core power (MWt). In the hottest of the four assemblies, a maximum specific power level of approximately 19.8 kw/ft and a peak pellet burnup of approximately 7300 MWD/MTU were achieved. The reactor was shut down for routine refueling during September 1966.

During subsequent irradiation, three of the four test assemblies developed fuel defects, detected by coolant activity monitoring. A defect in assembly G-3 was detected on October 5, 1966, during approach to full power following CVTR refueling operations. During reactor startup following removal of assembly G-3, a rod banking program was instituted to modify the axial flux distribution in the test assemblies so that design peak power levels could be achieved. A defect in test assembly G-1 occurred during this operation and was detected on November 21, 1966. This assembly was also removed. Higher coolant

Figure 3. CVTR Core and Location of Defected Assemblies

activity in test assembly G-4 was detected on January 24, 1967. Assembly G-4 contained both free-standing and thin-wall cladding. This assembly and the remaining test assembly, G-2, were both removed, since further irradiation was not possible because of the decision to terminate the CVTR operation by June 30, 1967.

High coolant activity, indicating fuel element defects, was observed in the pressure tubes containing standard (2.0 w/o U^{235}) CVTR assemblies 502-1-18 and 502-2-31. Although the exact dates are obscured by coolant activity from other sources, it is believed the defects occurred during operation on November 21, 1966. Each assembly (together with a sister assembly) originally occupied a peripheral core position (Figure 3) during CVTR Core I operation (6-10 kw/ft peak, approximately 9400 MWD/MTU average burnup) and was moved to a center position for Core II. These assemblies operated successfully during the first fuel cycle at about 10-12 kw/ft peak, prior to the occurrence of defects. Calculated burnups and power levels at the assumed time of defecting are listed in Reference 1. The assemblies were subsequently operated at slightly reduced power until plant shutdown, accumulating an additional peak burnup of approximately 1200-1500 MWD/MTU. The four special test assemblies and the two defected standard assemblies were then shipped to the Westinghouse Post-Irradiation Facility for examination.

No other fuel assemblies failed during the operation of the CVTR. Of the standard 2 w/o and 1.5 w/o fuel rods, two out of a total 1368 rods in the initial core failed. In addition, 418 other standard rods were irradiated without failure as replacements for depleted fuel or failed special test assemblies. Three of the twenty-four special test rods failed, as identified in the following section.

2.2 IRRADIATION PERFORMANCE, FAILED FUEL RODS

Table 3 summarizes the irradiation performance, clad and fuel design, and failure characteristics observed in the post-irradiation examinations. The burnups and power ratings listed here utilize gamma scans and radiochemical analysis data, and supersede previously reported information^[1, 4]. Design irradiation limits were not exceeded for any of the rods: standard fuel

TABLE 3

SUMMARY OF FAILED CVTR FUEL ROD CHARACTERISTICS

Assembly Number	Rod Number	Peak kw/ft ^a	Peak Mwd/MTU ^b	Date	Reactor Operation	Vendor	% Cold Work	Nom. Clad Thickness (mils)	Nom. Fuel Density (% T.D.)	Nom. Dia. Gap (Mils)	Fuel Column Length (inches)	Failure Characteristics ^d
Standard 502-2-31	1549	11.3	14,900	11/21/66 ^c (assumed)	Rod banking + power increase	A	15	25.75	93	6.0	95.4	Blisters & clad spalling, lower 10 in. rod. Upper end plug hydrided. Blister and pin-hole defect near lower end plug weld. Internal clad cracks at peak flux location (26-1/2 in.) during failure.
Standard 502-1-18	1475	12.5	15,700	11/21/66 ^c (assumed)	Rod banking + power increase	A	15	25.75	93	6.0	95.4	Blisters in upper 40 in. of rod. Clad failure at upper end plug weld. Internal clad cracks at peak heat flux location (26-1/2 in.).
Special G-1	24.732	21.2	9,400	11/21/66	Rod banking + power increase. Operated at 21.8 kw/ft on 10/5/66.	B	>40% + 950°F 4 hr. stress relief	24	93	7.8	77.8	Clad fracture, blisters ridges and cracks in lower 43 in. of rod. Severe internal corrosion and hydriding.
Special G-3	53.861	17.2	7,200	10/5/66	Approach to power after plant shutdown	B	>40% + 950°F 4 hr. stress relief	24	96	6.6	81.1	Clad failure at upper end plug; internal clad pitting.
Special G-4	83.831	16.0	8,600	1/24/67	Failed at steady state power. Operated at 17.6 kw/ft during rod banking experiment on 11/21/66.	A	>75% + 915°F 4 hr. stress relief	17	93	6.0	81.1	Clad blisters and cracking lower 9 - 18 in. of rod. Internal clad cracks, pitting and hydriding. Extensive clad ridging.

Vendor A - Tube reduced with sinking operation as final reduction (standard rods); G-4 rods were tube reduced to final size.

Vendor B - Tube drawing to final size.

a. Value for time of failure.

b. Based upon radiochemical measurements (Nd-148 analysis).

c. The standard assemblies continued operation (~1200-1500 MWD/MTU additional peak burnup) after failure until 1/24/67 shutdown.

d. The term hydriding is used generically to indicate reaction of the cladding with any hydrogen isotope.

rods were designed for a peak power level and peak burnup of 17.0 kw/ft and >30,000 MWD/MTU, respectively. Special test rods were designed to achieve 20.9 and 25.1 kw/ft and ~33,000 and ~40,000 MWD/MTU, depending on the enrichment. At the relatively low power levels achieved, DNB was not considered a likely cause of failure in either standard or special assemblies^[5]. Metallographic studies have confirmed this.

Furthermore, there is no consistent pattern in the failures to indicate a burn-up kw/ft performance limitation. In the case of the standard assemblies, neither 502-2-31 (occupying outlet leg 7K) or 502-1-18 (inlet leg 6I) was the peak burnup assembly in the CVTR core, or even in their respective U-tubes. In fact, the failed rods were not the peak burnup rods within these assemblies. In special assembly G-3, it was a 3.8 w/o enriched rod that failed (17.2 kw/ft peak), rather than one of the three 4.7 w/o enriched rods (20.6 kw/ft peak) in the same assembly.

The only significant similarity in irradiation history of the failed rods is the record of failure upon an increase in linear power level. Special assembly G-1 and the two standard assemblies are all believed to have failed during the November 11, 1966, rod banking experiment; special assembly G-3 failed on approach to power following the core shutdown for refueling in September 1966. However, special assembly G-4 did not fail until January 24, 1967, and at steady-state power. It should be observed that G-1 successfully experienced a peak of 21.8 kw/ft at the time assembly G-3 failed.

More complete descriptions of the failed rods and their performance are given in following sections devoted to nondestructive and destructive examinations, and to burnup and power history. The series of previous program progress reports give further detailed information on the nondestructive and destructive examinations.

SECTION 3

VISUAL AND METALLOGRAPHIC CHARACTERIZATION OF THE FAILED RODS

This section summarizes the main features of the clad and fuel of the failed rods, with special attention to defect areas. In addition, metallographic results from sister (unfailed) rods are presented where comparisons with the failed rods is particularly significant. The program was carried out on the basis of visually and destructively examining two sister rods from each standard assembly and one sister rod from each failed special assembly. Three additional special rods were examined to a limited extent late in the program. Complete data from visual and metallographic examination can be found in previous progress reports. Sectioning diagrams for all destructive examinations are given in Appendix D. In addition to the axial location and type of each section, a brief description is given in cases of particular interest.

3.1 SPECIAL ASSEMBLY G-1

Fuel rods destructively examined in this assembly were 24.732 (failed), 44.731, and 33.833. The design of these rods is summarized in Table 1, Section 1.2. A series of multiple defects and anomalies characterize failed rod 24.732. These are:

1. Local hydriding* attack at the crevice between upper end plug and clad;
2. A series of four clad circumferential ridges between 40 and 43 1/4 inches from the bottom. From observations of the inner surface

* For ease of exposition, the generic term "hydriding" is used throughout this report to describe the reaction of the Zircaloy clad with either hydrogen (mass one) or deuterium (mass two) isotopes. When the distinction between the two is important, it is noted explicitly.

after clad longitudinal slitting, the ridges appear to have formed at pellet interfaces. The maximum height of the ridges is 2.7 mils, as determined from profilometry.

3. A large (~7 mils high, ~360 mils wide) blister at 31 inches, which was cracked, is shown in Figure 4. The (white) discolored area on the clad shows the extent of the blister. Emission of water (and dissolved fission products) under the heat of illuminating lamps used during visual examination proved that the blister was defected. A secondary fracture of the rod occurred at this point during a handling accident.
4. A circumferential ridge (~5 mils high) at 17 inches; a secondary clad fracture also occurred here.
5. The original clad fracture at 12.8 inches is shown in Figures 5 and 6. The fracture was circumferential and occurred at a fuel pellet interface; a longitudinal metallographic section through the defect (with fuel removed) illustrates the clad hydriding, which is limited to the immediate fracture region. Also shown is a region of internal attack and incipient hydriding, near the fracture surface. The specimen was electroplated in an attempt to preserve the oxide surfaces layer.
6. A large clad blister (~11 mils high, ~200 mils wide) at ~12.1 inches from the bottom. This blister was also located opposite a fuel pellet interface.

Transverse metallographic sections were taken of rods 24.732 and 44.731 at 26 1/2 inches, the approximate axial location of the peak heat flux region during the rod banking experiment. Figure 7 shows the clad and fuel appearance for rod 24.732 and Figure 8 gives the fuel structure and autoradiograph of rod 44.731 at the same axial location. Both fuel rods show a similar fuel structure, with well-defined columnar grains and a center region clearly defined by the maximum extent of columnar grain growth. The autoradiograph in Figure 8 shows definite evidence of fission product redistribution analogous to the observed structural changes. Whereas the clad of rod 44.731 showed

Figure 4. Fuel Rod 24.732: Clad Blister, 31 Inches

X 4

Figure 5. Fuel Rod 24.732: Fracture Surface at 12.8 Inches, Lower Section

X 4

Figure 6. Fuel Rod 24.732: Longitudinal Section, 12.8-13.3 Inches; Original Fracture Surface and Localized Internal Hydriding

X 4

X 100

Figure 7. Fuel Rod 24.732: Fuel and Clad Structure, 26-1/2 Inches

X 4

X 4

Figure 8. Fuel Rod 44.731: Metallographic Section and Autoradiograph, 26-1/2 Inches

only minor hydride precipitation, the inner surface of rod 24.732 is characterized by extreme local oxidation and hydriding. The "sunburst" region results from coolant ingress following clad failure approximately 14 inches below this location.

An additional sister rod 33.833 from special assembly G-1 was examined metallographically, 26-1/2 inches from the bottom. The upper end plug vent hole in this rod had not been closed immediately after the end plug weld; evidence of internal contamination in a fuel rod (83.832, assembly G-4) with a similar fabrication history led to a partial examination of rod 33.833. Because this work was performed very late in the program, other possible hydrided areas (such as the upper end plug-clad crevice) could not be investigated. This is also true of rod 24.734 of assembly G-3. No evidence of internal clad oxidation or unusual hydriding was discovered in 33.833. Figure 9 shows the typical clad microstructure.

3.2 SPECIAL ASSEMBLY G-3

Fuel rods destructively examined from this assembly were rods 53.861 (failed), 53.862, and 24.734. The failure of rod 53.861 occurred at the top end plug region, immediately beneath the weld zone. The top end plug was observed to be completely disengaged from the fuel tube when examined in the hot cells. The clad was badly corroded, as shown in Figure 10; the reinsertion of the upper end plug into the lower section illustrates the extent of the missing clad metal. A longitudinal metallographic section of the clad fracture surface is shown in Figure 11. There is extensive local hydriding and cracking at this point; part of the clad obviously has spalled away, as observed in the visual examination. The end plug itself and the end plug-clad crevice below the weld show similar hydriding and corrosion. It should be noted that the failure was not located in pressure tube 3B (containing assembly G-3) until about three weeks after the initial indications of fission product activity in the CVTR coolant. During this time, the reactor was cycled at low power to produce activity bursts while individual pressure tubes were monitored in search for the core location of the defect.

X 100

Figure 9. Fuel Rod 33.833: Typical Clad Appearance, 26-1/2 Inches

X 3

X 3

Figure 10. Fuel Rod 53.861: Clad Failure at Upper End Plug

I.D.

O.D.

X 100

Figure 11. Fuel Rod 53.861: Clad Defect at Upper End Plug

No other defects were observed on rod 53.861, although profilometer scanning revealed circumferential ridging, from 1.0 - 1.5 mils height, spaced approximately 0.650 inches (the length of a fuel pellet) apart. The ridging generally was concentrated within 25 to 58 inches from the bottom of the rod.

The fuel clad of 53.861 was characterized by moderate hydride/deuteride precipitation and localized corrosive attack on the clad ID as shown in Figure 12. The clad of sister rod 53.862 (Figure 13) showed very light, circumferential hydriding with a slightly rough bore, but lacking the severe corrosive pitting observed in the failed rod. Both 53.861 and 53.862 had late vent hole closures of their upper end plugs. The fuel structures of both rods were similar, with a center region of equiaxed grain growth, cracking throughout the pellet cross-section, and minimal fission product redistribution.

Because it shared the fabrication anomaly of a late vent hole closure, fuel rod 24.734 was also examined at the ~51-inch axial location. Figure 14 shows the fuel structure and a section of clad that reveals an anomalous region of bore pitting and oxidation. Although this is indicative of possible rod contamination, no significant increase in clad hydriding (as compared to rod 53.861) was observed at this location. The columnar grains and center void in the fuel reflect the higher linear heat rating (~20.6 peak) experienced by 4.7 w/o enriched rods in this assembly.

3.3 SPECIAL ASSEMBLY G-4

Fuel rod 83.831 had a series of six clad blisters between 9 and 18 inches from the bottom, of which those at ~14 inches and 16 1/2-inches were defected (Figure 15). The largest (~7.8 mils high, 600 mils wide) of the blisters was at 14 inches, and this likely was the original defect. The crack in the other blister may have been the result either of handling damage or thermal shock during the liquid nitrogen-methanol leak test. No other defects exist on the rod; extensive clad ridging was found as well as on sister rod 83.832*.

* WCAP-3850-3.

X 100

X 250

Figure 12. Fuel Rod 53.861: Internal Corrosion Opposite Fuel Crack,
51-1/2 Inches

90°

X 100

180°

X 100

Figure 13. Fuel Rod 53.862: Typical Clad Hydriding, 51-1/2 Inches

X 4

X 250

Figure 14. Fuel Rod 24.734: Fuel Structure, Local Clad Corrosion at 51 Inches

X 3

X 3

Figure 15. Fuel Rod 83.831: Defected Clad Blisters at 14 and 16-1/2 Inches

In addition to the blisters, other anomalies found on rod 83.831 were micro-cracks and a very rough bore, and heavy hydriding (Figure 16). Internal clad corrosion, as seen in the upper photograph (51-1/2 inches from the bottom), is not unexpected in a failed rod; the external hydride/deuteride band (etched white) at 26-1/2 inches (lower photomicrograph) lacks an adequate explanation, however. Further examination of the clad region, between 26 and 27 inches, indicates the band is continuous (although varying in width) over this interval. Hot extraction H/D analysis indicates predominantly deuterium (~ 1000 ppm maximum), but the measured oxide film ($\sim 0.8 \times 10^{-4}$ inches maximum) does not indicate clad-coolant corrosion which can account for this. Sister rod 83.832 is characterized by a rough bore, ID cracks and microcracks, and local patches of hydriding, despite the fact this rod was not defective. The largest of the cracks (~ 4 mils deep) occurred at 26-1/2 inches, shown in Figure 17 (upper picture). Local hydriding on the clad bore was found at both 26-1/2 and 51-1/2 metallographic specimens; an example (with further clad cracks) is shown in the lower picture of Figure 17. Rods 83.831 and 83.832 had late vent hole closures; hydrogen/deuterium analysis has confirmed the existence of internal (hydrogen) contamination both in the failed and unfailed rods.

3.4 STANDARD ASSEMBLY 502-2-31

3.4.1 Failed Rod 1549

Defects occurred in rod No. 1549 at the lower end plug (pinhole defect), at $\sim 7-5/8$ and $9-3/4$ inches (blister defects), and at the upper end plug (clad cracking). The latter defect occurred during handling, and may be attributed to the severe hydriding discovered in the upper end plug region. Figure 18 shows a cross-section of the blistered lower end plug-clad crevice region, with the pinhole defect in the clad. The external appearance of the large clad blister defects is shown in Figure 19. Each clad penetration was approximately 150 mils in diameter. The defects were developments of localized clad blisters (approximately 300 mils wide) which subsequently cracked and broke away. Each was located on a clean part of the fuel rod opposite a heavily crudded (~ 9 mils thick) surface. A crud distribution map is given in Appendix B.

0°

X 100

180°

X 100

Figure 16. Fuel Rod 83.831: Clad Hydriding at 51-1/2 Inches (Top); at 26-1/2 Inches (Bottom)

X 100

X 250

Figure 17. Fuel Fod 83.832: Internal Clad Cracks and Local Hydriding, 26-1/2 Inches

X 50

Figure 18. Fuel Rod 1549: Defect Region at Lower End Plug

X 3

X 3

Figure 19. Fuel Rod 1549: Large Clad Defects, 7-5/8 and 9-3/4 Inches

X 100

X 4

Figure 20. Fuel Rod 1549: Longitudinal Section of 7-5/8 Inch Clad Defect

X 4

Figure 21. Fuel Rod 1549: Clad Bore, ~50 Inches

0°

X 100

X 4

180°

X 100

Figure 22. Fuel Rod 1398: Metallographic Section, 26-1/2 Inches

Sister rod 1257 was characterized by very light clad hydriding and an extremely rough bore, exceeding failed rod 1549 in severity. This rod was located in the inner ring of fuel rods (cf. Figure 2) and was the lowest-power rod in the assembly. Fission gas results were normal for this rod.

3.5 STANDARD ASSEMBLY 502-1-18

3.5.1 Failed Rod 1475

The defects on rod 1475 consisted of an upper end plug clad failure (Figure 23) and a series of four clad blisters (Figure 24) located between 65 and 83-1/4 inches from the bottom. At the upper end plug defect, the clad beneath the weld zone had spalled away, exposing the badly corroded end plug material beneath. Up to 200 mils of clad was missing. Of the clad blisters, the one at 65 inches was defected as indicated by the surface drop of water that was expelled through it. Another blister, at 76-3/8 inches, displayed spalling at its apex. Unlike failed rod 1549, standard assembly 502-2-31, most of the blisters on rod 1475 were located underneath (rather than opposite) a heavy crud layer. No further defects were noted on this rod, although there was considerable clad ID cracking observed metallographically at 26-1/2 inches from the bottom. No through defect was found at this point. Figure 25 shows a longitudinal metallographic section of the clad just below the failure. The extreme localization of the severe hydriding is apparent here, diminishing rapidly a short distance from the fracture end. The observable bowing out of the fracture end probably was caused by volume expansion from the local hydriding in the end plug beneath the clad.

With the exception of blistered regions, the clad at other locations was only moderately hydrided. A rough bore was also evident.

3.5.2 Sister Fuel Rods 1521 and 1591

Metallography on these rods showed normal fuel and clad appearance except for a large clad ID crack (26-1/2 inches) in rod 1521. This was the only instance of this phenomenon in an unfailed standard rod, of the four examined. The metallography on both rods is presented in WCAP-3850-3.

X 4

X 10

Figure 23. Fuel Rod 1475: Upper End Plug Clad Defect

Figure 24. Fuel Rod 1475: Clad Blisters at 65, 76-3/4, and 83-1/4 Inches

X 100

X 4

X 100

Figure 25. Fuel Rod 1475: Clad Fracture at Upper End Plug

SECTION 4

NONDESTRUCTIVE EXAMINATIONS

4.1 GAMMA SCANNING

The method and results of the gamma scanning of fuel rods which were subsequently destructively examined are presented in Section 6.2.2 and Appendix A. The data are treated in conjunction with burnup and power determinations.

4.2 LEAK TESTING, VISUAL EXAMINATIONS

Prior to visual examination, all fuel rods were routinely (bubble) leak tested by immersion in liquid nitrogen followed by methanol. Subsequent testing for secondary defects was performed on failed rods after the primary defects had been isolated from the remainder of the rod. Methods used for this operation included liquid nitrogen-methanol, dye-penetrant, and helium leak tests.

The full details of visual examinations in failed and sister fuel rods are given in the series of quarterly progress reports. A summary for the failed rods is presented in Section 3.

4.3 DIMENSIONAL MEASUREMENTS, PROFILOMETRY

Table 4 summarizes the length measurements made on the CVTR rods before and after irradiation. In the case of the special rods pre-irradiation length measurements were taken only in the case the rod deviated from specifications of 102.909 ± 0.058 inch. Failed rods 24.732 (assembly G-1), 53.862 (assembly G-3, 1549 (502-2-31), and 1475 (502-1-18) could not be measured because of the nature of the defects.

Systematic pre-irradiation diametral measurements were not taken on the CVTR rods and a meaningful comparison of pre- and post-irradiation data cannot be made. Profilometer scans were made of the rods scheduled for destructive examination, and measured diameters were generally within specifications of

TABLE 4
LENGTH MEASUREMENTS, CVTR RODS

<u>Assembly Number</u>	<u>Rod Number</u>	<u>Pre-irrad. Length, in.^a</u>	<u>Post-irrad. Length, in.^a</u>	<u>Length Change, in.</u>	<u>Percent Change</u>
502-1-18 (Standard)	1521	102.350	102.490	0.140	0.14
	1591	102.360	102.484	0.124	0.12
502-2-31 (Standard)	1257	102.369	102.472	0.103	0.10
	1398	102.359	102.456	0.097	0.09
G-1 (Special)	44.731	103.039 ±0.028	103.072	0.033	0.03
	33.833	103.072 ±0.028	103.056	(-0.016)	(-0.016)
G-3 (Special)	53.862	102.909 ±0.058	103.094	0.185	0.18
	24.734	102.909 ±0.058	103.033	0.124	0.12
G-4 (Special)	83.831	102.909 ±0.058	103.160	0.251	0.24
	83.832	102.909 ±0.058	103.149	0.240	0.23
G-2 (Special)	13.831	103.118 ±0.028	103.156	0.038	0.037

a. Measured lengths are with an accuracy of ±0.002 inches. Deviations (special rods) shown are for the cases where no pre-irradiation lengths were measured or where the measurement was from the lower end plug shoulder, rather than tip. Fabrication tolerances have been assigned here.

