NUREG/CR-2871
HEDL-TME 82-27
RW

CHARACTERIZATION OF LWR SPENT-FUEL RODS USED IN THE NRC LOW-TEMPERATURE WHOLE ROD AND CRUD PERFORMANCE TEST

Hanford Engineering Development Laboratory

Compiled by
R.E. Einziger
R.L. Fish

Prepared for the U.S. Nuclear Regulatory Commission

~~DO NOT MICROFILM~~
COVER

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission,
   Washington, DC 20555
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the NRC/GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations,* and *Nuclear Regulatory Commission Issuances.*

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

GPO Printed copy price: $5.00

NUREG/CR-2871
HEDL-TME 82-27
RW

NUREG/CR--2871

DE83 002133

CHARACTERIZATION OF LWR SPENT-FUEL RODS USED IN THE NRC LOW-TEMPERATURE WHOLE ROD AND CRUD PERFORMANCE TEST

Hanford Engineering Development Laboratory

Operated by Westinghouse Hanford Company
P.O. Box 1970 Richland, WA 99352
A Subsidiary of Westinghouse Electric Corporation

Compiled by
R.E. Einziger
R.L. Fish

Manuscript completed: August 1982
Date published: September 1982

Prepared for Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN No. B2435

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

NUREG/CR-2871
HEDL-TME 82-27
RW

CHARACTERIZATION OF LWR SPENT FUEL RODS USED IN THE NRC LOW-TEMPERATURE WHOLE ROD AND CRUD PERFORMANCE TEST

R. L. Fish
R. E. Einziger

ABSTRACT

Westinghouse Hanford Company (WHC) and EG&G-Idaho are conducting a joint evaluation for the Nuclear Regulatory Commission (NRC) to provide information in establishing licensing positions relative to long-term, low-temperature performance of spent fuel rods in dry storage and potential crud contamination for the dry storage cycle. The evaluation will establish the spent fuel performance of intact and defected light water reactor (LWR) fuel rods under inert and unlimited air environments. The four H. B. Robinson Unit 2 pressurized water reactor (PWR) fuel rods and four Peach Bottom-II boiling water reactor (BWR) spent fuel rods will be placed in a whole rod test furnace at a temperature between 230 and 245°C for a total of 50 months. Interim and final examinations are planned to assess behavior during the test. A literature search was conducted, and available nondestructive, metallographic, and fission gas data were compiled to provide a basis for selection of actual test rods and a description of the initial spent fuel test rod condition. The H. B. Robinson Unit 2 PWR fuel rods have been well characterized from other programs. The Peach Bottom-II BWR fuel rods are not as well characterized; however, eddy current examinations have been conducted on all the test rods. The visual examinations and crud characterization information to be gathered as part of the pretest rod characterization will be reported in mid-FY 1983.

ACKNOWLEDGMENTS

The authors would like to thank the investigators who gathered the characterization data as parts of other programs: B. A. Murdock, D.E. Owens and V.W. Storhok (EG&G-Idaho); S.M. Gehl (ANL), L.M. Lowry (BCL), J. Gonser (GE-San Jose), and R. A. Lorenz and J. H. Goode (ORNL). Their helpful discussions were invaluable.

CONTENTS

CONTENTS (Cont'd)

FIGURES

FIGURES (Cont'd)

TABLES

ACRONYMS

ANL	Argonne National Laboratory
BCL	Battelle Columbus Laboratory
BWR	Boiling Water Reactor
DE	Destructive Examination
EC	Eddy Current
EFPD	Effective Full Power Days
FCCI	Fuel Cladding Chemical Interaction
FCMI	Fuel Cladding Mechanical Interaction
GE	General Electric
ID	Inside Diameter
LANL	Los Alamos National Laboratory
LWR	Light Water Reactor
NDE	Nondestructive Examination
NRC	Nuclear Regulatory Commission
OD	Outside Diameter
ORNL	Oak Ridge National Laboratory
PIE	Postirradiation Examination
PWR	Pressure Water Reactor
WHC	Westinghouse Hanford Company

CHARACTERIZATION OF LWR SPENT FUEL RODS USED IN THE NRC LOW-TEMPERATURE WHOLE ROD AND CRUD PERFORMANCE TEST

I. SUMMARY

Westinghouse Hanford Company (WHC) and EG&G-Idaho are conducting a joint evaluation of light water reactor (LWR) spent fuel behavior during dry storage for the Nuclear Regulatory Commission (NRC). The objective is to provide information in establishing licensing positions relative to long-term, low-temperature performance of spent fuel rods in dry storage and potential crud contamination for the dry storage cycle. The evaluation will establish the spent fuel performance of intact and defected LWR fuel rods under inert and unlimited air environments. Four H. B. Robinson Unit 2 pressurized water reactor (PWR) rods and four Peach Bottom-II boiling water reactor (BWR) spent fuel rods will be placed in a whole rod test furnace at a temperature between 230 and 245°C for a total of 50 months. Interim and final examinations are planned to assess behavior during the test. Critical to the success of this evaluation is a knowledge of the condition of the fuel rods prior to the test. A literature search was conducted, and available data and information were compiled to provide a basis for selection of actual test rods and a description of the initial spent fuel test rod condition. Visual examinations and crud characterization information gathered as part of the pretest rod characterization will be reported in mid-FY 1983.

The H. B. Robinson Unit 2 PWR fuel rods have been well characterized from other programs. This information and a visual examination of the actual test rods and crud examination of a companion rod will establish the pretest character of these rods.

The Peach Bottom-II BWR fuel rods are not as well characterized as the H. B. Robinson Unit 2 fuel rods. However, eddy current (EC) examinations have been conducted on all test rods, and visual examinations are planned on the actual test rods. A crud examination will be conducted on a companion rod. A full metallographic examination of two samples cut from a companion

Peach Bottom-II fuel rod is planned during the first interim destructive examination (DE) to complement fission gas analyses already conducted on two companion rods and to provide adequate pretest DE information on this BWR fuel. Based on the available Peach Bottom-II fuel nondestructive examination (NDE) results and fission gas release measurements, the planned whole rod test using this fuel should provide a good representation of present-day BWR spent fuel behavior.

II. INTRODUCTION

WHC and EG&G-Idaho are jointly conducting low-temperature, long-term whole rod tests for the NRC to evaluate the behavior of spent fuel. The objective is to provide information in establishing licensing positions relative to the long-term, low-temperature performance of spent fuel rods in dry storage. The primary focus of the test will be to determine the contamination from within the rod due to cladding breach or fuel oxidation and from outside the rod due to crud spallation. To cover the wide range of conditions available for dry storage, an 8-rod matrix, shown in Table 1, was chosen. This matrix includes PWR and BWR rods, breached and unbreached rods, and inert and air atmospheres.

TABLE 1

WHOLE ROD FURNACE LOADING

No.	Type	Condition	Atmosphere
1	PWR	Intact	Argon
2	PWR	Defected	Argon
3	PWR	Intact	Air
4	PWR	Defected	Air
5	BWR	Intact	Argon
6	BWR	Intact	Air
7	BWR	Defected	Argon
8	BWR	Defected	Air

The test will consist of five runs of 10 months each. After each run, an NDE will be conducted. The first run will be at 230° ± 10°C. If no fuel rod or defect degradation is detected, the temperature of the remaining runs will be increased to 245 ± 5°C. After the second and fifth run, one or more rods will also be destructively examined. The results of each of these examinations will be compared to the initial condition of the fuel rod to determine any changes that might have taken place. Complete details of the test can be found in the technical test description.(1)

Rods available in the EG&G TAN facility storage pool from H. B. Robinson Unit 2 (PWR) and Peach Bottom-II (BWR) are being used in the test. Available rod characterization previously performed on these fuel rods is reported here. This information is used:

1) To determine the condition of the rods prior to testing so that the most suitable rods could be chosen and test conditions determined.

2) To minimize necessary pretest characterization.

3) To have an initial reference with which to compare the results of interim and post-test examination data. This comparison will provide the basis for assessing any change in the condition of the rods due to testing.

4) To relate the test rods to the general population of spent fuel rods available for storage and determine if they are adequately representative of the general spent fuel population.

While as much characterization as possible on the actual test rods is desirable, use of data from other programs can eliminate expensive duplication and enable most effective use of available resources. Rods from both the H. B. Robinson Unit 2 and the Peach Bottom-II assemblies have been used in other programs. The H. B. Robinson Unit 2 rods have been examined and tested at EG&G, Battelle Columbus Laboratory (BCL), Argonne National Laboratory (ANL), and Oak Ridge National Laboratory (ORNL). The Peach Bottom-II rods were examined at EG&G and at ORNL. The reports on these examinations and tests form the foundation of this compilation. Visual characteristics of the actual test rods along with extensive crud characterization will be conducted as a part of the test program and will be reported on at a later date. This report will contain two classes of information: 1) general assembly and rod description data along with reactor operating conditions and 2) NDE and DE data.