0.4860/0.4890 inch for special assembly rods and 0.4855/0.4895 inch for standard assembly rods. Exceptions occurred at defect (blistered) regions and at regions of locally high crud deposition or ovality on the standard rods. Typical profilometer traces illustrating anomalies of high ovality, blistering, and ridging are given in WCAP-3850-2 and WCAP-3850-3.

4.4 CRUD MAPPING, ANALYSIS

Crud deposition on the special assembly fuel rods was comparatively light (≤ 2 mils maximum) with no discernible pattern. However, one prominent characteristic of peripherally-located standard fuel rods was very heavy crud deposition. Figure 26 shows a typical heavy crud deposit on rod No. 1475, about 11-1/2 inches from the top. The uppermost part of the crud layer, part of which has broken away, is 8.6 mils thick (measured with filar eyepiece). Other crud deposits have a rippled appearance, as shown in Figure 27, located (~ 8 inches) near the bottom of rod No. 1549.

Crud deposits up to 15 mils thick were measured on fuel rods from both standard assemblies. On rods removed from peripheral assembly locations the crud, typically, was concentrated on one (half) face of the fuel rod, with little influence from the wire spacer. Observations of rods after removal of the surrounding hexagonal can showed that the crud layer in the peripheral rods faced outward (toward the can). Crud maps of the fuel rod surface are shown in Appendix B for rods 1549 (defected) and 1257 from 502-2-31, and rods 1475 (defected) and 1521 from 502-1-18. In each case, only the comparatively heavy crud deposits (i.e. ≥ 2 mils) are shown; the discontinuities in the patterns may also be a result of crud falling off during handling. Spot measurements taken of the maximum crud thickness are shown on the schematic rod diagrams. The front and reverse views of the rods have been juxtaposed to show the continuous patterns. Also shown are the defects on rods 1475 and 1549. The heavy crud on rod 1257, located in an inner position in the assembly (Figure 2) was concentrated mainly near the end plugs with the remainder generally bright and free of crud.

The measured surface concentrations and calculated density of the corrosion product deposition for selected fuel rods are listed in Table 5. The samples

TABLE 5

SURFACE CONCENTRATION AND DENSITY OF CVTR FUEL ROD CORROSION PRODUCT DEPOSITS

<u>Parameter</u>	<u>Fuel Rod Number</u>			
	<u>1549</u>	<u>1475</u>	<u>1613</u>	<u>1591</u>
Mass of Crud Collected (mg)	17.48	13.52	47.38	4.14
Surface Area Sampled (dm ²)	69.5 x 10 ⁻⁴	52.5 x 10 ⁻⁴	73.8 x 10 ⁻⁴	13.55 x 10 ⁻⁴
Surface ₂ Concentration (mg/dm ²)	2126	2569	6421	3050
Crud Thickness (mils)	9	8.2	7.8	11.5
Crud Density ^a (gm/cc)	0.93	1.23	3.24	1.04
Sample Location (Inches from botton)	~17.5	~91	~92	~76.5

a. Theoretical density for magnetite, Fe₃O₄, is 5.18 gm/cc.

X 3

Figure 26. Fuel Rod 1475: Clad
Crud Deposit, ~91 Inches

X 3

Figure 27. Fuel Rod 1549: Clad
Crud Deposit, ~8 Inches

were collected remotely by scraping the fuel rod surface and the area and thickness of each sample were measured through the stereomicroscope with a calibrated filar eyepiece.

The calculated crud densities agree well with the exception of rod 1613. In this case, it is believed that the measured crud thickness is less accurate than the other samples, which might account for the discrepancy. It is concluded, therefore, that the apparent density of the CVTR crud in the standard fuel rods is ~ 0.9 to 1.2 g/cc, about 20 percent of the density of magnetite.

The chemical composition and radiochemical analyses of the collected crud are presented in Table 6. The results here are consistent for samples taken from different fuel rods. The crud carried in the primary coolant had been previously sampled by filtration and analyzed^[9]. The crud chemical composition in Table 6 confirms the earlier results, which had been judged reasonable for the CVTR stainless steel-Inconel system.

TABLE 6
CHEMICAL AND RADIOCHEMICAL COMPOSITION OF CVTR FUEL ROD
CORROSION PRODUCT DEPOSITS

<u>Element</u>	<u>Fuel Rod Number</u>	
	<u>1613</u>	<u>1475</u>
	<u>Chemical Composition (%)</u>	
Fe	47.5	43.5
Ni	24.8	17.5
Cr	0.24	0.18
Co	0.12	0.04
Mn	0.52	0.51
Ag	0.001	0.003
Al	0.22	0.27
Be	0.09	0.07
Cu	0.03	0.03
Mg	0.05	0.05
Mo	0.03	<0.02
Pb	0.04	0.05
Si	0.17	0.22
Sn	0.04	0.06
Ti	0.008	0.007
Zr	0.11	0.03
Isotope	Radiochemical Composition dpm/mg oxide ^a	
Cr ⁵¹	1.16 x 10 ⁴	<5.7 x 10 ²
Mn ⁵⁴	6.05 x 10 ⁶	4.72 x 10 ⁶
Co ⁵⁸	5.26 x 10 ⁶	3.92 x 10 ⁶
Fe ⁵⁹	1.65 x 10 ⁵	3.38 x 10 ⁵
Co ⁶⁰	3.08 x 10 ⁷	3.91 x 10 ⁷

a. Activity on 11-15-67.

SECTION 5

ANALYSIS OF ARCHIVE MATERIALS AND FABRICATION HISTORY

5.1 STANDARD FUEL RODS

5.1.1 Ultrasonic Visual and Dye-Penetrant Examination

Approximately 80 fuel cladding tubes surplus to the requirement for standard fuel elements have been located and examined. Sixty-three of the tubes were ultrasonically tested by the vendor to test standards in use at the time of procurement. Seven of the tubes revealed significant ultrasonic indications and two tubes were rejectable. Two further tubes showed a high ultrasonic signal indicative of a very poor, rejectable, surface finish on the tube bore.

All tubes showing signals were sectioned at the defect indications to characterize the defects. With the exception of three locations on two tubes, all defects were found to be surface (OD) scores, scratches, or burrs. Only one of these defects had resulted in a rejectable indication. Because it was impossible after six years to relate outer surface defects to tube quality as received, no great significance can be attached to outer surface defects. The bore defects were observed to be two rejectable indications on one tube (No. 4) and a nonrejectable indication on tube No. 23. Both defects on tube No. 4 are 0.002 inch deep scores associated with a small (0.001 inch deep) radial crack (Figure 28). The defect on No. 23 is a score without radial cracking (Figure 29). Examples of the extremely rough bore condition on certain of the tubes are shown in Figure 30.

Samples of archive tubing have been split and subjected to visual and fluorescent penetrant examination of the tube bore. There were significant variations in bore quality with several tubes having extremely rough bore surfaces. A detailed fluorescent penetrant examination showed few penetrant indications on the bore only, and subsequent micro-examinations revealed only minor pits and bore roughness.

X 100

Figure 29. Cross-Section through A Defect Indication in Tube No. 23.
A Score Indication in the Bore with no Associated Radial Crack

X 100

Figure 28. Cross-Section through a Defect Indication in Tube No. 4.
A Score Indication in the Bore is Associated with a Small Radial Crack

X 100

Figure 30. Cross-Section through Bore Roughness on Tube No. 26 Showing Very Irregular Pattern of the Tube Wall.

5.1.2 Corrosion Resistance, Fluoride Content, Hydride Orientation

An initial series of corrosion tests (Test Series A) was carried out on "as-received" tubing samples in 750°F, 1500 psi steam and 680°F, 2700 psi water. Comparative controls were run on pickled Zircaloy-4 plate and on samples "as-received" which were pickled on the tube OD only to simulate actual production procedure. The weight gain results, Table 7, indicated that "as-received" tubing was corroding slightly faster than expected and that the visual appearance of the corrosion film, particularly on the tube bore, was gray/white in nature. It was discovered after completion of the test series that the corrosion samples had been wiped prior to weighing, possibly removing nonadherent corrosion product, and, therefore, an additional test series (Series B) was initiated.

Series B tests in 750°F, 1500 psi steam showed much higher weight gains than Series A tests on "as-received" and OD pickled samples. By inference from the weight gains and by visual appearance of the tube bore, corrosion was much greater than anticipated. Comparable samples which were completely pickled (ID and OD) showed normal behavior comparable to the control samples. Hot extraction analysis of archive (nonautoclaved) standard tubing revealed a low intrinsic hydrogen content of 1 - 5 ppm.

On the basis of these results, basic material quality appears satisfactory and, since the outside of the tubes was pickled prior to service, the possibility of a failure being in any way attributable to external corrosion and hydriding is remote. The poor corrosion resistance of the tube bore could, however, be a contributory factor to internal hydriding, especially since the quality of the oxide film was quite variable. It would also be anticipated that the poor bore corrosion resistance, which may be attributed to roughness and fluoride contamination, would have resulted in rapid deterioration of the clad in the event of a primary failure due to other reasons.

Eight samples of archive tubing were analyzed for bore fluoride level by the Thoron and Amadac methods. Samples were found to have a relatively high fluoride level of up to $1.1 \mu\text{g}/\text{cm}^2$. This level of fluoride would account for the observable white areas on the bore of the tubes after corrosion testing (Table 8).

TABLE 7
CORROSION OF CVTR STANDARD CLADDING
750°F, 1500 PSI STEAM TESTS

Test Series A <u>Sample</u>	Weight Gain (mg/dm ²)	
	<u>3-Day</u>	<u>14-Day</u>
As-received (degreased only)	20.2	33.2
As-received (O.D. pickled)	21.1	35.2
Control coupons - Zr-4 sheet	16.6	31.1
Test Series B <u>Sample</u>	Weight Gain (mg/dm ²)	
	<u>4-Day^a</u>	<u>14-Day</u>
As-received (degreased only)	30.25	58.7
As-received (O.D. pickled)	32.47	59.6
As-received (O.D. and I.D. pickled)	19.3	29.9
Control coupons - Zr-4 sheet	20.0	27.4

a. Overrun of intended 3-day test

680°F, 2700 PSI WATER TESTS

Test Series A <u>Sample</u>	Weight Gain (mg/dm ²)	
	<u>3-Day</u>	<u>14-Day</u>
As-received (degreased only)	12.6	18.6
As-received (O.D. pickled)	13.1	19.4
Control coupons - Zr-4 sheet	9.9	16.8

TABLE 8
 FLUORIDE ANALYSIS OF ARCHIVE CVTR CLADDING ($\mu\text{g}/\text{cm}^2$)

	<u>Sample</u>	<u>Thoron Method</u>	<u>Amadac Method</u>
	1	0.31	0.46
Standard	2	1.00	1.10
Zr-4	3	0.09	0.88
Cladding	4	0.09	0.83
Special Assembly	53.863	0.39	0.41
Standard Wall	53.863	0.42	0.45
Zr-4 Cladding	24.733	0.03/0.54	0.75
Special Assembly			
Thin Wall Zr-4			
Cladding		1.4	1.3

Samples of archive tubing were assessed for hydride orientation by deliberate addition of hydrogen. It was found that the hydride orientation was random/radial in nature (Figure 31). This finding confirmed the observed hydride orientation in a number of irradiated rods and was anticipated from the tube manufacturing method (i.e., sinking).

5.1.3 Tensile and Burst Properties

Representative tubing samples have been tensile tested in the longitudinal direction at room temperature and 725°F (Table 9). The results are consistent with the finished tubing condition (i.e., 15 percent cold worked) and do not reveal abnormally low values of either strength or ductility.

Samples of standard assembly archive tubing have been examined in closed-end burst tests at 75°F, 600°F, and 725°F, using a pressurization rate of 1000-1800 psi/minute. Though samples were chosen at random from the archive tubing, there were significant variations in hoop ductility of the tubing at room temperature dependent upon the bore condition, as seen in Table 10. The values of yield strength and ultimate strength did not appear to be related to bore condition and were of an acceptable level. In the room temperature tests, the value of hoop ductility varied from 4.7 to 6.6 percent for rough bore tubing, from 12.7 to 13.6 percent for bores of intermediate roughness, and from 22.9 to 29.7 percent for smooth-bore tubing. Photographs of representative fractures are shown in Figure 33. Tests at 725°F showed variations in hoop ductility of from 13.0 to 32.7 percent, but these variations could not be correlated with bore roughness. Additional burst tests at 600°F on "rough" and "smooth-bore" tubing did not reveal low ductility.

Samples of surplus standard assembly tubing containing welded lower end plugs were tested in a "closed-end" burst test at room temperature. The welds were of variable quality, some having rejectable porosity indications and a number of rewelds. With one exception, the samples failed in a ductile manner at the heat affected zone of the end plug weld. There is no evidence that either rewelding or porosity indications resulted in a low-strength, brittle weld (see Table 11).

TABLE 9
TENSILE PROPERTIES OF CVTR STANDARD CLADDING

<u>Identity</u>	<u>Test Temp. (°F)</u>	<u>0.2% Y.S. (psi)</u>	<u>UTS (psi)</u>	<u>Uniform Elong. (%)</u>	<u>Total Elong. (%)</u>
Tube A	RT	79,500	100,025	5.6	17.8
Tube A	RT	79,400	98,900	5.9	17.5
Tube B	RT	73,550	94,475	6.6	13.5
Tube B	RT	74,250	94,675	6.6	14.6
Tube A	725°F	43,625	51,600	4.9	18.8
Tube A	725°F	45,525	52,900	5.1	18.4
Tube B	725°F	43,750	50,350	4.1	10.0 ^a
Tube B	725°F	43,200	51,550	4.8	14.6

a. Fracture occurred at extensometer mounting.

TABLE 10
CLOSED END BURST TESTS ON ARCHIVE STANDARD TUBING

<u>Sample No.</u>	<u>Bore Condition</u>	<u>Test Temp. (°F)</u>	<u>0.2% Yield Strength (psi)</u>	<u>Ultimate Strength (psi)</u>	<u>Elongation (%)</u>
3	A	75	97,500	114,000	4.7
4	A	75	102,200	113,200	6.2
5	A	75	101,700	112,200	6.6
6	C	75	101,000	118,500	29.7
7	C	75	100,800	115,900	22.9
8	B	75	108,400	116,700	12.7
9	B	75	110,100	117,300	13.6
62	A	600	64,900	70,100	34.8
45	C	600	60,300	67,900	31.2
1	A-B	725	-----	62,100	16.2
2	A-B	725	-----	60,300	14.4
10	A	725	59,300	61,200	13.0
11	A	725	57,600	62,400	32.7

A - Bore of tube rough and pitted

B - Bore of tube better than A, but not smooth

C - Smooth bore.

TABLE 11

CLOSED-END BURST TESTS ON STANDARD ASSEMBLY END PLUG WELDS

<u>Rod No.</u>	<u>1st Weld</u>	<u>Rewelds</u>	<u>Disposition March 1962</u>	<u>Disposition March 1968</u>	<u>Ultimate Strength (psi)</u>	<u>Elongation (%)</u>
L1	R-porosity	2	S	R-porosity	108,000 ^a	18.8
L21	S	-	S	S		
L22	R-void	1	S	R-porosity		
L23	S	-	S	S	112,000	15.6
L25	R-void	1	S	R-porosity		
L26	R-void	2	R-porosity	R-porosity		
60	S	-	S	R-porosity	104,000	14.3
102	S	-	S	S		
148	R-porosity	-	R-porosity	R-porosity		
177	R-porosity	2	S	S	100,000	17.6
445	R-shrinkage	3	S	S		
791	R-porosity	1	R-shrinkage	S ^b		
908	R-porosity	1	S	R-porosity	104,000	13.2

All the above welds are bottom end plug welds.

a. Failed in tube, all other samples failed in the heat affected zone of the weld.

b. Evaluated for porosity and undercutting only.

Key R = Reject

S = Satisfactory

X 100

Figure 31. Hydride Orientation in Archive Standard Assembly Zircaloy-4 Clad

X 100

Figure 32. Hydride Orientation in Archive Special Assembly Standard Wall Zircaloy-4 Clad

Figure 33. Representative Burst Test Samples: Standard Clad

5.1.4 Processing Records

The following information has been obtained as a result of a study of available records on material purchase and fuel assembly manufacture and by discussion with material vendors.

The tubing for the standard assembly fuel clad was Zircaloy-4 made by a tube reduction process with a final "sink" pass resulting in 15 percent final cold work.

Approximately 5 percent of the tubes were rejected during ultrasonic testing prior to delivery by the vendor -- the type and incidence (per tube) of defects were not recorded.

Receipt visual inspection by Westinghouse identified certain rejectable surface features, principally areas of oxide film, roughness, and raised areas. Tubes showing defects were returned to the vendor and hand polished to remove the defects. It is apparent from the corrosion records that visual inspection was not always adequate to identify and remove all defects.

All fuel rods were corrosion-tested in 750°F steam for three days prior to service. On the basis of inspection reports, a high incidence of surface defects was reported either after pickling or after corrosion-testing. Complete reports are not available for all rods, but it is estimated that approximately half of all the rods were the subject of inspection comment. Without detailed knowledge of the inspection standard applied, it is not possible to assess the importance of the defects, but approximately 2 percent were rejected for white corrosion product, pits, rough areas, and dents. The most common defect was "blistering" present on approximately 40 percent of all rods. These blisters were the result of pitting by the pickle liquor around areas of oxide film not attacked during pickling.

The available data on corrosion-tested rods in assemblies 502-1-18 and 502-2-31 have been tabulated previously.* Other common defects noted were excess metal, discoloration, acid-staining and white corrosion product. The

* WCAP-3850-3, page 3-94.

recommended repair procedure for rods exhibiting defects was to remove the defects with Brightboy eraser and rerun a corrosion test for one day in 750°F steam to check the efficiency of the repair. There are no specific data on rods repaired in this way.

The fuel pellet loading data for the relevant standard fuel rods are shown in Table 12. The fuel moisture specification (75 ppm) was waived for the pellet vendor on condition that the pellets were to be dried immediately prior to loading. Analysis performed upon receipt of the standard fuel pellets (in early January, 1962) indicated that the moisture content was less than 10 ppm (Table 13). UO₂ pellet drying and loading of the standard rods occurred in mid-February and early March, 1962. The process procedures for pellet drying* included sample moisture checks from each drying tray, but no quantitative data were recorded. The loading and welding records indicate that the normal time interval between pellet drying and loading was less than one day after completion of drying, and the rods were loaded into the welding chamber on the date of pellet loading. It is of possible significance that the routine drying operations on three fuel pellet lots were interrupted because of oven malfunctions. Storage conditions during the interim cannot be ascertained, although it has been recorded that one of the pellet lots was moved to another oven. The pellet lots were subsequently redried one to two weeks later, but not all the pellets from each lot were specifically accounted for. It is, therefore, possible that some pellets inadvertently were not properly dried. These pellets could have been loaded into fuel rods 1475 and 1549 (both failed), 1398 (unfailed, anomalous clad ID oxidation), and 1521 (unfailed, normal).

Two archive standard fuel pellets were analyzed for moisture and volatile impurities (1000°C anneal, mass spectrometry). The results are given in Table 14, along with data on special assembly pellets. Because of the long storage time in an unregulated atmosphere, the moisture data in Table 14 may not be representative of actual loading conditions. Furthermore, the archive

* WCAP-3850-3, page 3-90.

TABLE 12
LOADING DATA ON STANDARD FUEL RODS

Assembly 502-1-18

<u>Rod</u>	<u>Enrichment (%)</u>	<u>UO₂ Batch</u>	<u>Fuel Weight (grams)</u>	<u>Fuel Length (in.)</u>	<u>Spacer Length^a (in.)</u>
1475	2	K05LR03	2289	95.20	0.800 nom. (2 Item 2)
1521	2	K05LR05	2291	95.58	0.600 nom. (2 Item 1)
1591	2	K05LR06	2291	95.16	0.900 nom. (3 Item 1)

Assembly 502-2-31

<u>Rod</u>	<u>Enrichment (%)</u>	<u>UO₂ Batch</u>	<u>Fuel Weight (grams)</u>	<u>Fuel Length (in.)</u>	<u>Spacer Length^a (in.)</u>
1257	2	K04LR05	2286	95.12	0.900 nom. (3 Item 1)
1398	2	K05LR01	2298	95.44	0.600 nom. (2 Item 1)
1549	2	K05LR05	2286	95.26	0.800 nom. (2 Item 2)

a. Spacer Length Item 1 0.290 in. to 0.310 in.
Item 2 0.390 in. to 0.410 in.

TABLE 13

CHEMISTRY TESTS, RECEIVED STANDARD FUEL PELLETS^a

<u>Rod Number</u>	<u>Lot Number</u>	<u>H₂O (ppm)</u>	<u>N₂ (ppm)</u>	<u>F⁻ (ppm)</u>
1475	K05LR03	<10	12	<1
1521, 1549	K05LR05	<10	13	<1
Other ^b	K03LR01, 03	<10	18-28	<3
	K04LR04, 08	<10	<10	<1
	K05LR08	<10	<10	<1

-
- a. Analyses performed on (5 pellet) samples of 2.0 w/o fuel lots in January 1962 ~ 1 month before drying and loading operations for the fuel rods.
- b. Rods manufactured with fuel pellets from these lots were not examined.

TABLE 14

CHEMISTRY TESTS, ARCHIVE FUEL PELLETS^a (FEBRUARY 1968)

<u>Assembly,</u> <u>w/o U-235</u>	<u>Lot</u> <u>Number</u>	<u>Density</u> <u>g/cc</u>	<u>H₂</u> <u>(ppm)</u>	<u>H₂O</u> <u>(ppm)</u>	<u>N₂</u> <u>(ppm)</u>	<u>F⁻</u> <u>(ppm)</u>	<u>Cl⁻</u> <u>(ppm)</u>	<u>Approximate Production Date</u>
Standard, 2 w/o	K04LR06	10.31	1	9	<40	<5	<5	~1961
		10.28	0.7	12	<40	<5	<5	
Special, 4.7 w/o	6	10.25	3	6	<40	<5	<5	~1965 - 1966
		10.27	1	10	<40	<5	<5	

a. NOTE: Identical samples (2) of each type pellet were analyzed. H₂O content was determined by vacuum drying; H₂ and other volatile impurities were determined by high temperature (1000°C) anneal and mass spectrometer.

standard pellets (lot K04LR06) were not from any of the lots loaded into the particular fuel rods examined in this program.