III. GENERAL FUEL ASSEMBLY INFORMATION

A. REACTOR, ASSEMBLY, AND ROD DESCRIPTION

1. H. B. Robinson Unit 2 PWR

The Carolina Power and Light Company's H. B. Robinson Unit 2 reactor is a 665-MW electric (2192-MWt) three-loop LWR located in Hartsville, SC. It began operation in March 1971.[2,3] The core consists of 157 Westinghouse 15 x 15 assemblies (Figure 1) with three enrichment zones.[2] Each assembly contains 204 fuel rods, 20 control rods, and 1 possible in-core instrumentation tube located as shown in Figure 2. The rods are held in place with seven grids per assembly.

Each rod consists of a column of cylindrical pellets stacked inside a metallic sheath. There is a plenum at the top of the fuel rod to accommodate gas released from the pellets during irradiation. The rods were pressurized with helium prior to irradiation. Although the initial internal pressure is proprietary, postirradiation examination (PIE) indicates (see later section) it was ∿200 psi. There are also indications that the rods might have been pressurized without purging the internal air.[5] Preirradiation characteristics of the fuel rods for Assembly BO-5 are given in Table 2.

2. Peach Bottom-II BWR

The Philadelphia Electric Company's Peach Bottom Unit-II is a 1065-MW electric BWR located at Peach Bottom, PA. It began commercial operation in July 1974. The core contains 764 General Electric (GE) assemblies (see Figure 3). During the first cycle of operation, the core contained three bundle types with average enrichments of either 1.10 wt% or 2.50 wt% (see Table 3). The assembly selected for this study (PH 462) is an improved design 7 x 7. The improvement involves the use of a moisture getter in the plenum region.[7] An extensive reactor description is given in Reference 6.

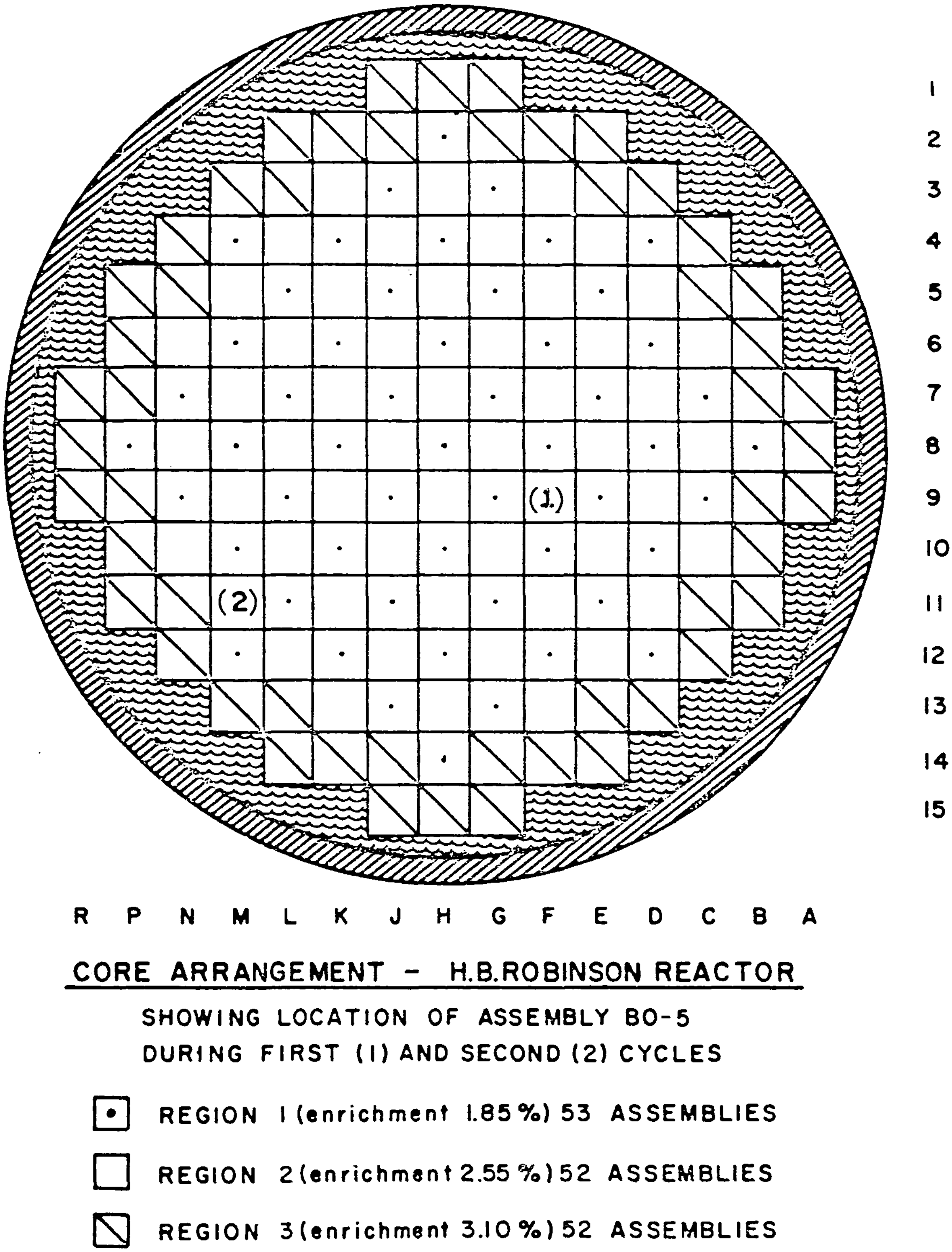


FIGURE 1. Core Arrangement of H. B. Robinson Unit 2 Reactor. (Ref 2)

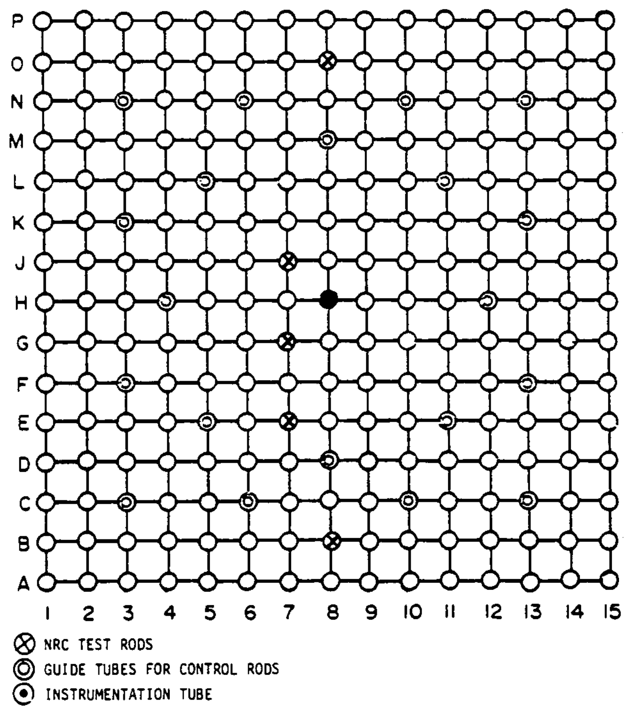


FIGURE 2. Schematic of Fuel Rod Array in H. B. Robinson Unit 2 Assembly BO-5. (Ref 2)