A summary of the weld closure records for the rods examined is shown in Table 15. The records indicate that both of the failed elements 1475 and 1549 (also unfailed rod 1521) were located in two welding lots (50 rods per lot) which showed a very high incidence of weld defects during initial welding of the upper end plugs. Acceptance rates on end-plug seal welds for all other lots was 90 to 95 percent on initial welding compared with 15 percent for the two welding lots containing the failed elements.

A total of thirteen lower end plugs welded to surplus standard cladding was radiographed using current radiographic equipment for production fuel rods. Of these thirteen welds, three had been rejected at the time of manufacture and ten had been accepted and later not loaded for reasons other than bottom end plug weld integrity. On current examination, five of the originally acceptable ten welds were judged unsatisfactory for excess porosity (Table 11). An examination has been made of the original x-ray film for the thirteen welds and for the six fueled rods currently examined. It was established that the original inspection decisions were justified on all rods on the basis of the original x-ray films, but there has been a significant improvement in the sensitivity of the radiographic equipment since the time of manufacture of the CVTR standard rods (March 1962).

Table 15 indicates that there was an electrode/weld contact during the rewelding of the upper end plug of rod 1475. This could be of significance to the failure located at the upper end plug. There is no evidence that the welding box atmosphere was inadequate during rod welding to the extent that contamination of the weld by nitrogen or oxygen was experienced. Analytical data on the purity of the helium atmosphere during end plug welding have not been found.

However, the atmosphere of unirradiated 1.5 w/o standard assembly fuel rods has been analyzed (Table 16). Because the rod had been corrosion tested in 750°F steam, the gases found may partly reflect volatile impurities driven off from the fuel and Al_2O_3 spacers within the rod, as well as the intrinsic

TABLE 15

WELDING DATA ON STANDARD FUEL RODS

Assembly 502-1-18

<u>Rod</u>	<u>Welding Lot</u>	<u>LEP</u>	<u>1st Welds UEP</u>	<u>FP</u>	<u>LEP</u>	<u>Rewelds UEP</u>	<u>FP</u>	<u>Remarks</u>
1475	30	R (porosity)	R (under-cut)	S	1	1	--	Electrode contact on reweld of upper end plug. Salvage lot 28. LEP rewelded in repair lot 22.
1521	31	R	a	a	1	a	a	
1591	32	S	S	S	---	---	--	

Assembly 502-2-31

<u>Rod</u>	<u>Welding Lot</u>	<u>LEP</u>	<u>1st Welds LEP</u>	<u>FP</u>	<u>LEP</u>	<u>Rewelds UEP</u>	<u>FP</u>	<u>Remarks</u>
1257	26	S	S	S	---	---	--	
1398	28	S	R (porosity)	S	---	1	--	UEP repaired in repair lot 25.
1549	31	S	R (under-cut)	S	---	1	--	UEP repaired in salvage lot 32.

a. Rod 1521 is not recorded for initial welding or upper end plug and filler plug in lot 31, but is recorded on salvage lot 31. The records do not clarify if this is a repair weld or an initial weld.

R = Reject
S = Satisfactory

UEP = Upper end plug
LEP = Lower end plug

FP = Filler plug

TABLE 16

ANALYSIS OF INTERNAL ATMOSPHERE IN CVTR CORE I (1.5 w/o) FUEL RODS^a

Sample No.	(mol. %)							
	<u>He</u>	<u>H₂</u>	<u>N₂</u>	<u>CO</u>	<u>CO₂</u>	<u>Ar</u>	<u>O₂</u>	<u>CH₄</u>
88	98.7	<0.01	0.41	<0.01	0.21	<0.01	<0.40	0.55
12	99.0	<0.01	0.66	<0.1	0.12	<0.01	0.20	<0.01

-
- a. The rods punctured for analysis were unirradiated 1.5 w/o rods manufactured for Core I. The analyses were performed on July 11, 1968, at the Battelle Pacific Northwest Laboratories, Richland, Washington. (Personal Communication, M. D. Freshley).

weld atmosphere. Examination of sections of the clad bore revealed evidence of oxidation, particularly at fuel pellet interfaces.

5.2 SPECIAL ASSEMBLY RODS

5.2.1 Ultrasonic and Metallographic Examination

A total of ten tubes comprising seventy feet of special assembly archive tubing have been ultrasonically tested against the specified standards in use at the time of the order. No rejectable defects were noted, but a few indications in excess of 50 percent of the specified standard were checked metallographically when they did not coincide with observable outer surface marks. Metallographic examination revealed a few bore pits of 0.001 to 0.0015 inch depth, but no incipient cracks of the type observed in the standard cladding.

Only one of the ultrasonically tested tubes was of 0.017 inch thin-wall Zircaloy-4 (reference irradiated rods 83.831 and 83.832). Since there was some doubt of the ultrasonic test sensitivity for this tube size, the tube was examined at five random locations. Some roughness was noted in the form of minor pitting to a depth of 0.0005 inch.

5.2.2 Corrosion Resistance, Hydride Orientation, and Fluoride Contamination

Samples from five pieces of archive tubing were tested for three days in 750°F, 1500 psi steam after: a) pickling on the bore and outside diameter and b) pickling on the outside diameter only. All the samples showed acceptable weight gains (Table 17). A black lustrous corrosion film was present on the surface of each of the samples. One of the tube samples which was pickled only on the outside surface contained a white corrosion product on the bore -- an indication of fluoride contamination during vendor pickling.

Both the 0.024 inch and 0.017 inch thick cladding exhibited radial hydriding characteristics in the outer rim of the cladding. This characteristic is believed to be a function of the extent of tube straightening after final stress-relief annealing. The bulk of the tube wall in the 0.017 inch cladding exhibited random hydride patterns. (Figures 31, 32, and 34 show typical orientation in standard and both types of special assembly cladding.)

TABLE 17

CORROSION TESTS OF CVTR SPECIAL ASSEMBLY CLADDING

Test Condition: Three days at 750°F, 1500 psi steam^a

Tube Number	OD/ID Pickled, mg/dm ²		OD Only Pickled ^b , mg/dm ²	
24.733	15.5	14.8	16.6 ^c	16.3 ^c
53.863	14.1	13.4	17.4	16.7
10-D-5	13.1	15.7	16.3	18.1
10-D-23	13.5	16.7	16.7	16.7
10-D-21	13.5	13.5	17.0	17.4

a. Maximum acceptable weight gain for this test = 22 mg/dm².

b. The tube bore was thoroughly degreased.

c. White line of corrosion product noted on the tube bore.

NOTE: All the above tubes were H-33 standard wall Zircaloy-4 clad.
The tubes 24.733 and 53.863 were loaded archive fuel tubes;
the remaining tubes were archive dummy tubes.

X 100

Figure 34. Hydride Orientation in Archive Special Assembly Thin-wall Zircaloy-4 Clad

Six samples of 0.024 inch archive tubing and two samples of 0.017 inch tubing were analyzed for bore fluoride contamination. One 0.024 inch tube (24.733), which had shown a white streak of corrosion product on the tube bore, was analyzed at $0.75 \mu\text{g}/\text{cm}^2$, and both samples of 0.017 inch cladding exhibited higher fluoride contamination levels. The results for fluoride analysis on all cladding samples are shown in Table 8.

5.2.3 TENSILE PROPERTIES

Tensile tests have been carried out on the 0.024 inch cladding (standard wall) at temperatures of 70°F, 600°F, and 725°F, and a strain rate of 0.005 inch/inch/minute. Due to the limited availability of the 0.017 inch (thin-wall) cladding tests were carried out at 600°F only. The results are presented in Table 18, and three points can be made. First, the 0.024 inch cladding strength does meet the specified minimum yield strength of cold worked, stress-relieved Zircaloy-4 at room temperature. Second, all samples show an acceptable ductility, both at room temperature and at elevated temperatures. Third, the 0.017 inch cladding has significantly better yield strength and tensile strength than the 0.024 inch cladding -- a characteristic which is a function of the method of tube fabrication.

5.2.4 DUMMY RODS AND FUEL ROD COMPONENTS

Three dummy rods (i.e., sealed rods containing Al_2O_3 pellets) were punctured to assess the quality of the welding atmosphere. One of the rods had been pre-oxidized for three days in 750°F oxygen. The results are shown in Table 19. It was noted that there was evidence of gaseous release from the Al_2O_3 dummy pellets during the corrosion test: the inside of the tubing was oxidized, and the pellets were discolored. Confirmation of the release of oxidizing gases from the pellets was obtained by analysis of fuel spacer pellets and dummy rod Al_2O_3 pellets heated to 482°F and 750°F (Table 20).

Analysis of archive special end plugs and Inconel springs revealed that the total hydrogen content was <10 ppm.

Analysis of the special assembly standard wall cladding for hydrogen indicated values of 10 - 12 ppm on a nonoxidized rod and 9 - 15 ppm on a

TABLE 18

TENSILE DATA ON ARCHIVE SPECIAL ASSEMBLY CLADDING

<u>Material</u>	<u>Test Temp. (°F)</u>	<u>0.2% Offset (psi)</u>	<u>UTS</u>	<u>Total Elong. (%)</u>	<u>Uniform Elong. (%)</u>
Special assembly	RT	73,500	95,000	19.3	7.3
	RT	77,000	99,800	19.2	9.6
Standard wall cladding (H-33)	600°F	43,600	53,300	12.4	5.7
	600°F	46,000	56,700	11.4	6.9
	725°F	41,600	49,400	12.6	4.9
	725°F	45,050	52,700	13.9	5.2
Special assembly thin wall cladding (H-33)	600°F	54,400	72,300	12.6	5.6
	600°F	54,950	72,800	12.8	7.5

TABLE 19

ANALYSIS OF ATMOSPHERE IN SEALED ARCHIVE DUMMY RODS

<u>Sample</u>	<u>He</u>	Molar Percentage					<u>Organics</u>
		<u>H₂</u>	<u>N₂</u>	<u>CO</u>	<u>CO₂</u>	<u>O₂</u>	
11 DH ^a	98.08	0.24	1.20	0.27	0.24	ND	0.1-0.5
12 D3	99.73	0.27	ND	ND	ND	ND	ND
10 D23	99.79	0.21	ND	ND	ND	ND	ND

a. 11DH corrosion tested for three days in 750°F oxygen prior to puncture

TABLE 20

ANALYSIS OF Al_2O_3 PELLETS USED IN FUEL AND DUMMY RODS

<u>Temperature</u>	<u>Total Gas, cc/gm</u>	<u>Molar Percentage Gas Release</u>					
		<u>H₂</u>	<u>N₂</u>	<u>O₂</u>	<u>CO₂</u>	<u>A</u>	<u>CH₄</u>
Fuel Rod Al_2O_3	Spacer Pellets:						
482°F	---	0.36	43.08	9.04	41.22	<0.01	6.65
750°F	0.008	0.72	42.99	1.30	42.34	0.50	13.38
Dummy Rod Al_2O_3	Pellets:						
482°F	---	<0.01	25.33	1.00	63.67	<0.01	10.0
750°F	0.018	1.78	22.73	0.21	68.09	0.42	6.76

pre-oxidized rod. Analysis of thin-walled cladding indicated an initial hydrogen level of 6 - 16 ppm.

The oxide film quality on all pre-oxidized dummy archive rods was excellent in that a black lustrous film was present and there was no evidence of white corrosion product. An attempt was made to measure the thickness of the pre-oxidized film on dummy rods, but the film was too thin for measurement by standard laboratory procedures. Thickness of the film is estimated to be less than 0.3×10^{-4} inches.

Two pre-oxidized and two unoxidized sealed dummy rods were measured for ovality and outside diameter by micrometer measurements at 6 inch intervals along the rods. The mean ovality on the pre-oxidized rods was 0.004 inch (with maximum values of 0.0008 - 0.0011 inch in a few positions) compared with a mean of 0.0003 inch (maximum value of 0.0008 inch) on the unoxidized rods. These changes are not significant and pre-oxidization treatment did not induce ovality in the rods.

A total of nine end plug welds comprising seven lower end plug welds and two upper end plug welds were radiographically checked against acceptance standards used during the fabrication (Table 21). Both upper end plug welds on dummy rods were rejectable, one for porosity and one for ID undercutting. One of the lower end plug welds was rejectable for porosity. The absence of adequate records and the fact that these rods were not used in the assemblies means that these data cannot be used to evaluate the weld quality of the fuel rods used in the special assemblies. A number of samples containing the reference end plugs were burst-tested at room temperature in a closed-end test. All samples failed in the heat affected zone of the weld. There is no evidence, for the two defected welds tested, that the deficiencies resulted in different strength and ductility than would result from acceptable welds. The welds in the special assembly rods show (Table 21) a comparable strength but lower ductility than the welds in the standard assembly rods (Table 11).

5.2.5 Processing Records

The standard wall cladding (Reference: Failed rods 24.732 and 53.861) was made by a plug-drawing process and resulted in approximately 40 percent cold

TABLE 21

CLOSED-END BURST TESTS ON SPECIAL ASSEMBLY END PLUG WELDS

<u>Rod No.</u>	<u>Radiograph Disposition</u>	<u>UTS (psi)</u>	<u>Elongation (%)</u>
11-D-? ^a (lower)	-	105,000	9.3
12-D-3 (lower)	S	109,000	12.7
10-D-45 (lower)	S	102,000	10.8
10-D-21 (lower)	-	-----	----
12-D-2 (lower)	S	-----	----
10-D-21 (upper)	R - ID undercut	108,000	10.1
10-D-5 (upper)	R - porosity	112,000	8.9
24.733 (lower)	S	-----	----
53.863 (lower)	S	-----	----
10-D-5 (lower)	R - porosity	-----	11.4
10-D-23 (lower)	S	-----	----
10-D-48 (upper)	-	110,000	9.0
10-D-48 (lower)	S	-----	----

a. Identification mark obscured by wire wrap weld

Key R = Reject

S = Satisfactory

work prior to a stress relief for four hours at 950°F. This process is known to result in marginal conformance to hydride orientation requirements and low tensile properties at elevated temperatures. The specification had no mandatory requirement for high temperature tensile properties (725°F). The following points have been noted by examination of test reports and inspector's notes. First, one tubing lot has an abnormally low ductility (8 percent) associated with low strength at 725°F. (This low ductility has not been confirmed on our own tensile tests.) Second, 50 percent of the tube failed initial vendor ultrasonic examination, but was accepted. The inspector's report indicates that the rejections were related to superficial defects and the ultrasonic tests carried out in this investigation substantiate this judgment.

The thin-wall cladding (Reference Failed rod 83.831) was made by tube-reduction with 75 percent cold work and stress-relief anneal for four hours at 915°F. The only point of note is that the fluoride test requirement on the tube bore was deleted at the vendor's request.

The fuel rod loading data for special assembly rods destructively examined are given in Table 22, and inspection reports on fuel pellet moisture determinations are shown in Table 23. Fuel pellet dimensions are presented in Table 24. There appears to be no direct evidence of a causal relationship between fuel pellet moisture content and rod failure, but two points may be noted in this respect. First, there is a discrepancy between the number of pellets formally released for loading after moisture determination and the number of pellets loaded, which could mean that some pellets may have been loaded without clearance for moisture content. Rods which have been specifically identified as having low density, low diameter pellets (which from the records were not formally released) are numbers 33.833, 83.831, and 83.832. Second, a series of six fuel rods, including failed rods 53.861 and 83.831, had late upper end plug vent hole closures. There was an unspecified time lag between welding of upper end plugs onto loaded fuel rods and the final weld closure of the vent hole. During this time, the unsealed rods were transported from the fabrication laboratory to another building for inspection, and then returned.

TABLE 22

SPECIAL ASSEMBLY FUEL ROD LOADING DATA

<u>Assembly</u>	<u>Rod Number</u>	<u>UO₂ Density (% T.D.)</u>	<u>Enrichment (% U-235)</u>	<u>Fuel Weight (gm.)</u>	<u>Fuel Length (in.)</u>	<u>Al₂O₃ Spacers^(d) (No.)</u>
G-1	44.731	93	4.7	1894.3	77.8125	2
G-1	24.732	93	4.7	1893.7	77.8125	2
G-1	33.833 ^(a)	93	3.8	1932.3 ^(b)	81.125	2
G-2	13.831	93	3.8	1968.5	81.156	2
G-3	53.862 ^(a)	96	3.8	2022.1	81.0625	2
G-3	53.861 ^(a)	96	3.8	2046.1	81.0625	2
G-3	24.734 ^(a)	93	4.7	1897.0	77.875	2
G-4	83.832 ^(a)	93	3.8	2135.2 ^(c)	81.156	1
G-4	83.831 ^(a)	93	3.8	2172.4 ^(c)	81.125	1

a. Late upper end plug vent hole weld.

b. Some low density (unreleased) pellets used.

c. Some low diameter (unreleased) pellets used.

d. Nominal dimensions of spacer: 0.300" L., 0.4285 in. OD, 0.187 in. ID.

TABLE 23

MOISTURE CONTENT OF UO_2 FUEL PELLETS

<u>Pellet Lot No.</u>	<u>Fuel Rod No.</u>	<u>Moisture Test Results (ppm)^b</u>						<u>Weighted Average Moisture (ppm)</u>
1 ^a	53.861	1.8	1.1	<1.0	<1.0	<1.0	<1.0	1.0
	53.862							
2 ^a	64.761	3.3	2.8	2.6	2.6	2.1	2.9	2.6
	64.762							
3 ^a	73.832	18.0	17.5	8.8	8.3	5.9	0.5	9.2
	73.833	1.4	1.1	1.0	0.8	0.8	0.8	1.0
4 ^a	83.831	1.0	1.6	1.0	1.3	1.1	1.0	1.3
	83.832							
5	W13.831	22.4	19.9	12.0	6.1	3.4	3.5 ^c	11.2
	13.831							
	13.832	2.5	2.3	1.8	1.4	1.3	1.3	2.4
	13.833							
	13.834	3.3	2.0	1.8				1.8
	13.835							
	13.836							
	13.837							
	13.838							
	13.839							
6	24.732	5.0	5.0	4.5	3.4	3.0	0.5	3.6
	24.734							
	44.731							
	44.732							

a. Quantity of pellets released less than quantity required.

b. A five-pellet sample was normally taken for moisture checks. For some pellet lots, more than one such sample was taken.

c. This lot was redried after initial results were obtained of 47.1, 23.4, 21.0, 11.0, 6.5, and 6.0 ppm.

TABLE 24

DIAMETERS OF SPECIAL ASSEMBLY FUEL PELLETS

<u>Lot No.^a</u>	<u>Specified Diameter, in.</u>	Percentage of all Pellets with Diameter Less Than				
		<u>0.4323</u>	<u>0.4326</u>	<u>0.4329</u>	<u>0.4332</u>	<u>0.4335</u>
1	0.4324 - 0.4334	0	13	70	100	---
2	0.4324 - 0.4334	5	7	35	78	100
5	0.4324 - 0.4334	0	6	52	94	100
		<u>0.4555</u>	<u>0.4558</u>	<u>0.4562</u>	<u>0.4566</u>	<u>0.4570</u>
3	0.4560 - 0.4570	0	5	76	100	100
		<u>0.4470</u>	<u>0.4473</u>	<u>0.4475</u>	<u>0.4478</u>	<u>0.4480</u>
4	0.4470 - 0.4480	0	30	60	100	100
		<u>0.4312</u>	<u>0.4315</u>	<u>0.4317</u>	<u>0.4320</u>	<u>0.4322</u>
6	0.4312 - 0.4322	0	33	70	100	100

a. See Table 23 for correlation of lot number and rod number.

Since these transfers were conducted during cold weather, it is postulated that moisture condensed within the rod each time the cold rod was returned to the room temperature environment. In addition, there was the possibility of internal organic contamination. Four of the mentioned six rods exhibited internal hydriding or excessive bore oxidation; two of the rods failed. Documentation of the internal hydrogen source for rod 24.732 is not available.

Table 22 lists the number of Al_2O_3 spacers used at the top of the fuel column for each special rod. Pre-loading moisture analysis gave 12.3 to 16.0 ppm for the spacers.

Reports of other manufacturing deviations on the failed and unfailed sister rods in the special assemblies follow:

Rod 53.861 (Failed Rod in Assembly G-3)

This rod was reported out of specification for ovality and camber at the top end plug due to the end plug being "cocked". This deviation was "subject to correction if it did not fit during assembly", but there is no evidence of corrective action. Deviations also were noted by oxide film quality, wire wrap pitch, and overall "envelope" dimensions. There also was a deviation on the position and orientation of the end plug hole (for the end of the wire wrap) in the bottom end plug.

Rod 53.862 (Unfailed Sister Rod to 53.861)

This rod was reported out of specification for camber, quality of oxide film, and orientation of the end plug hole. It was also noted that the welding of the wire wrap to the lower end plug was done in air -- completely destroying the end plug. There is no recorded evidence of subsequent rework and reinspection on this rod. The post-irradiation examination revealed that a lower end plug of slightly different design had been substituted.

Rod 44.731 (Unfailed Rod in Assembly G-1)

The only deviations reported were for length of rod and for failure to meet space envelope dimensions.

Rod 83.831 (Failed Rod in Assembly G-4)

This rod was rewelded at the upper end plug and the filler plug after evidence of porosity and deviations were also recorded for wire wrap pitch and space envelope. Some identity confusion was reported for this rod during processing.

Rod 83.832 (Unfailed Sister Rod to 83.831)

Deviations were reported for this rod on oxide film quality, end plug hole orientation, camber, and wire wrap pitch.

The deviations noted on the above rods were common to a number of unfailed rods in the assemblies. The most common deviations reported on many of the rods were:

1. Porosity on both end plug seals -- these were repaired;
2. Minor deviation from a black lustrous oxide film -- these are understood to be local areas of white speckle around repaired areas;
3. Dimensional deviations for wire wrap pitch, envelope, camber, ovality, cocked end plugs, and location of wire wrap holes in the end plugs.