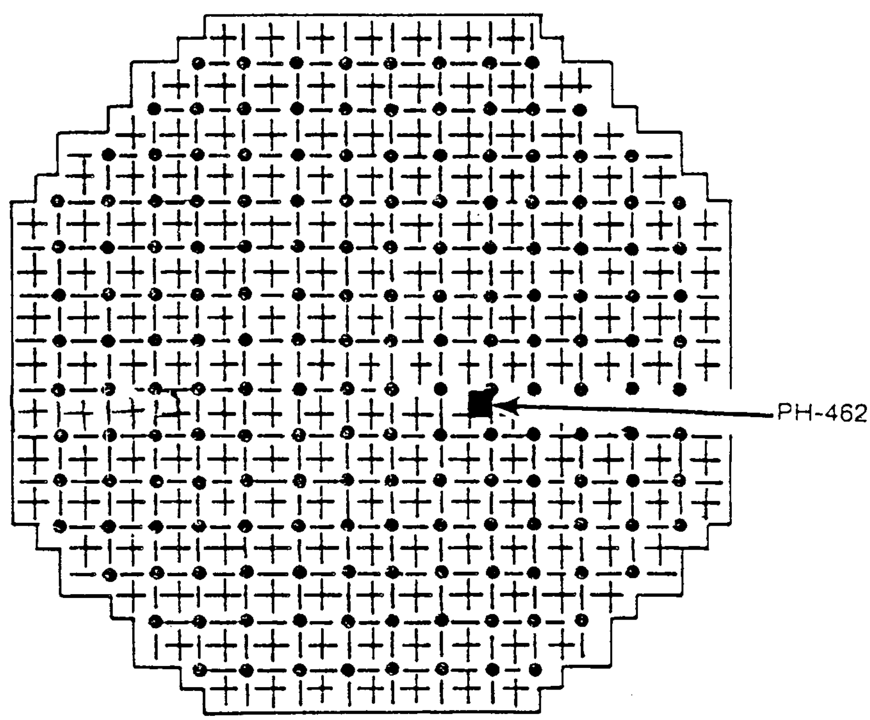


FIGURE 3. Peach Bottom-II Core Map. (Ref 7)

TABLE 2

PRE-IRRADIATION FABRICATION DATA FOR PWR RODS FROM H. B. ROBINSON UNIT 2 ASSEMBLY BO-5[4]

Outside Diameter	10.7 mm
Diametral Gap	0.165 mm
Cladding Thickness	0.62 mm
Cladding Material	Zircaloy-4
Overall Length	3.86 m
Pellet Material	Sintered UO_2
Density (% TD)	1.008×10^4 kg/m^3 (92)
Fuel Enrichment (wt%)	2.55
Pellet Diameter	9.3 mm
Pellet Length	15.2 mm
Fuel Stack Length	3.65 mm
Plenum Length	173.5 mm
End Cap Length	17.5 mm

TABLE 3

PEACH BOTTOM-II CYCLE 1 ASSEMBLY TYPES AND IDENTIFICATION[6]

Assembly No.	Array	Pellets	Enrichment (wt%)	Assembly
PH 001 to PH 168	7 x 7	UO_2	1.10	Type 1 with Gd_2O_3
PH 169 to PH 431	7 x 7	UO_2	2.50	Type 2 with Gd_2O_3 in 4 rods
PH 432 to PH 764	7 x 7	UO_2	2.50	Type 3 with Gd_2O_3 in 5 rods

The 49 rods include 5 Gd_2O_3 rods for control improvement, 8 tie rods, and 1 spacer rod. These rods are held in the bundle with the assistance of 7 grid spacers. The variation of initial enrichment within the bundle and the location of the different types of rods is shown in Figure 4. Note the orientation of the assembly with respect to the control rod blade since the configuration is not symmetric.

A typical rod is shown in Figure 5. It consists of sintered UO_2 pellets within a ®Zircaloy-2 sheath. The pellets are held in place by a spring in the plenum region that contains the moisture getter. The pellet length is proprietary but appears from PIE to be approximately equal to the diameter.[7] The rods were initially pressurized with between 1 to 10 psig of helium. Pre-irradiation fabrication data for the rods are given in Table 4.

B. ASSEMBLY BURNUP AND POWER HISTORY

1. H. B. Robinson Unit 2 Assembly BO-5

The assembly was irradiated in the H. B. Robinson Unit 2 core for cycles 1 and 2 for a total of 799 effective full power days (EFPD), then removed from the reactor on May 6, 1974. The assembly power history is given in Table 5.[3] The peak power (327 W/cm) occurred in December 1971 shortly after irradiation began. By the end of irradiation on May 1974, the power had dropped to 212 W/cm.

Two rods (F-7 and P-8) were sectioned, and burnup analyses were conducted on three samples from each rod. The F-7 sample was analyzed at ANL and the P-8 samples were analyzed at BCL. Details of the analysis method can be found in Reference 8. The results of the ^{148}Nd mass spectrometric burnup measurements are provided in Table 6. Other than the bottom of the rod where the neutron energy spectrum is different, the two analyses agree quite well and indicate a fairly flat flux profile.[10] (See Figure 6.)

®Zircaloy is a registered trademark of Westinghouse Electric Corp., Specialty Metals Division, Blairsville, PA.