An attempt has been made to correlate the type of deviation noted with the incidence of failure in the rods. There is no significant correlation although it has been noted that three rods showing internal hydriding did not have bleeder holes sealed immediately following the circumferential upper closure weld.

It is possibly significant that all of the failed rods were among those to be completed and accepted last. No failures have been observed on the series 13.831 through 13.839, and 44.731 through 44.732. The rods in these two series were the first rods started and were completed without undue difficulties in fabrication. The possible use of unreleased or substandard pellets on later rods has been mentioned previously.

SECTION 6

DESTRUCTIVE EXAMINATIONS

6.1 CLAD HYDROGEN/DEUTERIUM

A valuable analytical method for the post-irradiation examinations was hot extraction gas analysis of selected Zircaloy clad samples on the failed rods, particularly at defected locations. Because the CVTR coolant was heavy water, it was possible to gain greater insight into the nature of the massively "hydrided" regions by isotopic analysis of the extracted gases. The presence of deuterium (mass two) indicates clad-coolant reaction resulting from external corrosion or failure of the rod. The presence of large amounts of hydrogen (mass one) could come from only two sources: hydrogen inherent in the fuel rod components and fuel, or an internal contaminant. Clad samples were taken from routine locations for both failed and unfailed rods and from defect areas on the former. Sample size was normally a 1/4 inch ring of clad, further subdivided into four pieces; in the case of defect (hydrided) regions, attempts were made to isolate the defect for separate analysis.

After preparation, clad samples were transferred to the Analytical Services Laboratory for hydrogen and deuterium analysis by the hot extraction method. The evolved gases were recovered through a palladium thimble, allowing only the hydrogen isotopes to be collected in the gas measuring volume; the total gas concentration was determined from manometric measurement and the weight of the sample. A subsample of the gases collected from each clad was then analyzed in a CEC 21-103C mass spectrometer to obtain the ratios of masses 2, 3, and 4 (H_2 , HD, and D_2 , respectively).

It should be noted that molecular ratios of the gas as analyzed are not necessarily representative of the clad itself; a physical homogenization of the gases occurs during collection by hot extraction. Furthermore, experiments with known mixtures of hydrogen and deuterium indicate that the molecular distribution was subsequently altered when the extracted gases were passed

through the vacuum system. The equilibration of the hydrogen and deuterium was found to vary according to the original composition of the mixture. The relative abundance of hydrogen and deuterium atoms was not changed, however. Thus, the calculated totals of each are valid.

The majority of the H/D analysis data have been published in Appendix A, WCAP-3850-3. Included is information on sample weight and axial location, molecular abundance of the gas species, and the separate H and D contents expressed as ppm. Additional samples are reported in Appendix C of this report, for rod 1591 (upper end plug), 53.862 (upper end plug), 83.831 (14-inch blister), and 83.832 (burst section 17-5/8 - 25-5/8 inches).

The results of hydrogen/deuterium analyses are shown graphically in Appendix C for each failed assembly. In each figure the maximum (per sample) H/D content in ppm observed at a given location has been plotted as a function of axial position on the fuel rod. A solid line has been drawn through the failed rod H₂ data points; results from (unfailed) sister rods are shown for comparison. The axial locations of clad defects, anomalies, and peak kw/ft and burnup for the failed rod are shown on the upper horizontal scale. Deuterium content (solid symbols) is characteristically low at normal locations and on unfailed rods, with extreme peaks at defect regions on the failed rods. The most significant feature of these graphs is the discrepancy in hydrogen (mass one) content between failed and sister rods. In the case of the standard assembly 502-1-18 the difference is limited to axial locations at or in the immediate vicinity of clad defects; for other cases (e.g., special assembly G-1), the difference is general over most of the rod length. The magnitude of the differences between expected (unfailed rods) H₂ content and local extremes in the failed rods is greater than can be accounted for by the accuracy of the analyses and normal variations inherent in fuel rod components. In fact, the actual H₂ content in localized defect regions may be several times higher than the reported values because of the inability to completely isolate the defects for analysis. This is reflected in the deuterium content found in defect regions which is considerably less than the theoretical maximum of ~16,000 ppm. A fuller discussion of the H/D results is given under Sections 7.2 and 7.3.

6.2 BURNUP PERFORMANCE, FAILED CVTR ASSEMBLIES

6.2.1 Burnup Analyses

Twenty-two burnup samples were cut (clad and fuel) from the 12 CVTR rods destructively examined in the program. The samples were analyzed radiochemically and by mass spectrometer for heavy isotopes of U and Pu, for Nd-148, and (selectively) for Cs-137. The data from these analyses were reduced with the REBUF^[6] computer code to determine burnup and various other nuclear parameters. Table A-1 of Appendix A summarizes the constants used in the reduction of the measured data. The constants α^{49} , α^{41} and burnout factors for U-236 and Pu-242 were obtained from a LEOPARD^[7] calculation of the Yankee unit cell and were used as a substitute for unavailable CVTR fuel constants. The uncertainties attached to these constants (approximately ± 10 percent relative) are believed small enough to justify their use in reducing the CVTR data.

Table A-2 of the Appendix gives the identification and axial location of each burnup sample, and the nominal fuel composition for each rod. The measured pellet burnups are listed resulting from the three types of measurements, as well as calculated U-235 fractional depletion. The values of heavy element calculated burnup are generally high relative to the Nd-148 data; however, for all but two samples the two methods agree within the uncertainties of the data. The Cesium-137 burnup values are relatively low in the high power special assembly rods, presumably a reflection of the volatility and consequent tendency to migrate of this isotope. Cs-137 analysis was not performed on the failed standard fuel, because of the susceptibility of leaching by the coolant. The most accurate burnups are believed to result from the Nd-148 analysis because of that isotope's thermal and nuclear stability; these burnup values have been chosen exclusively for this study.

6.2.2 Gamma Scans and Normalized Fuel Rod Burnups

Gamma scanning was carried out on fuel rods from the failed CVTR assemblies using a miniature ionization chamber detector and recording the instantaneous current produced as the chamber was raised or lowered, at constant velocity,

along a vertical aluminum tube containing the fuel rod under examination. Initial scans were made with the wrapper wires in place and were used to determine the radial differences in total (integrated) activity in the standard assemblies. These data subsequently were used to assist in the selection of rods for destructive examination. Examination of these scans showed that the wrapper wires were producing perturbations in the measured axial activity levels. Therefore, rods selected for destructive examination were rescanned after removal of the spacer wires. Each rod was scanned three times (90° apart), allowing compensation for the flux skew across individual rods. Special holders were fabricated to center each rod in the holding fixture. Rods examined in this way were:

<u>Assembly No.</u>	<u>Fuel Rod No.</u>
Special G-1	33.833, W13.831, 44.731, 24.732
Special G-3	13.837, 53.861, 53.862, 24.734, 64.761, 64.762
Special G-4	13.838, 13.839, 83.831, 83.832
Standard 502-1-18	1475, 1591, 1521
Standard 502-2-31	1257, 1398

Fuel rod No. 24.732 had been fractured into four sections of varying length during a handling accident, so each section had to be scanned individually. The section 13 - 17 inches from the bottom of the rod was not scanned because of its short length; in reconstituting the axial gamma activity profile for this rod, it was necessary to extrapolate over this section. Failed rod No. 1549 from standard assembly 502-2-31 was scanned only once, with its spacer wire intact. The axial gamma scans not previously reported (averaged over the 0° and 180° orientations) are given in Figures A-1 through A-6. Table A-4 summarizes the quantitative gamma scan data, including axial peak-to-average ratios for each fuel rod.

Table A-5 lists earlier (August 16 and 21, 1967) gamma scanning performed on all 38 standard fuel rods. The spacer wires were intact and rod orientation was not controlled for these measurements, thus making them unsuitable for determining axial profiles. The total (integrated) activities of each rod, however, can be used to estimate rod-to-rod radial power distribution within an assembly. Total activity and local-to-average activity are listed in the

table; rod locations within the assembly are shown in Figure 2. Highest measured local-average rod activity is approximately 1.13 and 1.14 for assemblies 502-2-31 and 502-1-18, respectively. Table A-6 summarizes similar gamma scanning with spacer wires intact performed on fuel rods from special assembly G-1.

Fuel rod average burnups have been inferred from the axial gamma scans and the measured Nd-148 burnups at specific axial locations (Table A-2). These are given in Table A-3. With the appropriate peak/average ratio from the gamma scans, the rod peak burnups have been calculated. The data are given in the last column. An exception to the method of obtaining peak and average rod burnups was used in the case of rod No. 1475, standard assembly 502-1-18. The measured burnups for this rod were anomalously low: using the Nd-148 results, the average burnup for rod No. 1475 would be only 8010 MWD/MTU. Instead of relying upon the burnup data, the relative gross gamma activity of rod 1475 with respect to sister rods Nos. 1591 and 1521 was used to calculate burnup. As an approximation, the average burnup for rod No. 1475 has been taken as 9 percent lower than the mean burnup for Nos. 1591 and 1521.

6.3 FISSION GAS RESULTS, CVTR SISTER FUEL RODS

6.3.1 Fission Gas Analysis

Table 25 summarizes the fission gas collection and analysis performed on (unfailed) sister rods from each of the failed assemblies. Included are the total volume of gas collected, and mass spectrometric analyses giving the constituents of the gas extracted from the rods and isotopic content of the fission gases, xenon, and krypton. Each of the fuel rods was punctured by drilling through the cladding (plenum) under vacuum in a calibrated system. Volumes of the released gases then were measured, and individual samples were collected and analyzed at the Westinghouse Research and Development Center.

In the case of standard fuel rod No. 1398, a duplicate sample was analyzed, because of the unusual results. The data in Table 25 are an average of the two samples which gave similar results. Each of the fuel rods had been back-filled with one atmosphere of helium gas prior to irradiation. With the exception of rod No. 1398, the quantities of helium collected from the various

TABLE 25

FISSION GAS ANALYSIS, SISTER FUEL RODS

Assembly Rod No.	G-1 44.731	G-1 33.833	G-3 53.862	G-3 24.734	G-4 83.832	502-2-31 1257	502-2-31 1398 ^a	502-1-18 1521	502-1-18 1591
Total Volume ^b , cc	163.2	105.0	69.9	146.0	60.6	15.43	71.0	22.1	15.82
He - mol. %	16.87	49.34	58.89	34.25	68.52	93.07	0.235	73.79	75.99
H ₂	-	-	-	-	-	-	0.370	-	-
O ₂	8.70	0.25	0.23	0.39	0.25	1.07	7.505	1.75	2.65
N ₂	30.56	0.70	0.53	0.81	0.62	2.11	89.760	3.10	5.70
A	0.41	-	-	-	-	0.06	1.110	0.08	0.11
CO ₂	0.09	-	-	-	-	0.04	0.055	-	-
K _r	5.58	6.95	5.48	8.43	4.13	0.45	0.105	2.25	1.70
X _e	37.79	42.76	34.87	56.12	26.48	3.20	0.860	19.03	13.85
K _r -83	13.78	13.72	13.80	13.81	13.45	12.66	13.260	12.19	12.75
-84	27.29	27.61	27.31	27.54	27.51	29.26	31.060	28.93	30.43
-85	6.94	6.80	6.83	6.78	7.00	6.55	7.955	6.60	6.96
-86	51.99	51.87	52.06	51.87	52.04	51.53	47.725	52.28	49.86
X _e -131	10.66	11.54	10.87	10.73	10.92	10.13	11.460	9.08	9.10
-132	16.67	18.13	16.91	16.54	17.40	17.96	19.995	17.09	17.20
-134	29.85	30.98	29.27	29.55	29.24	29.52	26.265	26.96	33.24
-136	42.82	39.35	42.95	43.18	42.44	42.39	42.780	46.87	40.46

a. Averaged over two samples, with similar results.

b. At S.T.P.

fuel rods are commensurate with the initial backfill, as calculated from as-fabricated cold (internal) void volumes. This indicates that all of the free internal gases were collected from the rods. The results for special rod 44.731 in Table 25 are a correction of previously reported data (WCAP-3850-2).

6.3.2 Calculated Fission Gas Release

The percentages of total fission gases released have been calculated and are tabulated in Table 26. The method was to divide the quantity of $X_e + K_r$ collected by the total atoms of these gases produced by fission. For this purpose, the average burnups, as determined from Nd-148 analysis and a fission yield of 0.3 atoms per fission event, were used. No radiochemical burnup analyses were performed on rods No. 33.833 and No. 24.734; gross gamma activity measurements were used to estimate the burnups for these respective rods.

The excessive amounts (>1 mol. percent) of oxygen and nitrogen in standard fuel rods 1521, 1591, and special rod 44.731 indicate air leaks either in the collection system or the sample bulb sent out for analysis by mass spectrometer. Accordingly, the measured abundances of fission gas for these rods were corrected (upward) for dilution of the samples by air (O_2 , N_2 , Ar, and CO_2).

The results reflect the differences in power operation. For the special high-power rods, fission gas release is approximately twice as high for the 4.7 w/o enriched rods (which experienced extensive columnar grain growth) as for the 3.8 w/o enriched rods. Rod 83.832 (assembly G-4) release seems unusually low as compared to the other 3.8 w/o rods, 33.833 and 53.862. The standard fuel rods show extremely low fission gas releases, as expected for their low power operation. It is apparent from these results that fission gas internal pressure was not a factor in the integrity of the CVTR fuel rods.

6.3.3 Fuel Rod No. 1398, Assembly 502-2-31

As shown in Table 25, standard fuel rod No. 1398 (502-2-31) contained primarily nitrogen gas, with minimal amounts of helium (backfill) and fission gases. The presence of significant amounts of oxygen and argon gases might

TABLE 26
CALCULATED FISSION GAS RELEASE

Assembly No.	Rod No.	Avg. Burnup MWD/MTU ^a	Calculated % Release: $X_e + K_r^b$
G-1	44.731	7,870	29.89
	33.833	6,850	14.95
G-3	53.862	5,910	12.61
	24.734	7,150	26.34
G-4	83.832	7,490	4.40
502-2-31	1257	10,800	0.09
502-1-18	1521	14,330	0.57
	1591	14,440	0.31

-
- a. As measured from Nd-148 analysis; burnup for Rods 33.833 and 24.734 are estimated from gross gamma scan activities.
- b. Corrected for air leakage when calculated N_2 or O_2 exceeded 1 mol. %.

indicate an air leak in the sample, but the N_2/O_2 ratio far exceeds the theoretical value of 3.72. The percentage fission gas release has not been calculated (Table 26) because the collection results cast doubt upon the validity of the sample.

The metallographic samples from rod No. 1398 indicate evidence of internal contaminants: a thick (up to 3×10^{-4} inches), uniform, ID oxide film and high clad hydriding (not confirmed by hot extraction analyses). Even if one assumes an organic contaminant to account for these observations, however, the (corrected) amount of helium gas actually collected is too low (by two orders of magnitude) for the initial backfill at one atmosphere. Similarly, the assumption of an air leak (with the excess nitrogen assumed to be from an internal source) indicates an insufficient amount of He gas in the sample. This indicates that not all of the internal fuel rod atmosphere was successfully collected, making a fission gas release calculation from the data in Table 25 meaningless.

The possibility of a contaminated weld atmosphere in rod No. 1398 seems remote. Although welding records indicate an upper end plug reweld (the initial weld had been rejected for porosity), the vent hole plug weld was satisfactory -- thus, indicating fuel rod internal atmospheric integrity. Practice welds were routinely required to test the helium atmosphere before rod welding and vent hole closure; the rods were welded in lots of 50 within a glove-box. Thus, a bad weld atmosphere seems unlikely as an explanation for the observed results in rod No. 1398.

6.4 INVESTIGATION OF CLAD CRACK MORPHOLOGY

Large ID longitudinal cracks occurred at $\sim 26 \frac{1}{2}$ inches on three of the four failed rods (1549, 1475, 83.831) and two unfailed rods (1521, 83.832) which operated during the rod banking experiment on November 21, 1966. None of the cracks actually penetrated the clad, although extending as much as $\frac{2}{3}$ into the clad wall. Incremental grinding of transverse metallographic specimens indicated an axial length of at least 50 mils for the standard clad cracks; examination of a longitudinally slit section of rod 83.832 (assembly G-4) revealed an ID crack ~ 250 mils long. Examination of the cracks by both

metallographic and replication techniques is reported in WCAP-3850-2 and WCAP-3850-3.

An attempt was made to examine a fracture surface on rod 83.832 to investigate the possibility of a fatigue type of failure. A metallographic sample with a crack in a known position was broken apart as shown in Figure 35 (a) to give a fracture surface, as shown in Figure 35 (b), the original clad crack being located at the bottom of the sample. A plastic replica was taken of the fracture surface, shadowed with chromium and then with carbon. The lower region was removed, the plastic dissolved, and the carbon replica was examined in the electron microscope. The general structure away from the crack is illustrated in Figure 36 (a). Figure 36 (b) shows a structure which appears occasionally throughout the replica. This structure has been associated with groups or stringers of hydrides. What is taken to be the original crack surface is illustrated in Figure 37 (a) and (b). No evidence could be found of fatigue striations. Therefore, a fatigue mechanism can be discounted as a cause of crack propagation in this fuel rod.

6.5 BURST TESTS ON IRRADIATED TUBING

Six sections from four different irradiated CVTR tubes were tested to failure by internal pressurization. Prior to the hot cell tests, the burst test equipment and its operating procedures were developed by out-of-cell testing on unirradiated tubing. The burst test equipment includes: 1) an argon gas pressurization tank and pump; 2) data recorder; 3) a strain sensing transducer; and 4) a tube furnace with thermocouple attachments.

For each test, the diametral expansion of the tube is continuously plotted as a function of internal pressure. The diametral expansion is transmitted to a linear variable differential transformer (LVDT) by a push rod that maintains the LVDT out of the heating furnace. The motion of the LVDT plunger generates a signal which is recorded on the X-Y-Y' plotter simultaneously with a signal from a pressure cell on the pressurization equipment.

Before each test, pressure readout was calibrated against a Heise gage, the transducer was calibrated using pins of known diameter, and the X-Y-Y' recorder pen travel was calibrated against a portable potentiometer.

Figure 35. Fuel Rod 83.832: Fractography, Clad Crack at 26 1/2 Inches

X 3,200

(b)

X 2,500

(a)

Figure 36. Fuel Rod 83.832: Clad Fracture Surface

X 2,500

(a)

X 4,100

(b)

Figure 37. Fuel Rod 83.832: Clad Crack Surface, 26-1/2 Inches

Each of the six tests was run at a specimen temperature of 650°F with a pressurization rate of 500 psi per minute.

Difficulties with transducer operation prevented evaluation of failure strains from the instrumentation data. Metallographic techniques, therefore, were adopted. The burst specimens were metallographically evaluated at the positions of maximum strain. Ductility measurements were made by measuring the outer and inner circumferences of each section on enlarged photographs and glass plates and comparing the half-thickness circumference with the average pre-burst circumference as determined by micrometer measurements of the OD and reported nominal wall thicknesses.

The results are summarized in Table 27. The ductility of the burst test specimen taken from the failed rod is less than 1 percent circumferential strain, with the fracture occurring through a hydride rim on the outer diameter of the rod. The failure zone contains a large concentration of randomly oriented hydrides, in a quantity far in excess of that which one would calculate from the corrosion reaction on the cladding (Figure 38). The ductility of the thick-walled tubing sections from unfailed rods was greater than 3 percent with no evidence of hydrides at the fracture zone (Figure 39). The failure on these tubes is characterized by shearing (necking) and a 45° fracture tip. Sections of thin-walled tubing from unfailed rods exhibited 1.8 and 1.4 percent ductility. However, hydrides and a dense hydride outer rim were observed in the fracture zone of one sample and not in the other. In addition, the hydrided sample showed a brittle 90° fracture tip while the hydride-free sample showed a ductile 45° fracture tip (Figures 40 through 43).

In all tests, failure strains were greater than observed strains determined by profilometry on fuel rods. This observation is consistent with the determination that observed clad cracks did not penetrate the fuel rod clad and indicates that reduced clad ductility did not result in the observed failures.

TABLE 27

IRRADIATED CVTR CLADDING -- BURST TEST DATA

<u>Rod and Sample Location</u>	<u>Nominal Thickness (inches)</u>	<u>Condition^a</u>	<u>Failure Strain (Percent)</u>	<u>Nominal Failure Stress (psi)</u>	<u>Remarks</u>
1. 1475 - B1 (16-3/4 to 24-3/4 in.)	0.02575	F	0.9	64,700	Hydride rim on outer diameter with extensive random hydride distribution; 90° fracture tip.
2. 53.862 - B2 (34-5/8 to 42-5/8 in.)	0.024	U	4.8	73,800	No hydrides visible; necked-down fracture tip.
3. 1591 - B2 (35-1/2 to 43-1/2 in.)	0.02575	U	20.1	84,400	No hydrides visible; 45° fracture tip.
4. 1591 - B3 (43-1/2 to 51-1/2 in.)	0.02575	U	3.0	93,700	No hydrides visible; 45° fracture tip.
5. 83.832 - B2 (17-1/8 to 25-1/8 in.)	0.017	U	1.8	42,920	Outer diameter hydride rim with extensive random hydride distribution; 90° fracture tip.
6. 83.832 - B3 (56 to 64 in.)	0.017	U	1.4	Uncertain	Hydrides not visible at fracture zone; 45° fracture tip.

a. U = unfailed rod
F = failed rod

X 100

Figure 38. Burst Test Fracture, Failed Rod 1475

X 100

Figure 39. Burst Test Fracture, Unfailed Rod 53.862

X 100

Figure 40. Burst Test Fracture, Unfailed Rod 83.832 (56 - 64 Inches)

X 100

Figure 41. Burst Test Fracture, Rod 83.832 (17-1/8 - 25-1/8 Inches)

X 8.125

Figure 42. Cross-Section through Burst Tested Thin-Wall Rod
83.832 Showing 45° Fracture Tip and Ductile Failure
Mode

X 12.25

Figure 43. Cross-Section through Burst Tested Thin-Wall Rod
83.832 Showing 90° Fracture Tip and Brittle Failure
Mode

SECTION 7

FAILURE ANALYSIS

7.1 BURNUP AND POWER HISTORIES

7.1.1 Comparison of Predicted and Measured Burnup Data

Table A-3 of Appendix A lists predicted burnup levels, based upon fuel rod loading and exposure. Equivalent exposure hours, as tabulated in Appendix III of CVNA-287^[4], are listed for the Core I and Core II operation of each fuel assembly. Details of the calculations for average assembly burnup and for individual fuel rod burnups are given in Appendix A. The calculations utilize several hot channel factors, based upon critical experiments and flux wire data obtained during reactor operation. Table A-3 also lists the measured (Nd-148 analysis) burnups of the rods destructively examined.

Comparison of predicted* and measured average burnups for the special test assemblies shows reasonable (within ± 10 percent) agreement. The measured values are somewhat lower than predicted for assemblies G-1 and G-4, and slightly higher for assembly G-3. The differences are magnified in the peak burnup numbers, due to measured (gamma scan) axial hot channel factors, that in most cases exceed estimates based upon flux wire data.

The average (measured) standard rod burnups are compatible with the predicted burnup range for the standard assemblies if a ± 15 percent variation about the average rod burnup is assumed for the latter calculations. Previous burnup estimates^[1, 4] used a burnup distribution of ± 25 percent, which gamma scan results (cf. §6.2.2) have shown to be excessive. The substantial overestimate of predicted peak burnups in Table A-3 is largely a function of the excessively high axial factor used in previous reports and adopted here.

* CVNA-274^[1] and CVNA-287^[4]: Previously reported predictions covered only irradiation up to the time of failure, and not the total exposure.

In summary, the predicted average fuel rod burnups are generally consistent with measured values, within the uncertainties of the data. The discrepancies in peak burnup values result largely from local peaking factors, especially for the standard assemblies. The results verify the adequacy of nuclear design methods for the CVTR core.

7.1.2 Maximum Power Densities at Failure

The peak power density for each rod at the time of failure and the maximum achieved during its entire irradiation period are listed in Table 3, "Summary of Failed CVTR Fuel Rod Characteristics". With the exception of the special G-1 and G-4 assemblies, the rods failed at the time of their maximum power density. Power histories of the failed assemblies have been reported^[4] utilizing various combinations of thermal and hydraulic measurements and flux wire data, together with critical experiment results.

In brief, the power densities at failure were obtained from the following data:

1. Thermal and hydraulic measurements giving the radial power sharing of the pressure tubes containing a failed assembly. A direct measurement of the primary coolant flow and temperature rise in the appropriate U-tubes was available from plant instrumentation.
2. The power splits between the two U-tube legs of each pressure tube were obtained by TURBO* code calculation. Good agreement between previous measurements and TURBO* predicted radial factors lends confidence to predicted power splits.
3. Flux wire experiments at the time of failure or with a control rod configuration similar to conditions existing at failure have been used to obtain an axial peak-to-average factor (F_z) for each of the failed assemblies.
4. Critical experiments provide the various factors necessary to calculate power sharing within the assemblies.

Considerable confidence can be placed in the first three categories since they offer an essentially instantaneous measurement of assembly power behavior. The pre-irradiation critical experiment data, however, are subject to some uncertainty when applied to power calculations. The measured burnups and gamma scan data have been evaluated in an attempt to confirm the information in category (4).

The results indicate that previously calculated* linear power levels at failure for the special test assemblies do not need modification. The burnup distributions among the special rods are consistent with previous critical experiment results, as explained in detail in Appendix A. In the case of the standard fuel assemblies, however, previously reported kw/ft at the time of failure are probably excessive. As discussed in §7.1.1, the gamma scan and burnup data indicate a distribution of ± 15 percent about the assembly average fuel rod. The peak power levels of failed rods 1549 and 1475 in Table 3 have been modified to reflect this. A fuller discussion is given in the Appendix.

7.2 CVTR CLAD HYDRIDING

7.2.1 Identifiable Sources of Hydrogen

Table 28 lists known sources of hydrogen (mass one) picked up in the fuel clad. Two types of sources are listed:

1. Internal

Hydrogen found intrinsically, or as part of a contaminant, in fuel rod components, in the weld atmosphere, in the coolant (in the case of clad failure), and resulting from the fissioning process in the fuel.

2. External

This category includes hydrogen in the fuel clad resulting from normal clad-coolant corrosion and from pre-irradiation corrosion tests.

* CVNA-274^[1]: Table 10, page 58; or CVNA-287^[4], Table 7.1, page 173.

TABLE 28

KNOWN SOURCES OF HYDROGEN, CVTR FUEL ROD COMPONENTS

Identifiable Sources	Type of Source	Magnitude of Source: ppm	
		Standard Fuel Rods	Special Fuel Rods
Internal			
1. Fuel tubes	Intrinsic hydrogen	1-5 H	6-16H, 0.017 in. Zr-4 9-15H, 0.024 in. Zr-4
2. UO ₂ fuel	Moisture (H ₂ O), gaseous hydrogen	10, 12 H ₂ O; 1H	$\left. \begin{array}{l} \leq 5-10 \text{ H}_2\text{O} \\ 1, 3\text{H} \end{array} \right\} 93\% \text{ T.D., } 4.7 \text{ w/o}$ $\left. \begin{array}{l} \leq 1.8 \text{ H}_2\text{O} \\ (\text{N.D.}) \text{ H} \end{array} \right\} 96\% \text{ T.D., } 3.8 \text{ w/o}$ $\left. \begin{array}{l} \leq 1.6 \text{ H}_2\text{O} \\ (\text{N.D.}) \text{ H} \end{array} \right\} 93\%, 3.8 \text{ w/o}$
3. Al ₂ O ₃ spacers	Moisture (H ₂ O), organic contamination	N.D.	12.3-16.0 H ₂ O Trace CH ₄ , H ₂ from 750°F anneal
3. Fuel spring (chrome-plated)	Gaseous hydrogen	N.D.	4, 8H
5. 304 SS spacer sleeve (chrome-plated)	Gaseous hydrogen	N.A.	N.D.
6. Weld atmosphere	Moisture (H ₂ O), gaseous hydrogen, organics	$\leq 0.01\% \text{ H}_2$; $\leq 0.55\% \text{ CH}_4$	$0.21-0.27\% \text{ H}_2$; $\leq 0.5\% \text{ CH}_4$
7. Tritium	Ternary fission product (UO ₂)	≤ 0.013	≤ 0.009
External			
1. Internal coolant, failed fuel rods	Light water (H ₂ O) in coolant	$\leq 1.5 \text{ mol. } \% \text{ H}_2\text{O}$	$\leq 1.5 \text{ mol. } \% \text{ H}_2\text{O}$
2. External coolant - clad corrosion	Light water (H ₂ O) in coolant	$\leq 1.5 \text{ mol. } \% \text{ H}_2\text{O}$	$\leq 1.5 \text{ mol. } \% \text{ H}_2\text{O}$
3. Corrosion test (pre-irradiation)	Standard rods: 750°F steam, 3-4 days	$\leq 10 \text{ H}$	N.A.
	Special rods: 750°F oxygen, 3 days	N.A.	negligible

N.D. = Not determined directly.

N.A. = Not applicable.

The type and magnitude of the hydrogen sources are the result of the available pre-irradiation records and tests conducted upon archive materials during the current program*. For conservatism, the highest hydrogen content found among duplicate samples is assumed, except where explicitly noted otherwise. In cases where information is inadequate or lacking, certain assumptions and estimates have been made, as explained in the following section. Table 29 presents the calculated hydrogen content, in ppm, that would result in the Zircaloy clad from the various hydrogen sources, assuming uniform concentration and nominal fuel rod dimensions. Because of susceptibility to (internal) hydriding, a part of the end plug is arbitrarily included, adding a total (for two end plugs) of slightly less than 15 grams to the fuel tube material exposed to hydriding.

7.2.2 Description of Hydrogen Source Information (Table 28)

1. Internal Sources

a. Fuel tubes, intrinsic

This source refers to normal hydrogen content in the clad material, exclusive of corrosion tests, irradiation, etc. Archive standard and special test assembly fuel tubes were analyzed by the hot gaseous extraction technique.

b. UO₂ Fuel

Moisture data (Table 13) were available from analyses performed (by vacuum drying) upon receipt of the standard fuel pellets, in early January, 1962. In addition, two archive fuel pellets have been analyzed for both moisture content and for hydrogen gas (1000°C anneal and mass spectrometry). Results are given for both standard and special pellets in Table 14.

* Data on special assembly tubing, UO₂ pellets, Al₂O₃ spacers, weld atmosphere, and other rod components plus standard assembly corrosion tests, weld atmosphere, and fuel pellet chemistry tests, are given in Section 5.

TABLE 29

CALCULATED CLAD HYDRIDING FROM KNOWN SOURCES

<u>Source</u>	<u>No. 24.732</u>	<u>No. 53.861</u>	<u>No. 83.831</u>	<u>No. 1549</u>	<u>No. 1475</u>
Fuel tube	9-15 ^a	9-15 ^a	6-16 ^a	1-5	1-5
UO ₂	10.25-19.96	6.23-16.60	13.46-24.06	13.027	13.027
Al ₂ O ₃ spacers	0.020	0.020	0.014	0.025	0.025
Spring	0.230	0.230	--	0.218	0.218
Spacer sleeve	0.316	0.398	--	--	--
Weld atmosphere	0.119	0.094	0.135	0.034	0.034
Tritium	0.043	0.031	0.051	0.060	0.066
Corrosion test	--	--	--	10	10
Coolant (external)	0.031-0.050	0.037-0.044	0.046-0.260	0.052-0.160	0.052-0.160
<u>Subtotals</u>	20.0-35.7	16.1-32.4	19.7-40.5	24.42-28.52	24.42-28.53
Coolant (internal)	1.21 (end plug) 13.85 (31 in.) 11.32 (17 in.)	28.38 (end plug) 0.71 (25 in.)	3.95 (51 in.) 8.35 (14 in.) 40.88 (9 1/2 in.)	0.38 (74 in.) 0.79 (9 1/2 in.) 4.74 (end plug)	16.43 (100 in.) 19.38 (73 1/2 in.)
<u>Totals</u>	21.2-36.9 33.9-49.6 31.3-47.1	44.2-60.8 16.8-33.1	23.7-44.5 28.1-48.9 60.6-81.4	24.80-28.90 25.21-29.31 29.16-33.26	40.85-44.96 43.80-47.91

a. Includes any hydrogen picked up during corrosion test (pre-oxidation at 750°F).

Moisture data on the special assembly fuel pellets (Table 23) were obtained from pellet drying tests conducted prior to fuel rod loading; archive fuel pellets (93 percent T.D., 4.7 w/o U-235) were analyzed for both moisture and hydrogen content. Because the records indicate the use of unreleased pellets in some of the special assembly rods, these data may not be definitive.

Table 29 lists the clad hydrogen pickup from the UO₂ pellets for each failed rod. Two levels of fuel pellet hydrogen content (1 ppm and 3 ppm, as analyzed in the archive pellets) were assumed in all types of special fuel, thus accounting for the range of hydrogen pickup for each failed rod under this category.

c. Al₂O₃ Spacers

Al₂O₃ spacers used at the top of the fuel column of the special test assemblies were tested for moisture prior to loading (§5.2.5). In addition, archive spacers were analyzed (Table 20) currently for volatile gases by high-temperature annealing and mass spectrometry. The results indicate traces of organic contamination which may have resulted during storage. In any case, the quantity of gas released (as H₂ and CH₄) was insignificant.

d., e. Springs, Spacer Sleeves

Special assembly archive (Inconel) springs, which had been chromium-plated, were analyzed for hydrogen (§5.2.4). The same content is assumed for the standard assembly springs.

Records indicate that it was necessary to chrome-plate the stainless steel spacer sleeves used in the standard-wall special rods in order to produce a sufficient wall thickness. No spare sleeves were available for testing, so the same (ppm) hydrogen content as found in the archive springs is assumed. Spacer sleeves were not used in the standard assembly rods.

f. Weld Atmosphere

Data on possible hydrogen contamination of weld (internal rod) atmosphere has been taken from results on an unirradiated 1.5 w/o standard assembly fuel rod (Table 16) and from archive special assembly dummy rods (Table 19). The difference between calculated clad pickups for standard and special assembly fuel rods (Table 29) is mostly a factor of the considerably larger fission gas plenums in the latter.

g. Tritium Production in the Fuel

Assuming a production rate of one atom tritium per 12,500 fissions, the buildup within the fuel was calculated for the optimal (highest burnup) cases for standard and special assemblies, i.e.: No. 1591 and No. 83.832. It was assumed conservatively that all the tritium had diffused out of the fuel to be picked up by the clad. The result was less than one tenth ppm in each case. In the mass spectrographic analysis of gases extracted from clad samples, the mass three quantities were assumed to represent HD molecules resulting from equilibration. Because of the relatively small amounts of tritium produced, this assumption is justified for the purpose of hot extraction analysis calculations.

2. External Sources

a. Coolant: Failed Fuel Rods

The major source of deuterium found in failed rods is heavy water (D_2O) coolant. Breach of the clad during failure allows coolant ingress, with subsequent radiolytic decomposition by fission fragments, thereby liberating deuterium gas to react with the clad. Because of a hydrogen overpressure, progressive degradation (i.e., formation of light water) of the CVTR coolant occurred during the greater part of its operation. Although a deuterium overpressure was instituted at the beginning of the second fuel cycle (October 1966), approximately 1.5 mole percent^[9] of the coolant was H_2O during the period encompassing

the fuel rod failures. For failed rods, therefore, a proportionate amount of hydrogen would accompany internal deuteriding of the clad. Depending on axial position, metallographic studies and H/D analyses have shown wide variations in internal clad-corrosion.

b. Corrosion Tests

The standard assembly rods were subjected (OD only) to a nominal three day, 750°F, 1500 psi steam test after pickling (0.003 in., OD). In a number of rods (including No. 1475 and No. 1521) superficial defects on the clad surface were noted; the recommended procedure in such cases was to repair the defects and repeat the corrosion test for one day. Corrosion tests (Table 7) of archive tubing indicate a hydrogen pickup of ~10 ppm as an estimate for pre-irradiation condition after a three-to-four day test. This is based upon corrosion measurements upon tubing samples pickled on both OD and ID.

The special test assembly rods were furnace pre-oxidized at 750°F for three days, resulting in negligible hydrogen pickup.

c. Clad-Coolant Corrosion

Normal (external) corrosion during irradiation is assumed to have produced hydrogen (from the 1.5 mol. % light water) in the same proportion as deuterium to react with the clad. Accordingly, the amounts of hydrogen accumulated this way can be calculated from the deuteriding measured on unfailed rods.

For the standard unfailed rods, the range of maximum deuteriding observed was ~8 - 24 ppm. An exception was rod 1398 which had regions of relatively high deuteriding (up to ~82 ppm). For the special unfailed rods, the maximum deuteriding was ~5 - 7 ppm with the exception of thin-wall rod 83.832 (up to ~38 ppm).

7.2.3 Comparison of Clad Hydrogen Levels: Failed Rods versus Known Sources

Table 29 lists the calculated contribution of hydrogen to each of the failed rods from the various sources listed in Table 28. As can be seen, the two major factors are intrinsic clad and UO_2 -produced hydrogen. The variation in estimated UO_2 contribution reflects chiefly the two different (1, 3 ppm) hydrogen gas levels assumed for the fuel. The rest of the internal sources (excluding coolant) are essentially negligible, contributing a total of less than 1 ppm to the clad in all cases. The subtotals exclude effects of coolant ingress and reaction with the clad. The final totals include hydrogen pickup resulting from coolant ingress; this effect (shown separately), is assumed to be proportional to the amount of deuterium that reacted with the clad. Calculated internal hydriding is given at several discrete locations on each rod where deuterium content was actually measured. Thus hydrogen pickup for rod 24.732 is calculated at the upper end plug, and at 31 and 17 inches from the bottom, respectively.

In Table 30 each failed fuel rod is listed together with measured (H/D analysis) hydrogen and deuterium content at discrete locations which showed exceptionally high hydrogen levels. These data are included in Figures C-1 through C-5 and in general represent defected or blistered regions of the failed rods. For comparison, the calculated total hydrogen in the clad from known sources (Table 29) is tabulated for each axial location chosen. The next column is also taken from Table 29 and shows the contribution of all sources except internal coolant-clad hydriding. The distribution has been assumed to be uniform. The final column shows observed (H/D analysis) clad maximum hydrogen levels in unfailed sister fuel rods.

Special Assembly Failed Rods

The measured hydrogen content in the failed special rods is significantly higher than can be accounted for from known sources. The discrepancy between the two is well outside the error limits of the analysis, estimated as between five and six percent. This is in spite of some conservative assumptions in the calculated figures, as discussed previously.

TABLE 30

COMPARISON OF OBSERVED AND CALCULATED CLAD HYDRIDING

Assembly No.	Rod No.	H ₂ /D ₂ Analysis Sample Locations (in. from Bottom)	Observed ppm (max.)		Sample Description	Calculated Hydrogen (Total)	Calculated H ₂ Minus Internal Coolant Source	Observed ppm H ₂ (max.) in Sister Fuel Rods
			H ₂	D ₂				
Standard 502-2-31	1549	74-1/2 - 74-3/4	62.8	56.4	Routine	24.8-28.9	24.4-28.5 (uniform)	1257: 36.8-46.0 (82-1/8 - 82-1/2 in.)
	1549	9-1/4 - 9-1/2	40.6	116.8	Routine	25.2-29.3		36.8-49.7 (25-3/4 - 26 in.)
		Lower end plug	59.2	697.7	Opposite pinhole defect	29.2-33.3		1398: 25.7-38.8 (82-1/8 - 82-1/2 in.) 27.3-39.7 (25-5/8 - 26 in.)
Standard 502-1-18	1475	100 - 100-1/4	112.0	2417.0	Clad defect at upper end plug	40.9-45.0 43.8-47.9	24.4-28.5 (uniform)	1521: 26.1-46.8 (25-5/8 - 26 in.)
		73-1/2 - 73-7/8	241.9	2852.3	Clad blister			1591: 13.5 (upper end plug) 24.3-26.2 (25-5/8-26 in.)
Special G-1	24.732	Upper end plug	70.0	178.1	Local hydride/deuteride, end plug - clad crevice	21.2-36.9	20.0-35.7 (uniform)	44.731.
		31 - 31-1/4	101.0	2035.0	Blister defect, secondary clad fracture	33.9-49.6		20.7-22.1 (50-1/4 - 50 1 2 in)
		16-3/4 - 17	141.0	1666.0	Circumferential ridge, secondary fracture	31.3-47.1		20.6-27.2 (25-3/4 - 26 in.)
Special G-3	53.861	Upper end plug	157.0	4176.0	Defect region	44.2-60.8	16.1-32.4 (uniform)	53.862:
		25-1/4 - 25-1/2	104.3	104.7	Routine	16.8-33.1		11.9 (upper end plug) 18.4-28.2 (50-5/8 - 51 in.) 13.7-35.7 (25-5/8 - 26 in.)
Special G-4	83.831	50-5/8 - 51	70.4	581.0	Internal H ₂ /D ₂	23.7-44.5	19.7-40.5 (uniform)	83.832:
		14 - 14-1/4	92.3	1242.0	Blister defect	28.1-48.9		22.6 (50-5/8 - 51 in.)
		9-5/16-9-11/16	209.0	6016.0	Clad blister	60.6-81.4		32.3 (25-5/8 - 26 in.) 110.6 (21 in.)

In addition, the calculated hydrogen content subtotals (excluding the effect of internal clad-coolant reaction) agree reasonably well with maximum measured values from the sister rods. Because these rods did not fail, and (with one exception) appeared normal under metallographic examination, they provide a good bench-mark for evaluating the failed rods. The one notable exception is fuel rod No. 83.832 from special assembly G-4, where one of the H₂ samples (~21 inches) far exceeded the calculated maximum with 110.6 ppm. This sample was taken from a burst test section (17-5/8 - 25-5/8 inches from the bottom) which had subsequently shown metallographic evidence of hydriding. In addition, small regions of internal hydriding had been discovered at higher axial locations (cf. §3.3) indicating that rod No. 83.832 was abnormal although it did not fail.

Standard Assembly Failed Rods

Comparison of observed clad hydrogen levels at discrete locations on the failed rods 1549 and 1475 with calculated levels from known sources indicates a significant difference. However, only rod 1591 falls within the calculated range for the standard rods; the other sister fuel rods show comparatively higher measured amounts of hydrogen. Thus, failed rod 1549 does not, on the whole, show a conclusive hydrogen excess when compared with the sister rods (cf. Figure C-2). The explanation may be that rod 1591 hydrogen levels are normal and the other sister rods reflect internal sources which were underestimated or unaccounted for. It also should be noted that most of the defect areas on fuel rod 1549 were not analyzed for hydrogen. Of these, only the lower end plug (opposite the pinhole defect) region was sampled; the results here (59.2 versus 13.5 ppm hydrogen for 1591) do show evidence of internal hydriding.

There is no similar ambiguity in the results at (defect locations) for failed rod 1475. There is clear evidence of excessive hydriding even in comparison with the inexplicably high analytical results observed in three of the four sister rods.

7.2.4 Summary and Discussion

The comparison of measured clad hydrogen levels between failed and unfailed fuel rods has shown a discrepancy, particularly at the blister or defect locations of the failed rods. This has been documented in Appendix C, Figures C-1 through C-5. The only exception is for rod 1549, where the results are not conclusive. This was previously discussed in §6.1.

In this section, the known sources of clad hydriding have been tabulated and the individual contributions to clad hydriding have been calculated and summed. The calculated clad hydriding indicates that the known sources are not sufficient to account for the comparatively high hydrogen content of the failed fuel rod clad. However, there is a reasonably good agreement between calculated hydrogen levels and measurements upon the unfailed rods. These results are based on the premise that hydrogen from known sources is distributed uniformly in the fuel clad. This is an idealization, because some localization of internal hydriding must occur, as the investigations of this program have shown. A quantitative estimate of this effect cannot be made from the information available, but it is obvious that certain regions of the clad, such as fuel pellet-interfaces and end-plug-clad crevices, are particularly susceptible to hydriding.