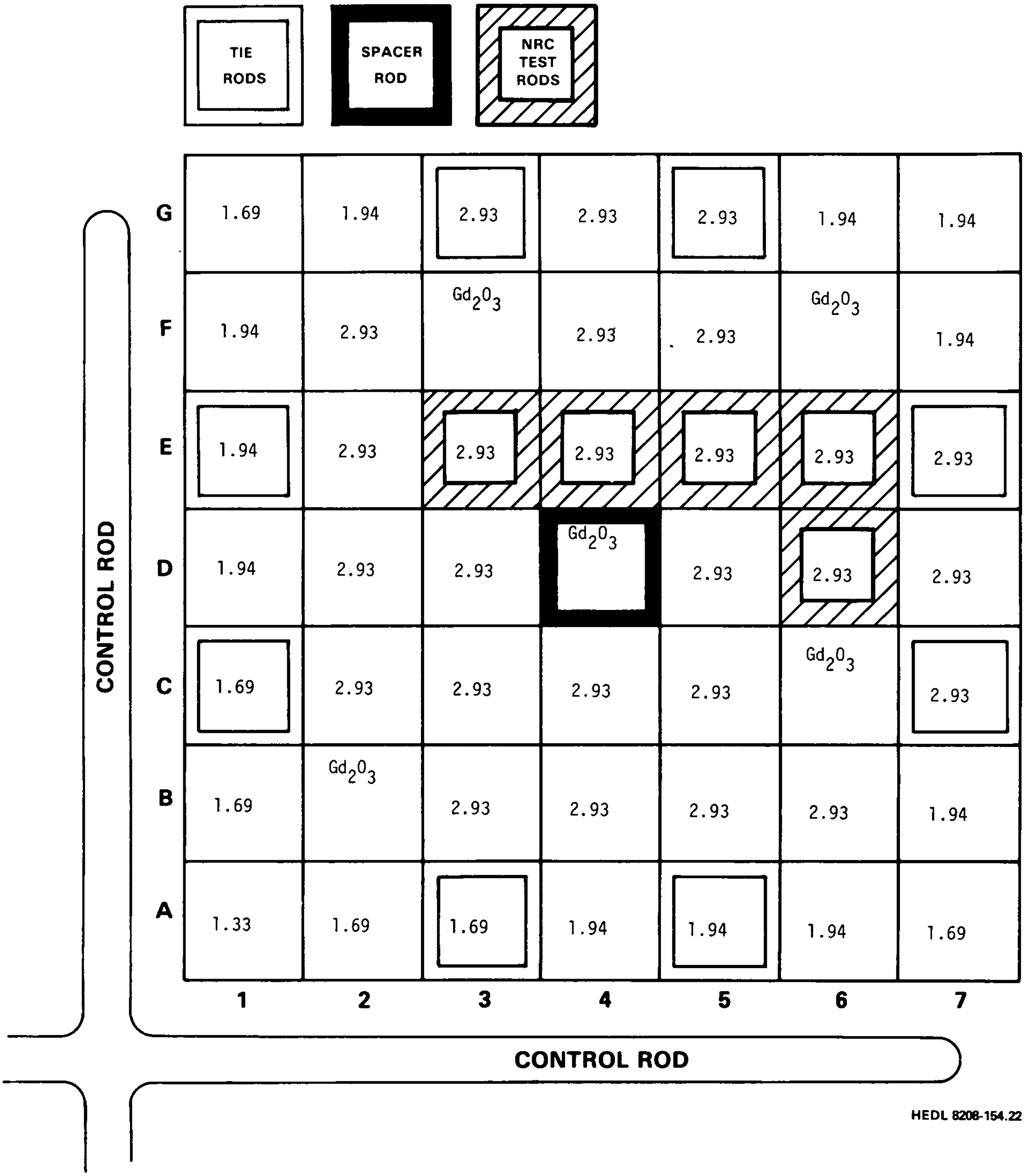


FIGURE 4. Initial Enrichment of Peach Bottom-II Assembly PH 462. (Ref 6)