The question arises whether or not localization of hydriding from known sources could account for the high levels measured in the failed rods. The evidence from unfailed (sister) rods suggests not. Extreme localization of clad hydriding does not occur in these rods, with approximately the same hydrogen levels as the calculated values. The axial distributions (Appendix C) of unfailed rod hydriding are relatively uniform. If a proportional degree of localization were applied to the calculated results, it still would not be sufficient to account for observations in the failed rods. One exception is fuel rod 83.832 but this is a special case, as discussed previously (§7.2.3). Thus, the assumption of uniform hydriding is reasonable, at least for the purpose of the calculations. Furthermore, one of the failed rods (24.732, special assembly G-1) shows a generally higher degree of hydriding than other sister rods; simple localization of the identified internal hydrogen sources is obviously not an adequate explanation here.

Other internal sources of hydrogen than those identified in Table 28 must have been present in the failed rods and unfailed rod 83.832 (special assembly G-4). The hydrogen available from such sources would have to be of sufficient magnitude to raise clad hydride levels generally, as in rod 24.732, or to raise the level of hydriding at potential localization sites to significant values, such as in rod 1475. Positive identification of further internal hydrogen sources has not been possible, but enough circumstantial evidence (§5.1.4, 5.2.5) is available to suggest the presence of contaminants.

7.3 FAILURE MECHANISMS

None of the failed special or standard rods has a common combination of fabrication variables or operating conditions, suggesting the possibility of multiple failure mechanisms operating in these rods. To assure effective experimental investigations, a list of possible failure mechanisms was developed and each of these mechanisms evaluated for applicability to the five failed rods. This was done in a semi-quantitative fashion by assigning a relative probability of the mechanisms being operative in each case. This probability cross-listing of mechanisms and rods then was reviewed to determine what critical tests could be performed to accept or reject the mechanism in the case of each failed rod.

The areas which received prime emphasis were: fuel-clad mechanical interaction, as-fabricated fuel rod integrity, and determination of whether the defect regions were composed of zirconium hydride or zirconium deuteride. The most likely mechanisms for each particular fuel rod failure are given below, along with a summary of the supporting evidence.

7.3.1 Standard Assemblies

Maximum clad stress-strain behavior of failed special test assemblies has been reported in CVNA-274, based upon calculated power and burnup histories. The calculations were made with an assumed combination of clad and fuel pellet tolerances that would produce a minimum initial cold gap. Such a calculation was performed for the power history applicable to core positions 8C/6K, corresponding to assembly 502-1-18 irradiation. The results showed a

clad tangential stress of ~23,000 psi and 0.20 percent total clad strain at the time of the rod banking experiment. The peak power used was the previously calculated 13.6 kw/ft, which has been shown (cf. §7.1.2) to be an excessive estimate for the failed rods. Clearly, the calculations show stress-strain levels less than expected to produce actual clad failure due to clad-fuel mechanical interaction. Observed average diameters at 26-1/2 inches on the failed rods do not indicate any instance of excessive fuel rod swelling. Although definitive pre-irradiation diametral measurements are not available, the post-irradiation data can be compared with the fabrication (minimum) tolerances to calculate an upper limit for possible (residual) strain levels:

<u>Rod No.</u>	<u>Axial Location</u>	<u>Average Diameter (26-1/2 in.)</u>	<u>Maximum Change</u>	<u>Maximum Strain</u>
1475	26-1/2	0.4870 in.	+0.0015 in.	≤0.31%
1549	26-1/2	0.4868 in.	+0.0013 in.	≤0.27%

These strains are far less than would be expected to produce clad failure. Nonetheless, there were large clad ID cracks observed at 26-1/2 inches from the bottom of both failed standard rods 1475 and 1549, and in unfailed rod 1521. This indicates that locally higher clad stresses must have existed to produce the cracks. Whatever the explanation for the cracks, there is no direct evidence relating them to the actual failures. No through cracks were found in the failed rods in the vicinity of 26-1/2 inches, either by metallography, or leak testing. The observed failures were well removed from this point on the rod. For this reason, together with the absence of excessive clad strains, it has been concluded that clad-fuel mechanical interaction by itself did not produce the standard fuel rod failures.

The most likely failure mechanism is a combination of local clad (internal) hydriding and increased clad stresses during or shortly following the rod banking experiment. Hydrogen/deuterium analyses on samples from rod 1475 (502-1-18) show definite indications of internal hydriding at the upper end plug and 73.6 inch clad defects, as discussed in §6.1 and 7.2.3. It is not

known which of these defects was the initial one, although fuel rod integrity would be more vulnerable at the end plug region. Experience in the PRTR (Appendix E) has shown that the end plug-clad crevice is particularly susceptible to gas phase hydriding resulting from an internal (hydrogen producing) contaminant. The differential volume change at this region resulting from the formation of zirconium hydride is likely to produce considerable local strains. Metallography (§ 3.5.1) of the upper end plug clad fracture surface confirmed that considerable outward bowing occurred.

The likely failure sequence of rod 1475, therefore, is that the locally hydrided and embrittled clad at the upper end plug was initially defected by local stresses; subsequent coolant ingress enlarged the initial defect and produced secondary blister defects or enlarged existing blisters (perhaps the 73.6 inch blister). The role of the fuel rod power changes in producing local stresses to defect the rod is not clear in the case of rod 1475. The downward shift of the axial power distribution during the rod banking would have maximum effect (from fuel-clad interaction) in the lower part of the rod. The effects on the upper end plug region involve a complex interaction of several factors, including contact with adjacent fuel rods, with the upper grid assembly, and with the wire spacer on the rod itself. No model is available to calculate the quantitative changes in local stresses at the upper end plug clad region, but it is probable that they were enough to defect the embrittled rod. Another possibility* is failure during the period of reactor power cycling between November 21 and 30, 1966, when the defect in special assembly G-1 was being sought. A conventional control rod configuration was in effect during this time.

* It should be pointed out that the identified failure date of November 21, 1966, (rod banking experiment) for the standard assemblies is only an assumption. Actual confirmation of failure was December 29, 1966, for pressure tube 6J (containing 502-1-18) and January 13, 1967 - January 16, 1967, for pressure tube 7J (containing 502-2-31). After startup of the reactor on December 9, 1966, following removal of failed special assembly G-1, it was observed that the coolant activity did not recede to a "normal" level. A program of core scanning was then begun to locate further defected fuel assemblies. Thus, the actual failure time of the standard assemblies cannot be exactly determined.

Fabrication records on rod 1475 indicate rewelding at the upper end plug was necessitated by a tungsten electrode short during the first weld. However, the reweld did pass the subsequent x-ray examination. Metallography of the defect area did not reveal any inclusions in the weld; one may conclude that the upper end plug weld was sound.

The extremely heavy crud deposition probably did not have a significant effect on the standard rod failures. There is no correlation between the existence and location of clad blisters and crud distribution for the failed rods, as shown in Appendix B. Away from the localized defect areas, the relatively low-level clad hydride/deuteride distribution in rods 1549 and 1475 was not greatly affected by the crud.

The failure mechanism of rod 1549 probably involved local internal hydriding combined with increased clad stresses in the bottom of the rod during the rod banking experiment. In this case, there is suggestive, but not conclusive, evidence of internal hydriding (cf. § 7.2.3). The shifting downward of the axial power distribution on the fuel rod undoubtedly was crucial here, for the defects occurred in the lower 10 inches of rod 1549. Furthermore, the increase in clad-fuel interaction above normal operation was comparatively higher in the bottom ~25 inches of the fuel than would have occurred from a conventional power increase with the axial peak at ~45 - 50 inches. An estimate of the power increase at 25 inches fuel height can be obtained from a series of flux wire measurements made during Core II CVTR operation, reported in Table 9, CVNA-274, page 57. Power levels at the normal axial peak (~45 - 50 inches) and at 25 inches are listed. The linear power level at 25 inches during the rod banking operation was ~30 percent above typical levels achieved during the previous fuel cycle, May - August, 1966.

The existence of local clad hydriding at either 7-5/8 (fuel pellet interface) or 9-3/4 inches from the bottom probably produced a defect when combined with increased fuel-clad interaction during the rod banking experiment. Subsequent power operation enlarged the initial defect and produced secondary effects at the lower end plug (clad blistering, pinhole defect) and the upper end plug (local deuteriding).

Testing of archive standard fuel rod components (fuel tubes, UO₂ pellets, springs, spacers, and end plugs) did not reveal an obvious (internal) source of hydrogen, although the long storage time under questionable conditions may be a factor. One possibility might be the use of inadequately dried fuel pellets in the failed rods. Examination of the fabrication records (cf. § 5.1.4) indicates that some pellets loaded into these particular rods could have inadvertently been omitted in the oven-drying process. Excessive moisture in the fuel could produce problems if concentrated locally at a few points along the fuel clad. Metallographic and hydrogen/deuterium analyses have shown that extreme localization has occurred for both hydrogen and deuterium in the failed fuel rods. This is particularly true at pellet interfaces (opposite a defect) and at the clad-end plug crevices. Localization of hydriding within the fuel rods could be further enhanced by residual fluoride contamination (from pickling) on the clad bore; tests of archive tubing showed indications of such contamination. Examination of the failed rods showed extremely rough bores (also observed in some archive tubes) which would increase the probability of fluoride retention.

7.3.2 Special Test Assemblies

Stress-strain histories were calculated for special assemblies G-3 (core position 3B) and G-1 (8R) on the basis of minimum fuel-clad diametral gaps and estimated burnup and power histories^[1]. The results are considerably below stress-strain levels from clad-fuel interaction likely to produce clad failure. Maximum possible residual strains, based upon fabrication tolerance extremes, are tabulated below for the failed special rods:

<u>Assembly</u>	<u>Rod No.</u>	<u>Axial Location</u>	<u>Average Diameter</u>	<u>Maximum Change</u>	<u>Maximum Strain</u>
G-1	24.732	26-1/2 in.	0.4880 in.	+0.0020 in.	0.41%
G-3	53.861	48-1/2 in.	0.4872 in.	+0.0012 in.	0.25%
G-4	83.831	26-1/4 in.	0.4887 in.	+0.0027 in.	0.55%

Measured (by profilometer) diameters are given at the approximate axial position of the peak heat flux at failure. Because of these results, it is

concluded that mechanical strains alone did not cause the fuel rod failures. Clad ID cracking was observed in the two 0.017-inch clad rods 83.831 and 83.832 in G-4, but this was not the cause of failure, as explained below.

Fuel rod 24.732 is postulated to have failed as the result of internal hydriding and local fuel-clad stresses produced during the rod banking experiment on November 21, 1966. The fuel rod had a series of circumferential ridges and blisters in the lower 43 inches. Hydrogen/deuterium analyses have definitely established the ~17 inch circumferential ridge and the ~31.1 inch blister (defected) as sites of excessive hydriding (mass one). An unidentified internal source apparently produced local clad hydriding at most or all the anomalous regions in the lower part of the rod. Coolant ingress following failure undoubtedly contributed to deterioration of the rod; localization of hydrogen/deuterium was particularly prominent at fuel pellet interfaces (four out of five anomalous areas).

The increase in clad stresses in the lower part of the rod produced the failure, either at the ~12.8 inch fracture zone, or at the ~31.1 inch blister. Rod 24.732 operated successfully at 21.8 kw/ft at the time of the G-3 failure, yet failed at 21.2 kw/ft during the rod banking experiment. The most obvious explanation for this would be that the axial positions of localized clad hydriding were concentrated below 43 inches on the rod. Fuel clad stress increases from higher power levels, thus, were less critical under a normal axial power distribution (peaking at 45 - 50 inches) than during the rod banking experiment (peaking at ~26-1/2 inches). Using the detailed power history curves for G-1 in CVNA-279^[10], and assuming a ~10 percent lower power (from gamma scan) at 26-1/2 inches than at the "normal" peak, the 21.2 kw/ft experienced on November 21, 1966, represents a power increase of ~20 percent over peak cycle 1 operation, and ~10 percent above brief prior peaks in cycle 2. The proportion of the power increase is lower than for the standard assemblies because of the comparatively low changes in the axial peaking factor F_z on the core periphery.

Fuel rod 53.861 is postulated to have failed as a result of local hydriding and clad stresses at the upper end plug. This rod was among the group of special rods that had late upper vent hole welds; hot extraction analysis of

the upper end plug and adjacent clad of 53.861 showed anomalously high hydrogen (mass one), evidence of internal contamination. In addition, inspection records revealed that the upper end plug of 53.861 was misaligned and apparently was inserted into the assembly without a correction of this condition. Unlike the standard fuel rods, which rested on the lower grid, the special assembly rods were suspended from the upper grid assembly, which had been modified to allow individual fuel rod removal. Thus, unusually high local stresses at the upper end plug region undoubtedly resulted from the misalignment. During the May - August, 1966, operation the end plug-clad region became hydrided, which then led to failure on startup for the next fuel cycle on October 5, 1966.

Fuel rod 83.831 of G-4 failed at steady state power on January 24, 1967, at a considerably lower peak (16.0 kw/ft) than experienced earlier during the rod banking experiment* (17.6 kw/ft). Both 83.831 and sister rod 83.832 showed ID cracking of varying severity in metallographic specimens at 26-1/2 and 51-1/2 inches. None of the cracks penetrated the clad, however, as shown by both helium and liquid-nitrogen leak tests. Significantly, these two rods also had late vent hole closures. Post-irradiation examination produced both metallographic and analytic evidence of internal hydriding. The failure mechanism of 83.831 is postulated as follows:

Internal contamination during fabrication produced local clad hydriding and probable minor blistering. It is not understood why the fuel rod did not then fail during the rod banking experiment when clad stresses in the lower part of the rod (where the failure ultimately occurred) were at a maximum. A small clad blister 14 inches from the bottom finally cracked under constant power conditions two months later. The circumstances of failure imply that either normal mechanical vibration or slowly increasing clad strains from irradiation fuel swelling produced the defect of the blister. Alternatively, gradual deterioration

* Monitoring of pressure tube 3R (containing G-4) coolant activity on December 16, 1966, showed no evidence of defective fuel rods^[1]. This confirmed that G-4 did not fail during the rod banking experiment on November 11, 1966.

such as observed on blisters of standard rods, may have been responsible. The other clad blisters between 9 and 18 inches from the bottom of 83.831 are probably secondary, i.e., a consequence of the failure.

7.4 FACTORS NOT RESPONSIBLE FOR FAILURE

It must be concluded that the observed fuel rod failures are not uniquely related to specific CVTR design factors, such as D₂O coolant, neutral pH, individual pressurized U-tubes, and fuel assembly design. One supporting observation is the low statistical incidence of failure among the standard fuel rods: only two failed out of the initial core of 1368 rods plus 418 replacement rods. And despite the higher failure incidence of the special assemblies (3 out of 24 rods failed), no consistent pattern exists in the fuel rod failures from the standpoint of fuel rod design, materials, fabrication methods, or power history.

Specific observations on the fuel rods examined support the above conclusions. Visual examinations on the standard rods have confirmed the existence of heavy crud, characteristic of the neutral pH coolant chemistry of the CVTR. Deposition on the standard fuel rods (Appendix B) has been shown to be localized, rather than general. Despite localization and extreme thickness of the deposits, however, this did not result in adverse thermal and hydraulic conditions or lead to excessive clad corrosion. There was no visual or metallographic evidence of inadequate coolant flow or DNB conditions; in general, measured (external) oxide films and deuterium levels for the standard rods were within predicted limits. Exceptions (unfailed rods 1398 and 1521 for oxide, rod 1398 for deuterium pickup) may be related to higher clad temperatures resulting from crud localization. The special assembly fuel rods showed very light and uniform crud deposition. External clad corrosion and deuteriding appeared normal, conforming to predictions based upon experience in light water reactors. Analysis of crud deposits (standard rods) gives approximately the same composition as samples filtered from the coolant. The low density of the deposits may account for the minimal effects on clad corrosion.

The fuel assembly design was validated by both standard and special assemblies. There were no deleterious effects from the mechanical stresses imposed by the thermal baffles, end grid plates, wire spacers, etc. In particular, there was no evidence of excessive clad fretting or corrosion caused by the fuel rod spacer wires. Examination of the clad beneath the spacers revealed occasional regions of surface roughening, but nothing more serious.

There is no correlation between the observed failures and individual fuel rod design (cf. § 2.2), nor is there evidence of a burnup-kw/ft limitation in fuel rod performance. Several fuel rods had internal cracks at the axial location of the peak heat flux during the rod banking experiment, but no through-defects of the clad were found. It is concluded that the ID cracking is not related to the actual failures, which were produced by other factors. Post-irradiation diametral measurements and fission gas collection show strains and gas releases, respectively, that are well within tolerable limits.

Burnup analyses and metallographic examination of the fuel conform well to predictions; the measured burnups and calculated power histories of the special assemblies are believed to be very accurate, whereas, the standard fuel operated slightly below the somewhat conservative estimates. Thus, the fuel rod mechanical and nuclear designs have been substantiated by the examination results.

An intensive examination of fabrication records and a testing program upon archive fuel rod materials were performed. For the standard fuel rods, tests indicate acceptable fuel and clad quality: tests of archive fuel pellets confirm the density, moisture, and impurity levels found in pre-irradiation tests. The external corrosion rate of (pickled) autoclaved fuel tubes was satisfactory; hydride orientation confirmed expectations. Several samples of archive tubing exhibited indications of bore fluoride contamination; however, it is not known whether or not these results are representative of fabricated fuel rods. By itself, ID fluoride contamination would not be expected to cause clad failure. In the absence of an internal hydrogen contaminant, fluorides on the bore would not be significant. With an internal hydrogen

source, hydriding of the clad could be accelerated, particularly if localized residual fluorides were present. There is no evidence concerning the uniformity of the fluoride contamination, however. Several tubing samples exhibited extremely rough bores, similar to metallographic observations on irradiated rods. There is no apparent correlation between bore roughness and failure among the standard rods. Tensile and burst tests on standard archive tubing indicate acceptable strength and ductility. Hoop ductility variations with bore condition (lower for rough bores) at low temperatures were not observed at elevated temperatures. Post-irradiation burst tests of unfailed rods also showed high ductility (§ 6.5). Analysis of fuel rod components (fuel tubes, Al_2O_3 spacers, springs, etc.) revealed no significant sources of hydrogen that might lead to fuel rod failure.

Examination of standard fuel rod fabrication records indicated a high incidence of rejectable surface defects after corrosion testing. However, the records indicate the defects were superficial and presumably were adequately repaired. Similarly, a high rate of weld repairs (for porosity and undercutting) is probably not significant in terms of fuel rod integrity. Closed-end burst tests on archive end plug welds that exhibited porosity showed that these welds were not inferior in strength or ductility. Metallography on failed rods 1475 and 1549 end plugs revealed only minor porosity. Similarly, no definite evidence of weld contamination was found on the upper end plug of rod 1475, which (according to records) had a tungsten electrode short during re-welding operations.

Examination and testing of archive special assembly fuel materials indicate satisfactory quality. Ultrasonic inspection showed the tubing (both standard and thin-wall types) surfaces to be satisfactory. The corrosion performance of the tubing was good, with low weight gains and excellent oxide appearance. There were indications of excessive bore fluoride retention, however, both from the corrosion tests and analytical checks. Tensile tests resulted in good yield strengths for both types of clad at room temperature and acceptable ductility at both room and elevated temperatures. The tensile strength of the standard-wall (0.024 inch) tubing was low at 725°F, a characteristic of the tube fabrication process. However, the tubing still should have been

more than adequate for the operating characteristics of the special assemblies. Post-irradiation burst tests confirmed the adequacy of clad ductility. Neither standard or thin-wall clad approached the failure strains observed in these tests.

Tests of archive fuel rod components for hydrogen revealed traces of organic contamination in the dummy rod weld atmosphere and in Al_2O_3 spacers, but the magnitude was not significant. The UO_2 pellets that could explicitly be accounted for (archive pellets, recorded pre-irradiation examinations) were normal with respect to diameter, density, and moisture.

The fabrication records indicate numerous manufacturing deviations for the special assemblies, the most common of which were end plug porosity, minor oxide film defects, and dimensional anomalies. No correlation has been established between the special rod failures and these particular deviations.

SECTION 8

SUMMARY AND CONCLUSIONS

8.1 STANDARD FUEL ASSEMBLY FAILURES

Failures occurred in standard fuel rods 1475 (12.5 kw/ft, 15,700 MWD/MTU peak) and 1549 (11.3 kw/ft, 14,900 MWD/MTU peak) from CVTR assemblies 502-1-18 and 502-2-31, respectively. Both rods are believed to have failed during a rod banking experiment on November 11, 1966, and they continued operation for an additional 1200 - 1500 MWD/MTU peak burnup increment. Fuel rod 1475 was characterized by a clad failure at the upper end plug and a series of clad blisters in the upper 40 inches. Fuel rod 1549 had two large clad blister-defects at 7-5/8 and 9-3/4 inches from the bottom, respectively, and a clad pinhole defect at the lower end plug. Both failed rods had severe clad hydriding in the immediate vicinity of the defects. No other standard fuel rods in the CVTR core failed (a total of 1368 rods were initially irradiated, plus 418 replacement rods).

Clad hydrogen/deuterium analyses showed definite local hydriding (mass one) at defect locations of fuel rod 1475, and indicated a slightly excessive hydrogen concentration at the pinhole-defect region of rod 1549. The results on 1549 are not conclusive, because the other defect areas were not sampled. The postulated failure mechanism for the two standard rods is local clad (internal) hydriding combined with increased clad stresses at the time of the rod banking experiment. These factors produced a small initial defect in the embrittled clad which became enlarged and subsequently produced multiple secondary defects resulting from coolant ingress. Because testing of archive fuel rod components did not reveal an excessive internal source of hydriding, the most likely explanation is the use of inadequately dried fuel pellets. Fabrication records suggest this possibility. The observed localization of clad hydriding within the failed rods may have been influenced by residual fluoride contamination of the bore.

8.2 SPECIAL FUEL ASSEMBLY FAILURES

Fuel rod 24.732 (21.2 kw/ft, 9400 MWD/MTU peak) from special assembly G-1 failed during the rod banking experiment. This rod was fractured ~13 inches from the bottom, and contained a series of clad blisters and circumferential ridges in the lower 43 inches of the rod. Fuel rod 53.861 (17.2 kw/ft, 7200 MWD/MTU peak) from G-3 failed on approach to power, October 5, 1966. This rod had an upper end plug clad failure. Fuel rod 83.831 (16.0 kw/ft, 8600 MWD/MTU peak) was the last special assembly rod to fail, on January 24, 1967. The failure occurred at a cracked clad blister ~14 inches from the bottom, one of several blisters between 9 - 18 inches.

All the failed special assembly fuel rods have shown anomalously high clad hydrogen concentrations at defect locations. The failures are concluded to have resulted from the combination of internal hydriding and local clad stresses at the time of failure. The source of the internal hydriding observed in the failed special assembly rods is probably a function of fabrication procedures, although incomplete records prevent an exact identification of the problem. The most likely source is moisture or organic contamination. A series of six fuel rods, including failed rods No. 53.861 and No. 83.831, were not sealed immediately following welding of their upper end plugs. Handling of these rods prior to the final vent hole closure welds is believed to have resulted in hydrogenous contamination. Four of the rods exhibited internal hydriding or excessive bore oxidation; two of the rods failed.

8.3 GENERAL CONCLUSIONS

1. Internal hydriding has definitely been established in four out of five failures. Moisture contamination incurred in the fabrication of the failed rods is the most likely cause of hydriding.
2. The failure mechanism consists of a combination of local hydriding and mechanical stress at the point of hydriding. Neither of the two alone would be expected to produce a failure, under the conditions existing in the CVTR.

3. Evaluation of the fuel rod design, materials, specified fabrication methods and power levels demonstrated that these factors were not responsible for the failures. The lack of a consistent relationship between the defect type and location and the design, fabrication and operating variables, further supports this conclusion.
4. The failures are not uniquely related to specific CVTR design factors, such as D₂O coolant, neutral pH, U-tube or fuel assembly design. Despite heavy deposits of crud, characteristic of the neutral pH system, clad corrosion and deuterium pickup were generally within predicted levels. No evidence of deleterious thermal and hydraulic effects could be found. Clad fretting from fuel rod spacer wires proved to be insignificant, and fuel rod design parameters (diametral gap, plenum length, fuel density, etc.) have been shown to be unrelated to the failures. The metallographic results and burnup analyses confirm the predicted power histories of the fuel assemblies.

8.4 RECOMMENDATIONS

One of the objectives of this program is to identify the necessary revisions to the fuel design, fuel and cladding specifications, fabrication procedures, or fuel operating conditions, which, if utilized, would have prevented the fuel cladding defects. A review of the design parameters and the operating conditions has led to the conclusion that these were not causal factors in the observed failures. Therefore, no recommendations are appropriate in these particular areas.

In addition, a review was made of fuel and cladding specifications, fabrication procedures, and quality assurance procedures. Specific areas of fuel specifications reviewed were: moisture and volatile impurity content, enrichment, O/U ratio, density, and dimensional variations. Tests of both special and standard archive fuel pellets, in addition to the available inspection records, indicate that the specifications were met. Post-irradiation examinations and analysis indicate that the fuel specifications were satisfactory for the fuel design and operating conditions.

Because the standard fuel rod failures involved local internal hydriding of the clad, a specific recommendation concerning fuel pellet moisture was considered. Although the low incidence of rod failure (two in 1786) suggests that the reported 10 ppm fuel moisture content is satisfactory, the lack of a quantitative measurement of the contaminant level responsible for the two standard fuel rod failures and the large difference in actual (~ 10 ppm) and specification limit (75 ppm) moisture levels does not allow a judgement on the acceptability of the moisture specification in force at the time of standard rod fabrication. The special rods also exhibited internal hydriding, but more positive evidence of contamination of the fuel exists, which is not related to the moisture specifications.

Specifications on clad tubing were reviewed, combined with a study of archive samples. The areas investigated were tube dimensional tolerances, bore roughness (ultrasonic tests) and fluoride retention, corrosion performance and hydride orientation, and mechanical properties (tensile and burst tests). The specified mechanical properties of the tubing were satisfactory for the fuel operating conditions. Relatively low room temperature hoop ductility values (5 - 6 percent) were measured on archive standard tube samples having a rough bore but no variation in the hoop ductility values with varying bore roughness was observed in tests performed at 600°F. The measured room temperature failure strain of ~ 6 percent on the rough bore tubing and the minimum ductility of irradiated clad samples, 1 percent, clearly exceed the 0.3 percent maximum possible hoop strain determined for the failed rods. The special assembly tubing showed even greater margin between the maximum possible clad strain in the failed rods and the measured hoop ductility of irradiated samples. Accordingly, no changes in the mechanical property specifications are indicated.

Because of a possible contribution to localization of internal hydriding, bore fluoride contamination ($0.1 - 1.1 \mu\text{g}/\text{cm}^2$) in archive standard tubing merits further discussion. It should be noted that there was no specification in effect for bore fluorides at the time the standard assembly fuel tubing was purchased. Localized fluoride contamination is of concern, but the surface averaged values obtained by the methods used in checking the

standard tubing archive samples do not help to establish a quantitative localized contamination limit which would be appropriate for procurement specification use. Instead, a qualitative test is recommended, requiring a uniform black oxide film on the bore following a three-day, 750°F steam test on as-received tubing. Such a requirement was already included in the special assembly Zircaloy clad specifications.

The specifications for other fuel rod components such as end plugs, springs, Zircaloy rod stock, and spacer pellets were reviewed, and all were judged satisfactory for the CVTR fuel service requirements.

The manufacturing records and procedures were reviewed for the failed assemblies for all stages of fabrication, including: receipt, inspection, and cleaning of fuel rod components; manufacture, drying, and loading of fuel pellets; welding, inspecting, and vent hole closure of end plugs; corrosion testing and repairing of fuel rod OD surfaces; and wire wrapping and assembling of completed fuel rods. For the standard fuel rods, the available evidence suggests that the failures are related to a fabrication deviation (inadequate drying of a small number of fuel pellets). Since the process as specified produced pellets of satisfactory dryness, no change in procedures is necessary to prevent future, similar failures. Therefore, the only recommendation is a continued adherence to established manufacturing procedures.

Because of a significantly higher incidence of failure and more positive evidence of manufacturing deviations for the special assembly fuel rods, the following recommendations are offered to prevent similar failures in the future:

1. The fabrication procedures for experimental fuel rods should incorporate a routine similar to that used in production fuel assemblies. Specifically, individual route cards which include quality control check points should be required. Adherence to fabrication and material specifications should be strictly maintained with careful investigation of unavoidable deviations.
2. Particular importance should be placed on preventing contamination of the fuel at all stages of experimental fuel rod fabrication.

Systematic checks of fuel moisture and internal hydrogen content should be made and recorded prior to loading of fuel rods. Final sealing of the rods should be performed as soon as possible after fuel loading, to prevent internal contamination.

The quality assurance aspects of both standard and special test assembly fuel fabrication were reviewed to determine how this aspect of the fabrication operations related to the rod failures. It is important to distinguish between fabrication of the standard assemblies which were made as a regular core production order and fabrication of the special test assembly fuel which was a developmental activity.

As noted above, the introduction of moisture into the standard rods which failed apparently occurred as a result of deviation (interrupted drying cycle) from the specified fabrication procedure. This possible deviation was identified through review of fabrication records; related inspection records could not be found.

The identified deviation suggests that the pellets were redried and no inspection record retained, or the pellets were not redried. On the basis of the former condition, increased care in recording and retention of inspection data must be recommended. The latter condition (which is more likely, since light hydrogen was found in the failed rods and because extensive CVTR core fabrication and inspection documentation has been confirmed in other parts of this investigation) emphasizes the need to adhere to the specified manufacturing procedures.

The CVTR special assembly fuel was made under a development program, not as a regular core production effort. The quality control aspect of the job was well executed by the responsible organizational component, and only after product or process deviations were identified does review of the quality assurance information suggest areas for improvement. Development work normally requires extensive measurement and documentation of the test specimens. Most frequently, these data are collected by laboratory and development engineering personnel conducting the program, rather than through increased scope of the quality assurance component activity. Specific recommendations which result from this evaluation are:

1. Prime responsibility for the accumulation of all pre-irradiation data (not only inspection data) should be assigned to the quality control function.
2. The disposition of deviations from the process or product specifications should be handled with the same formality of documentation, review, and approval as is used routinely in core production. While manufacturing, design, and quality assurance personnel resolve deviation cases in normal manufacturing, the functional review and approval sequence should be extended to include the responsible development engineer in the case of test programs.
3. Deviations are handled on an individual case basis. If more than one deviation is identified in a fuel rod, the combined effect of all accepted deviations should be assessed and formally accepted using the same procedure with which the individual deviations are handled before the test rod is accepted for reactor insertion.
4. The identification, selection, and retention of archive samples and the complete documentation of fabrication and inspection data for all test fuel must be rigorously enforced to assure successful completion of fuel development programs.

SECTION 9

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SECTION 10

REFERENCES

1. M. G. Balfour, E. S. Foster, H. M. Ferrari, E. Paxson, J. D. Korthur, and L. S. Watley, "Design, Fabrication, and Operation of the CVTR High-Power, High-Burnup Test Assemblies", CVNA-274, June 1967.
2. A. Selz, L. J. Balog, L. P. Hornak, and G. H. Eng, "Mechanical Design of the CVTR Fuel Assembly and Thermal Baffle", CVNA-94, June 1962.
3. R. O. Sandberg and A. A. Bishop, "CVTR Thermal-Hydraulic Design for 65 MW Gross Fission Power", CVNA-227, February 1965.
4. R. J. Nath and F. M. Alligood, Jr., "CVTR Fuel Management", CVNA-287, June 1967.
5. R. O. Sandberg, "Analysis of CVTR Thermal-Hydraulic Performance at 65 MW Operation", CVNA-286, June 1967.
6. R. J. Nodvik, "Evaluation of Mass Spectrometric and Radiochemical Analyses of Yankee Core I Spent Fuel", WCAP-6068, March 1966.
7. M. J. O'Boyle, "Yankee Core Evaluation Program Progress Report", WCAP-6083, August 1967.
8. H. C. Hecker, J. Jedruch, G. N. Hamilton, and J. A. Roll, "Power Distributions and Control Rod Worths in a Two-Region, Clustered $\text{UO}_2\text{-D}_2\text{O}$ Lattice", CVNA-134, September 1963.
9. D. G. Sammarone and W. W. Cameron, "Chemical Surveillance of the Carolinas-Virginia Tube Reactor", CVNA-285, June 1967.
10. S. Bartnoff and D. N. Fultonberg, "Research and Development Program Quarterly Progress Report", CVNA-279, December 1966.

APPENDIX A
BURNUP AND POWER CALCULATIONS
GAMMA SCAN DATA

1. BURNUP DATA: TABLES A-1, A-2, AND A-3

The nuclear parameters used to reduce the burnup data (cf. § 6.2.1) are presented in Table A-1, and the reduced burnups are given in Table A-2 for each radioisotope analyzed. Normalized average fuel burnups are listed in Table A-3. The peak-to-average axial power ratios (Table A-4) have been used to calculate peak power levels for each fuel rod.

2. PREDICTED BURNUPS: TABLE A-3

Table A-3 lists predicted burnup levels, based upon fuel rod loading and recorded exposure. Tabulated first are total equivalent exposure hours for Core I and Core II operation, as tabulated in CVNA^[4]. An equivalent exposure hour (EFPH) is defined as the core average power fuel assembly operating one hour at full plant power (65 MWt) multiplied by a local-to-average radial factor. The latter is obtained from thermal and hydraulic monitoring of individual pressure tubes within the core.

Average Fuel Rod Burnup

The following calculation converts equivalent exposure hours into assembly average burnup:

$$\left[\frac{\text{MWD}}{\text{MTU}} \right] \frac{\text{Assembly}}{\text{Average}} = \text{EFPH} \times \frac{65 \text{ MW/core}}{72 \text{ assemblies/core}} \times \frac{1 \text{ day}}{24 \text{ hr}} \times \frac{1}{\text{gU/assembly}} \times \frac{10^6 \text{ gU}}{\text{MTU}}$$

For the special test assemblies the total weight (in grams) of the UO₂ as measured during loading of individual rods is used to calculate assembly

TABLE A-1

NUCLEAR PARAMETERS USED IN REDUCTION OF CVTR BURNUP DATA

Parameter	Burnup Indicator		
	Heavy Element	Nd-148	Cs-137
U-235 Mev/fission ^(a,b)	201 ± 4	201 ± 4	201 ± 4
U-238 Mev/fission ^(a,b)	198 ± 4	198 ± 4	198 ± 4
Pu-239 Mev/fission ^(a,b)	208 ± 4	208 ± 4	208 ± 4
Pu-241 Mev/fission ^(a,b)	210 ± 4	210 ± 4	210 ± 4
δ^{28} , total of fissions in U-238 relative to sum ^(a) of total fissions in U-235, Pu-239 and Pu-241.	.0862 ± .0086		
α^{49} , total captures/total fissions in Pu-239 ^(a)	.577 ± .058		
α^{41} , total captures/total fissions in Pu-241 ^(a)	.330 ± .033		
U-236 burnout correction factor ^(a)	1.060 ± .006		
Pu-242 burnout correction factor ^(a)	1.111 ± .022		
Fission product burnout correction factor ^(a)		1.005 ± .001	1.0
Half life, years ^(c)		stable	30.0 ± .600
U-235 fission product yield, atom/fission ^(c)		0.0166 ± .0003	.0618 ± .0013
U-238 fission product yield, atom/fission ^(c)		.019 ± .001	.062 ± .004
Pu-239 fission product yield, atom/fission ^(c)		.0165 ± .0005	.0656 ± .0020
Pu-241 fission product yield, atom/fission ^(c)		.0188 ± .0009	.0646 ± .0032

Notes: (a) Nominal values shown here were obtained from a LEOPARD calculation of a Yankee 3.4 w/o unit cell. Actual values used in data reduction are burnup - dependent.

(b) Includes kinetic energy of fission fragments and neutrons, energy of beta and gamma decay, and energy of radiative capture in fuel, cladding and moderator.

(c) Reference: B. F. Rider, C. P. Ruiz, J. P. Peterson, Jr., F. R. Smith, "A Survey and Evaluation of Thermal Fission Yields for U-235, Pu-239, U-233, and Pu-241", USEAEC Report, GEAP-5356 (Sept. 1967).

TABLE A-2

MEASURED BURNUP, CVTR FUEL RODS

Fuel Assembly	Fuel Rod Sample No.	Inches from Bottom of Fuel Column	Initial Fuel Composition (weight pct.)				Measured Burnup (MWD/MTU)			U-235 Fractional Depletion
			U-234	U-235	U-236	U-238	Nd-148	Heavy Element	Cs-137	
G-1	44.731 BU-1	25-1/2 - 26	0.027	4.7	0.012	95.261	8441 ± 216	8648 ± 546	5998 ± 410	.213 ± .008
	44.731 BU-2	50 - 50-1/2	0.027	4.7	0.012	95.261	9064 ± 236	10334 ± 565		.248 ± .008
G-1	24.732 BU-1	9-3/4 - 10-1/4	0.027	4.7	0.012	95.261	6495 ± 168	7272 ± 532		.179 ± .009
	24.732 BU-2	56 - 56-1/2	0.027	4.7	0.012	95.261	8893 ± 225	9367 ± 543		.230 ± .008
G-3	53.861 BU-1	22-1/2 - 23-1/8	0.031	3.8	0.015	96.154	6374 ± 166	6646 ± 434	5216 ± 357	.199 ± .008
	53.861 BU-2	50-1/2 - 51-1/8	0.031	3.8	0.015	96.154	7059 ± 182	7033 ± 429	5853 ± 400	.211 ± .008
G-3	53.862 BU-1	23-1/2 - 24-1/8	0.031	3.8	0.015	96.154	6103 ± 157	6209 ± 425	5384 ± 368	.187 ± .008
	53.862 BU-2	50-1/2 - 51-1/8	0.031	3.8	0.015	96.154	6795 ± 174	6990 ± 434	6220 ± 425	.210 ± .008
G-4	83.831 BU-1	23-1/2 - 24-1/8	0.031	3.8	0.015	96.154	7855 ± 203	7805 ± 451	6769 ± 462	.232 ± .008
	83.831 BU-2	50-1/2 - 51-1/8	0.031	3.8	0.015	96.154	8400 ± 215	8730 ± 444	7268 ± 496	.258 ± .008
G-4	83.832 BU-1	23-1/2 - 24-1/8	0.031	3.8	0.015	96.154	7829 ± 216	7955 ± 445	7376 ± 504	.238 ± .008
	83.832 BU-2	50-1/2 - 51-1/8	0.031	3.8	0.015	96.154	8725 ± 235	8785 ± 449	9045 ± 618	.261 ± .008
502-1-18	1475 BU-1	23-1/4 - 24-1/4	0.014	2.0	0.037	97.949	9161 ± 224	9612 ± 303		.476 ± .005
	1521 BU-1	25-1/2 - 26	0.014	2.0	0.037	97.949	16705 ± 370	17022 ± 543	17617 ± 1183	.721 ± .003
	1591 BU-1	25-1/2 - 26	0.014	2.0	0.037	97.949	16474 ± 360	16443 ± 518		.707 ± .003
502-2-31	1257 BU-1	25-1/2 - 26	0.014	2.0	0.037	97.949	12918 ± 296	13076 ± 400		.597 ± .004
	1257 BU-2	82 - 82-1/2	0.014	2.0	0.037	97.949	8560 ± 198	8707 ± 295		.426 ± .006
502-2-31	1398 BU-1	25-1/2 - 26	0.014	2.0	0.037	97.949	15776 ± 345	15582 ± 484	10998 ± 744	.538 ± .005
	1398 BU-2	82 - 82-1/2	0.014	2.0	0.037	97.949	11197 ± 259	11326 ± 349		.682 ± .003
502-2-31	1549 BU-1	6-5/8 - 7-3/4	0.014	2.0	0.037	97.949	10918 ± 256	11492 ± 349		.544 ± .005
	1549 BU-2	23 - 23-5/8	0.014	2.0	0.037	97.949	14230 ± 324	14224 ± 449		.630 ± .004
	1549 BU-3	75 - 75-1/2	0.014	2.0	0.037	97.949	12207 ± 293	12743 ± 386		.586 ± .004

TABLE A-3

PREDICTED AND MEASURED FUEL ROD AVERAGE BURNUP

Predicted Assembly Average Exposure							Uranium Loading Per Rod (grams)	Fuel Rod Predicted Burnup			Fuel Rod Measured Burnup		
Fuel Assembly	Core I EFPH	Core II		Accumulated EFPH	Accumulated MWD/MTU	Fuel Rod No.		(MWD/MTU)			MWD/MTU		
		Cycle 1 EFPH	Cycle 2 EFPH					Average	Peak/Avg ^(h)	Peak	Average ^(g)	Peak/Avg ^(j)	Peak
G-1	none	1648	469	2117	7826	44.731 24.732	1669.8 1669.3	8660 ^(c) 8670	1.169 1.169	10,840 10,860	7870 7900	1.168 1.191	9190 9410
G-3	none	1563	259	1822	6617	53.861 53.862	1803.6 1782.4	5760 ^(d) 5830	1.169 1.169	7210 7300	6110 5910	1.183 1.169	7230 6910
G-4	none	1365 ^(b)	1094	2459	8312 ^(b)	83.831 83.832	1914.9 1882.1	8050 8190	1.160 1.160	10,000 10,180	7360 7490	1.166 1.171	8580 8770
502-1-18	8251	3005	1808	13064	12910	1475 1521 1591	2002.7 ^(e) 2002.7 2002.7	10970 ^(f) to 14850	1.25 1.25 1.25	14,330 to 19,400	13,180 ⁽ⁱ⁾ 14,330 14,400	1.190 1.199 1.181	15,680 17,180 17,010
502-2-31	8413	2798	1522	12733	12580	1257 1549 1398	2002.7 ^(e) 2002.7 2002.7	10690 ^(f) to 14470	1.25 1.25 1.25	13,960 to 18,900	10,800 12,040 13,600	1.224 1.238 1.178	13,220 14,910 16,020

- Notes: a. Equivalent exposure hours (EFPH) from Appendix III, CVNA-287
b. Re-evaluation of reported Cycle 1 burnup for G-4 (core location 3S).
c. Includes actual fuel rod loading and a 1.09 factor for rod enrichment variation.
d. Includes actual fuel rod loading and a 0.91 factor for rod enrichment variation.
e. Individual rod loading based upon an initial core loading of 6852 pounds UO₂ (CVNA-147).
f. Assumes a ±15% variation about the assembly average burnup.
g. Normalized values, based upon Nd-148 (Table A-2) and gross gamma scan (Table A-4) data.
h. Time-average, axial peak/average factor. (CVNA-274)
i. Extrapolated from gross gamma activity distribution, assembly 502-1-18.
j. Table A-4.

average burnup; the nominal standard assembly weight is determined from the initial core loading of 6852 lb UO₂. The relationship for standard assembly burnup and equivalent exposure hours is then:

$$\left[\frac{\text{MWD}}{\text{MTU}} \right]_{\text{standard assembly average}} = 0.988 \text{ EFPH} \quad (1)$$

For the standard fuel assemblies, the assembly average burnup is equivalent to the average fuel rod burnup, since all rods are identical in fuel design. Because of differing enrichments, fuel densities and fuel loading, "average assembly" burnup in the special assemblies is an artificial quantity. Corrections must be made to convert this into actual fuel rod burnups, dependent on the type or rod being considered. For assemblies G-3 and G-1, with mixed (3.8 w/o and 4.7 w/o) enrichments:

$$\begin{aligned} \left[\frac{\text{MWD}}{\text{MTU}} \right]_{\text{4.7 w/o rod average}} &= \left[\frac{\text{MWD}}{\text{MTU}} \right]_{\text{assembly average}} \times \frac{(\text{gU}) \text{ assembly average rod}}{(\text{gU}) \text{ 4.7 w/o rod}} \times F \text{ w/o} \\ &= \left[\frac{\text{MWD}}{\text{MTU}} \right]_{\text{assembly average}} \times \frac{1}{6} \times \frac{(\text{gU}) \text{ total assembly}}{(\text{gU}) \text{ 4.7 w/o rod}} \times F \text{ w/o} \end{aligned} \quad (2)$$

where:

$F \text{ w/o}$ = is a power sharing factor measured in the critical experiments.*

$F \text{ w/o}$ = 1.09 for 4.7 w/o enriched rods, and 0.91 for 3.8 w/o enriched rods in the case of special assemblies containing each kind of fuel rod.