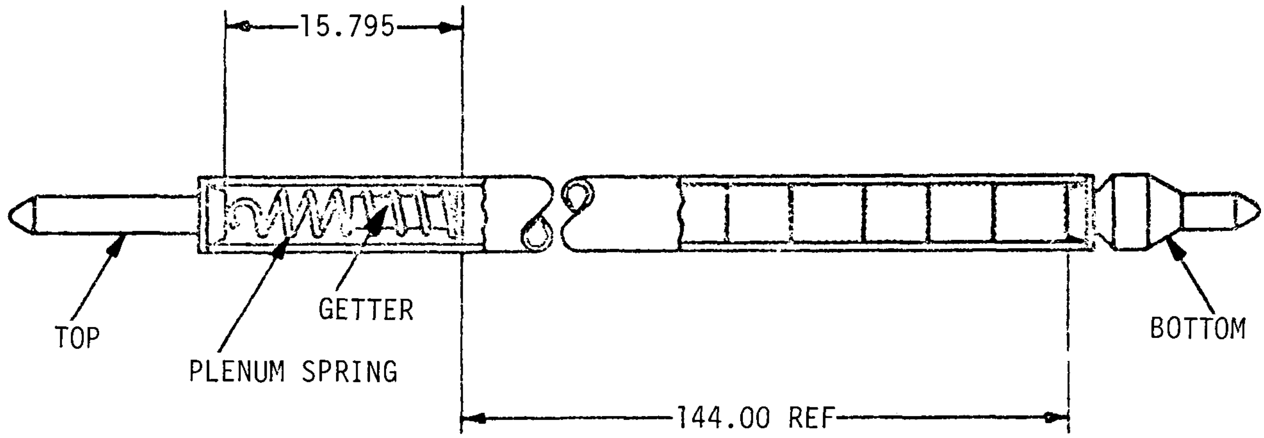


FIGURE 5. Typical Fuel Rod from Peach Bottom-II. (Ref 6)

TABLE 4

PRE-IRRADIATION FABRICATION DATA FOR BWR RODS FROM PEACH BOTTOM-II ASSEMBLY PH 462[6]

Outside Diameter	14.30 mm
Diametral Gap	0.15 mm
Cladding Thickness	0.94 mm
Cladding Material	Zircaloy-2
Pellet Material	Sintered UO_2
Density (% TD)	1.042×10^4 kg/m^3 (95)
Fuel Enrichment (wt%)	see Figure 4
Pellet Diameter	12.12 mm
Pellet Length	∿12 mm[7]
Fuel Stack Length	3.66 m
Plenum Length	401.3 mm

TABLE 5

POWER HISTORY FROM H. B. ROBINSON UNIT 2 ASSEMBLY BO-5[3]

Cycle I

Date	Power (W/cm) Average	Peak	EFPD
Oct. 1971	229.15	312.55	95.7
Nov. 1971	228.62	316.91	120.0
Dec. 1971	233.05	326.52	149.9
Jan. 1972	237.58	313.20	177.2
Feb. 1972	239.32	311.27	205.3
Mar. 1972	228.85	285.54	235.5
Apr. 1972	229.15	292.40	265.2
May 1972	224.75	285.81	270.7
June 1972	224.75	285.81	289.5
July 1972	226.62	283.02	312.6
Aug. 1972	215.89	271.24	341.2
Sep. 1972	212.97	256.48	369.0
Oct. 1972	214.71	275.08	398.6
Nov. 1972	207.65	273.31	423.6
Dec. 1972	204.41	269.96	443.9
Jan. 1973	213.72	268.68	460.4
Feb. 1973	218.48	278.98	475.7
Mar. 1973	200.50	263.66	487.2

Cycle II

Date	Power (W/cm) Average	Peak	EFPD
May 1973	177.57	228.88	6.5
June 1973	168.51	215.40	25.4
July 1973	182.95	240.56	58.8
Aug. 1973	184.36	244.34	87.6
Sep. 1973	178.75	231.38	116.6
Oct. 1973	172.68	227.11	145.2
Nov. 1973	174.61	219.11	166.2
Dec. 1973	175.73	222.25	192.6
Jan. 1974	176.65	222.55	221.2
Feb. 1974	176.03	221.01	247.8
Mar. 1974	175.50	219.20	278.1
Apr. 1974	175.50	219.17	307.2
May. 1974	175.07	211.72	311.8
TOTAL			799 EFPD

Average Burnup - 28 026 MWd/t
Peak Burnup - 31 363.9 MWd/t
Removed May 6, 1974

TABLE 6

BURNUP ANALYSES FOR FUEL RODS F-7[9] AND P-8[8] FROM H. B. ROBINSON UNIT 2

Rod	Identity	Distance from Rod Bottom (m)	Burnup (at.%)	Burnup (MWd/t)	Relative Gamma Scan Intensity (GSI)
F-7	a	0.006	1.26	12,100	21.0
	b	0.569	3.14	30,150	75.0
	c	0.930	3.12	29,950	78.9
P-8	a	0.305	2.56	24,570	43.0
	b	1.740	3.22	30,920	54.5
	c	2.019	2.88	27,620	48.5