Peak Fuel Rod Burnups

For the standard fuel rods, equation (1) is multiplied by the appropriate hot-channel factor to give:

* See CVNA-287, Reference 4.

$$\left[\frac{\text{MWD}}{\text{MTU}} \right]_{\text{rod peak}} = \left[\frac{\text{MWD}}{\text{MTU}} \right]_{\text{rod average}} \times (1.045) (1.15) \times \bar{F}_z \quad (3)$$

where 1.045 and 1.15 represent the engineering and assembly fuel rod cluster factors, respectively. The time-average axial factor has been taken as $\bar{F}_z = 1.25$, based upon previously reported burnup data^[1,4] for the time of failure.

For the special assemblies, equation (2) becomes:

$$\left[\frac{\text{MWD}}{\text{MTU}} \right]_{\text{rod peak}} = \left[\frac{\text{MWD}}{\text{MTU}} \right]_{\text{rod average}} \times (1.045) (1.025) \times \bar{F}_z \quad (4)$$

where 1.025 represents the flux skew factor measured in the critical experiments. The axial hot channel factors $\bar{F}_z$ are obtained from flux wire measurements.

3. POWER LEVELS OF CVTR FUEL RODS AT FAILURE

Special Assemblies

Peak linear power densities (kw/ft) at failure were calculated from the expression:

$$\left(\frac{\text{kw}}{\text{ft}} \right)_{\text{peak}} = \left(\frac{\text{average U-tube power}}{\text{total length fuel}} \right) (F_{\text{RU}}) (F_{\text{split}}) (F_z) (F_{\text{w/o}}) (F_\phi) (F_q^E) \quad (5)$$

where:

- F_{RU} = local assembly power/core average U-tube power
- F_{split} = local assembly power/total U-tube power
- F_z = axial peak-to-average power ratio
- $F_{\text{w/o}}$ = power sharing factor to convert assembly average power density to local rod power density for fuel assemblies of two enrichments

$F_{w/o} = 1.00$ for assembly G-4, which has the same enrichment
(3.8 w/o) for all fuel rods

$F_{\phi} =$ factor to account for difference in fuel rod power sharing
due to a flux gradient across the assembly

$F_q^E =$ engineering factor
 $= 1.045$ as determined in Safeguard Analysis*

The first four factors in equation (5) were measured for the time of failure, and are considered accurate. Evaluation of the burnup data confirms the critical experiment factor $F_{w/o}$. Special assemblies G-1 and G-3 occupied symmetric core locations, and $F_{w/o}$ can be inferred from the measured peak burnups (Table A-3) if the data from the two assemblies are normalized with respect to exposure (EFPH) and fuel loading. This gives a ~ 1.07 variation about the assembly average power density, as compared to a predicted value of 1.09 from critical experiments. Further confirmation can be obtained from the gross gamma scans performed on the rods from G-1 (Table A-6). Taking corrected data, the ratio of the integrated activities of the 4.7 w/o rods to the assembly average is ~ 1.10 .

The flux skew peaking factor $F_{\phi} = 1.025$ is consistent with observed peak burnup variations among fuel rods of the same enrichment of 1.012, 1.023, and 1.011 for assemblies G-1, G-3, and G-4, respectively.

Standard Assemblies

The peak kw/ft at failure were calculated from the expression:

$$kw/ft = \frac{\text{average U-tube power}}{\text{total length fuel}} (F_{RU-tube}) (F_{split}) (F_z) (G_o) (F_q^E) \quad (6)$$

* CVNA-90, Final Hazards Summary Report, Volume II (December 15, 1961).

where:

$$\begin{aligned} G_o &= \text{radial cluster factor: highest average fission rate across a} \\ &\quad \text{fuel rod/average fission rate in a fuel rod within the 19-rod} \\ &\quad \text{assembly} \\ &= 1.25, \text{ from critical experiments} \end{aligned}$$

Measured values of the assembly radial power sharing and the axial hot channel factor are available for the time of failure. However, previously reported values of 13.6 kw/ft and 12.3 kw/ft for assemblies 502-1-18 and 502-2-31, respectively, assume a 1.25 radial cluster factor. Several relevant observations can be made from the burnup data:

1. In each case, the failed rod was not the maximum burnup rod in its respective assembly. This is observed from the gamma scan and Nd-148 results.
2. The measured burnup activity distribution (from gamma scans) varies by only ± 15 percent (approximately) about the assembly average.

The previously assumed radial peak rod-to-average rod power ratio of 1.25 represents the highest (and therefore conservative) factor expected for the standard assemblies, based upon critical experiment results.* For example, the peak-to-average power away from a control rod was found to be only ~ 1.16 . Assembly 502-2-31 (7K) was located next to banked control rod group II at the time of failure. However, the rods were banked at a height of approximately 45 inches (above the bottom of the fuel); radial power distribution at the axial location of the peak power (~ 25 inches) would be relatively unaffected by this rod configuration. Assembly 502-1-18 (6I) was not adjacent to a control rod at this time. For this reason, the measured peak-average power distribution ratio of 1.15 seems a reasonable estimate for the standard assemblies at the time for failure.

As mentioned, the post-irradiation measurements indicate the failed rods were not the peak burnup rods in their respective assemblies. However, a precise

* See CVNA-134, Reference 8.

radial factor cannot be assigned to these rods at failure because of the following uncertainties: a) inherent measurement errors in the burnup (± 4 percent) and gamma scan (± 5 percent) results, and b) previous irradiation of the assemblies at peripheral core locations with individual rod orientation not specified. Therefore, it is somewhat conservatively assumed that the appropriate radial factor for the failed fuel rods is 1.15. Calculated peak power ratings at failure then are 11.3 and 12.5 kw/ft for rods 1549 and 1475, respectively.

4. AXIAL GAMMA SCANS

Table A-4 summarizes the CVTR gross gamma scan results. Tables A-5 and A-6 present gamma scan data with spacer wires intact, performed on the standard fuel rods and special assembly G-1 fuel rods, respectively.

Axial gross gamma activity as a function of fuel column height is plotted in the following Figures (A-1 to A-6) for:

Fuel Rods 1257, 1398, and 1549 from standard assembly 502-2-31
1591 and 1475 from standard assembly 502-1-18
24.732 from special assembly G-1

The gamma scans from other CVTR rods destructively examined are reported in WCAP-3850-2 and WCAP-3850-3.

TABLE A-4

SUMMARY OF CVTR FUEL ROD GAMMA SCANS

<u>Fuel Assembly</u>	<u>Core Location</u> <u>U-Tube</u>	<u>Fuel Rod</u>	<u>Length of Fuel (Inches)</u>	<u>Irradiation Time^(a) days</u>	<u>Date of Final Irradiation</u>	<u>Date of Gamma Scan</u>	<u>Cooling Time (days)</u>	<u>Rod Average Gamma Activity^(b) x 10⁹</u>	<u>Peak/Avg Ratio</u>	<u>Notes</u>
G-1	8R	8Q	44.731	77.81	149	12/ 2/66	10/20/67	0.927	1.168	c
			24.732	77.81	149	12/ 2/66	10/24/67	0.884	1.191	c,e
G-3	3B	3C	53.861	81.06	127	10/27/66	1/29/68	0.351	1.183	c
			53.862	81.06	127	10/27/66	1/29/68	0.335	1.169	c
G-4	3R	3S	83.831	81.12	196	1/24/67	1/29/68	0.669	1.166	c
			83.832	81.16	196	1/24/67	1/29/68	0.689	1.171	c
502-1-18	8B/6J	8A/6I	1475	95.4	733	1/24/67	10/30/67	1.140	1.190	c,f
			1521	95.4	733	1/24/67	10/30/67	1.206	1.199	c
			1591	95.4	733	1/24/67	11/ 8/67	1.125	1.181	c
502-2-31	8R/7J	8S/7K	1257	95.4	733	1/24/67	10/30/67	0.893	1.224	c
			1398	95.4	733	1/24/67	10/19/67	1.071	1.178	c
			1549	95.4	733	1/24/67	8/21/67	1.522	1.238	d

Notes: a. Calendar days of reactor operation at some power level.

b. Rod activity not corrected for decay; rods with dissimilar irradiation and cooling histories not directly comparable.

c. Scanned with spacer wire removed. Average of axial scans with separate orientations at 0°, 90°, and 180°.

d. Scanned with spacer wire intact.

e. Scanned separately in the three sections: 0-13 in., 17-31 in., and 31-103 in., Gamma activity was extrapolated for unscanned 13-17 in. section.

f. Upper ~ 6 in. of activity curve extrapolated using rod #1591 data.

TABLE A-5

GROSS GAMMA ACTIVITY DISTRIBUTION: STANDARD CVTR ASSEMBLIES

Assembly 502-2-31 (8/21/67)Assembly 502-1-18 (8/16/67)

<u>Rod No.</u>	<u>Integrated Gamma_g Activity x 10⁹ amp-in.</u>	<u>Local- Average Ratio</u>	<u>Rod No.</u>	<u>Integrated Gamma_g Activity x 10⁹ amp-in.</u>	<u>Local- Average Ratio</u>
1398 ^a	145.62	1.056	1475 ^a	162.38	1.054
1369	155.92	1.131	1534	156.00	1.013
1549 ^a	143.19	1.039	1524	163.60	1.062
1368	145.67	1.057	1521 ^a	175.83	1.142
1257 ^a	117.61	0.853	1581	162.38	1.054
1216	151.56	1.099	1568	166.55	1.081
1613	123.31	0.894	1487	153.65	0.998
1354	146.57	1.063	1580	169.08	1.098
1189	139.65	1.013	1649	153.12	0.994
1410	141.42	1.026	1492	164.93	1.071
1682	121.16	0.879	1545	135.29	0.878
1339	128.33	0.931	1481	157.60	1.023
1224	140.38	1.018	1642	148.50	0.964
1387	124.62	0.904	1576	148.86	0.966
1191	153.28	1.112	1499	140.55	0.912
1171	139.54	1.012	1572	131.76	0.885
1230	121.40	0.881	1647	139.94	0.909
1240	127.46	0.925	1464	130.37	0.846
1227	152.65	1.107	1591 ^a	165.36	1.074

a. Destructively examined.

TABLE A-6

GROSS GAMMA ACTIVITY, G-1 FUEL RODS

The fuel rods were gamma scanned between June 1 - 9, 1967, with wire spacers intact; data for rod No. 24.732 is not available because of the rod fracture. The data have been corrected for decay to the date June 1, 1967, and normalized to a fuel column length of 77.8 inches. Rods 33.833 and 44.732 were each scanned twice.

<u>Fuel Rod No.</u>	<u>Fuel Column Length (inches)</u>	<u>Corrected Integrated Activity ($\times 10^9$ amp-in.)</u>
33.833 ^a	81.13	200.3
33.833 ^a	81.13	200.1
44.731 ^a	77.81	230.2
44.732	77.81	229.4
44.732	77.81	232.3
33.832	81.06	183.4
W13.831	80.25	187.5

a. Destructively examined.

Figure A-1. Fuel Rod 1549, Axial Gamma Activity Profile
(Wire Spacer Intact)

Figure A-2 Fuel Rod 1398: Axial Gamma Activity Profile

A 15

Figure A-3. Fuel Rod 1257: Axial Activity Profile

Figure A-4. Fuel Rod 1475: Axial Gamma Activity Profile

Figure A-5. Fuel Rod 1591: Axial Gamma Activity Profile

Figure A-6. Fuel Rod 24.732: Axial Gamma Activity Profile

APPENDIX B

CRUD MAPS, STANDARD FUEL RODS

Figure B-1. Fuel Rod 1475: Crud Distribution

Figure B-2. Fuel Rod 1521: Crud Distribution

Figure B-3. Fuel Rod 1549: Crud Distribution

Figure B-4. Fuel Rod 1257; Crud Distribution

APPENDIX C

ADDITIONAL HYDROGEN/DEUTERIUM DATA

AXIAL HYDROGEN/DEUTERIUM DISTRIBUTION

TABLE C-1

ADDITIONAL HYDROGEN/DEUTERIUM DATA

Rod. No.	Axial Location (In. From Bottom)	Sample	Weight (gm)	Species (Mol.%)			ppm ^(a)	
				<u>H₂</u>	<u>HD</u>	<u>D₂</u>	<u>H</u>	<u>D</u>
1591	100-100-1/4	(End Plug)	0.8583	92.5	7.2	0.3	13.5	1.1
53.862	100-100-1/4	(End Plug)	0.5713	84.5	14.7	0.8	11.9	2.1
83.831	14-14-1/4	A	0.0977	3.3	19.3	77.4	92.3	1242.0
		B	0.1189	26.5	45.5	28.0	43.9	82.2
		C	0.1098	33.5	45.0	21.5	47.6	74.8
		D	0.0968	58.0	30.0	12.0	51.6	38.9
83.832	~21	A	0.1703	45.5	42.4	12.1	63.0	62.0
		B	0.1552	46.4	41.6	12.0	73.4	71.5
		C	0.1114	53.8	36.8	9.4	110.6	84.5
		D	0.1378	44.3	42.4	13.3	75.4	79.4

a. Parts per million (by weight): H is calculated as mass one, D as mass two.

Figure C-1. Clad Hydrogen/Deuterium Distribution. Assembly 502-1-18

Figure C-2. Clad Hydrogen/Deuterium Distribution, Assembly 502-2-31

Figure C-3. Clad Hydrogen/Deuterium Distribution, Assembly G-1

Figure C-4. Clad Hydrogen/Deuterium Distribution. Assembly G-3

Figure C-5. Clad Hydrogen/Deuterium Distribution, Assembly G-4

APPENDIX D
SECTIONING DIAGRAMS

KEY TO FIGURES D-1 THROUGH D-10:

Met (T) = transverse metallographic section
Met (L) = longitudinal metallographic section
H/D = hydrogen/deuterium analysis section

Figure D-1. Fuel Rod 1475. Sectioning Diagram

Figure D-2. Sister Fuel Rods 1521 and 1591: Sectioning Diagrams

Figure D-3. Fuel Rod 1549: Sectioning Diagram

Figure D-4. Sister Fuel Rods 1257 and 1398 Sectioning Diagrams

Figure D-5. Fuel Rod 24.732: Sectioning Diagram

Figure D-6 Sister Fuel Rods 44 731 and 33 833 Sectioning Diagrams

Figure D-7. Fuel Rod 53.861: Sectioning Diagram

Figure D-8. Sister Fuel Rods 24.734 and 53.862:Sectioning Diagrams

Figure D-9 Fuel Rod 83 831 Sectioning Diagram

Figure D 10 Fuel Rod 83 832 Sectioning Diagram

APPENDIX E

RELATED FAILURE EXPERIENCE

Because of certain parallels to the CVTR failures, a brief summary of relevant failure experience at other sites (mostly heavy water reactors) is presented. No attempt has been made to produce a comprehensive survey.

1. FAILURE EXPERIENCE AT THE PRTR

A series of 36 Zircaloy clad $\text{UO}_2\text{-PuO}_2$ fuel (swaged and vipac mixed) rods failed in the PRTR between January, 1963, and January, 1965. Thirty-four of the failures have been attributed to impurities in the fuel material, resulting in failure at very low burnups (generally less than 500 MWD/tonne). Discovery and elimination of the impurity problems were successful in eliminating the failure of this type of rod. The fluoride (from the PuO_2) contaminated rods were dissimilar to the CVTR type of failure: the defects were characterized by radial pinholes, with no attendant hydriding.

Failed fuel rods which had been contaminated by moisture or hydrocarbons during fabrication processes were characterized by localized hydriding and clad deterioration. The crevice regions of the end caps were particularly susceptible to hydriding, and the massive hydriding at these regions was similar in appearance to blister defects in the CVTR rods. The hydriding at the end caps has been attributed to a gas phase mechanism, resulting from the fuel contaminants. Similar results were produced in Zircaloy tubes in the laboratory.

One failure occurred at relatively high burnup (3900 MWD/MTU) with small pinholes in the clad. There were several local regions of massive hydriding, visible on the outside of the clad as grey spots. The fuel material in this rod was not suspected of the same contamination causing the other failures.

It has been suggested that the local hydriding represents a slower reaction caused by a relatively low level of fuel contamination (40 ppm of oil was observed).

REFERENCES

1. M. D. Freshley, Battelle Pacific Northwest Laboratories, Richland Washington, Personal Communication.
2. Ceramics Research and Development Reports: HW-81600, HW-81601, HW-81602, BNWL-91, and BNWL-150.

2. FAILURE EXPERIENCE AT SAVANNAH RIVER

A series of failures occurred in vibratory compacted and swaged annular fuel rods, irradiated at relatively low temperatures and coolant pressures in Savannah River production reactors. Out of a total of 39 short tubes (7 assemblies) with outer and inner sheaths of Zircaloy-2, seven failed. The types of failure were sheath cracking resulting from internal hydriding and embrittlement of the Zircaloy and collapse of the inner sheath caused by excessive internal gas pressure. Four of the failures exhibited longitudinal cracks of the outer clad; failure at low circumferential strains (~ 0.5 percent) was indicative of severe embrittlement. Metallographic examination showed severe local hydriding at the fuel interfaces with both inner and outer sheaths. The hydriding (identified as $\text{ZrH}_{1.5}$) occurred in a few randomly spaced clusters, similar to CVTR observations, or thin semi-continuous layers on the clad ID. Hydride deposits also were found in the inner sheaths of two unfailed rods.

The failures were attributed to UO_2 fuel moisture. Incorporation of a drying step in the fabrication process eliminated the problem. In order to prevent failures, tubes are dried for 20 hours at $250 - 275^\circ\text{C}$ before swaging and sealing, thus reducing residual UO_2 moisture to ~ 15 ppm. Other fabrication procedures adopted for HWCTR fuel elements to prevent a recurrence of previous failures were vacuum outgassing of the fuel at $\geq 1300^\circ\text{C}$ prior to loading and loading of the tubes in a dry, inert atmosphere.

More recent irradiation of 97 annular Zircaloy-clad fuel tubes in the Heavy Water Components Test Reactor (HWCTR) resulted in 11 failures. Eight of the failures have been ascribed variously to sheath wrinkling, mechanical wear from excessive vibration, a faulty weld closure, and molten UO_2 -sheath interaction. Three of the failures, however, are as yet unexplained. One of the tubes failed with a small hole in the outer sheath; metallographic examination revealed local ID pitting and oxidation, a zirconium hydride layer on the outer sheath surface (up to one-half of the wall), and cracking of both inner and outer surfaces. Metallographic results for the other failed tubes have not been reported yet.

REFERENCES

1. T. H. Crim, G. R. Cole, and W. G. Holmes, "Irradiation of Tubular UO_2 Fuel Elements in the HWCTR", DP-995, November 1967.
2. G. R. Caskey, et al., "Failures of UO_2 Fuel Tubes by Internal Hydriding of Zircaloy-2 Sheaths", CEND-153 (Volume II), Symposium on Powder Packed Uranium Dioxide Fuel Elements, November 30 - December 1, 1961.
3. R. H. Hood (Compiler), "Heavy Water Moderated Power Reactor, Progress Report", DP-665, September 1962.

3. RECENT ZIRCALOY-2 CLAD FAILURES IN THE HALDEN REACTOR

Four fuel rods of pelleted UO_2 from assembly IFA-21 developed longitudinal crack defects, after operation of 9500 MWD/t at a nominal heat rating of 500 w/cm. Metallographic examination revealed multiple ID cracks and incipient cracks, as well as local deuteriding on the crack surfaces.

The relatively ductile appearance of the crack defects and a correlation between clad cracking and fuel pellet cracking at the pellet-clad interface have been attributed to a fuel-clad ratchetting mechanism. Local strains produced in the clad, together with neutron irradiation embrittlement are postulated as the primary cause of failure. The observed "hydriding" (assumed to be primarily deuterium) merely enhances propagation of the failure.

This failed mechanism contrasts with the CVTR findings, in which the defect cracks appeared to be a consequence of internal hydriding. Clad ID cracks at the axial location of the peak power during the rod banking experiment were not the result of local hydriding. Moreover, they did not actually penetrate the clad wall and cause failure.

REFERENCES

1. Gustaf Ostberg, "Observations on the Role of Hydride Embrittlement in a Failure of a Zircaloy-2 Clad Fuel Element", AB Atomenergi, Stockholm, Sweden. Contribution to the International Conference on the Use of Zirconium Alloys in Nuclear Reactors, "Zirconium 68", Marianske Lazne, Czechoslovakia, October 1 - 3, 1968.
2. S. Djurle, "Post-irradiation Evaluation of the Third Swedish Fuel Assembly", AE-RMA-810, IFA-21, December 7, 1967.

4. RECENT CANADIAN RESULTS: FUEL ROD FAILURES FROM INTERNAL HYDRIDING

A series of fuel rod failures has occurred in prototypic and development tests for the Douglas Point and Pickering Pressurized Heavy Water Reactor and the Gentilly Boiling Light Water Reactor. The tests are being carried out in the U-2 (pressurized light water) and U-1 (boiling light water) loops of the NRU reactor.

Several of the U-2 loop defects were caused by mechanical damage or gross overpower due to local flux distortions; the third group of failures has been attributed to internal clad hydriding and blistering resulting from residual moisture and hydrogen in the fuel pellets. These failures are characterized by gross hydriding and cracking of the Zircaloy clad, similar to the CVTR failures. Annealing at 1000°C of archive fuel pellets (enriched UO_2) used in the Pickering rods which failed indicates significant amounts of hydrogen gas plus a lesser quantity of moisture may have been released during operation.

Failures have occurred in boiling light water (Gentilly) fuel rod tests (natural UO_2) in the U-1 loop, characterized by local clad hydriding (Zircaloy-4) and blistering. The failures have been attributed to moisture from sodium silicate used to hold molybdenum discs to the top of the fuel column during fabrication. The discs are used to prevent hot UO_2 from entering the void space provided in hollow end caps in the test elements. Hydrating of the sodium silicate is believed to have contributed enough moisture contamination of the rods to cause failure.

REFERENCES

1. R. D. Page, "Engineering and Performance of Canada's UO_2 Fuel Assemblies for Heavy Water Power Reactors", AECL-2949, September 1967.
2. R. D. Page and A. D. Lane, "The Performance of Zirconium Alloy Clad UO_2 Fuel for Canadian Pressurized and Boiling Water Power Reactor", AECL-3068, May 1968.
3. A. S. Bain, AECL, Chalk River, Ontario, Personal Communication.