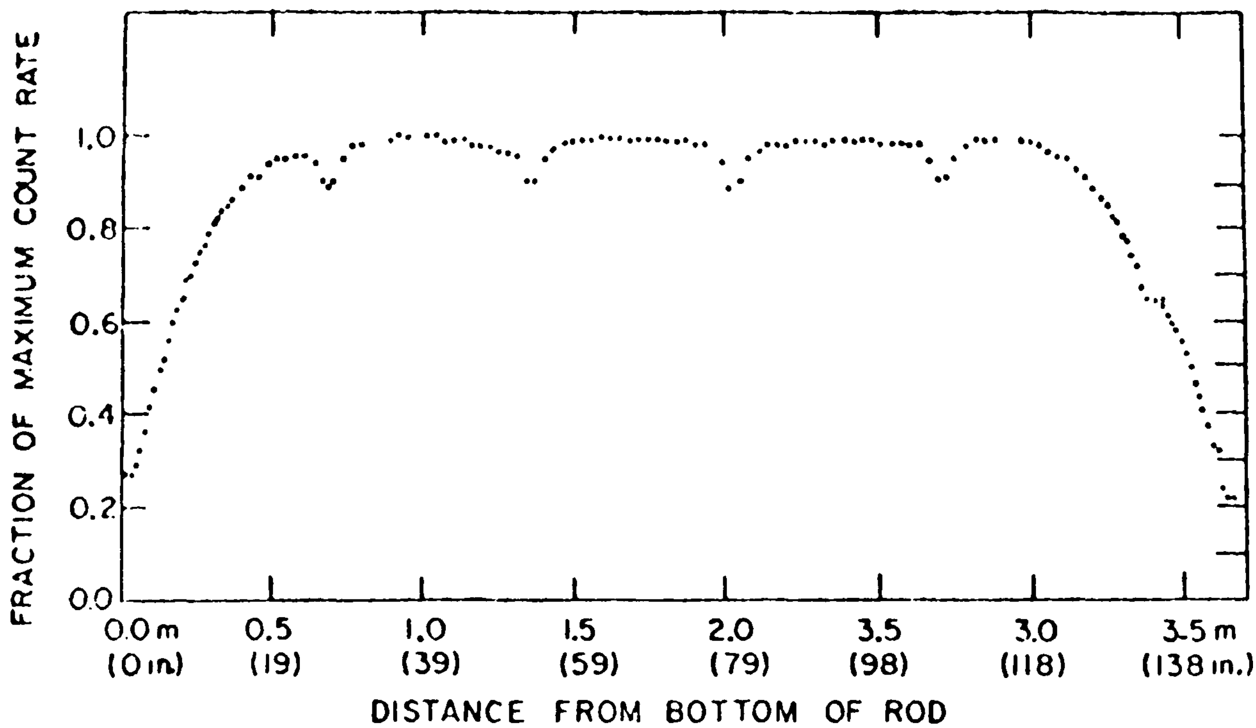


FIGURE 6. Axial Profile of Average Fuel Pellet Gamma Activites of Rod F-7 from H. B. Robinson Unit 2, Normalized from Gamma-Scan Gross-Count-Rate Data. (Neg MSD-62082, Ref 10)

A major use of the gross gamma scan is to determine the axial fluence profiles on a rod and the differences in the fluence across an assembly. The 15 rods gamma scanned at BCL had a peak-to-average gamma scan intensity ratio that ranged from 1.21 to 1.25.[3] This indicates that the profile from rod to rod was very similar. A typical profile for the gamma scan is shown in Figure 7.[3] The gamma scans are on file at BCL and were examined by WHC personnel. The average gamma intensity as a function of position in the assembly is shown in Figure 8.[8] Only BCL rods are included in this diagram due to calibration differences among gamma-scanning units. The average gamma intensity is fairly constant across the assembly with $\pm$5% variation.

2. Peach Bottom-II Assembly PH 462

The burnup of this bundle is not precisely defined. The reported average bundle is 11,900 MWd/t.[7] The reported average core burnup is 10,100 MWd/t.[11] The official GE average bundle burnup is 12,890 MWd/t. A companion assembly in the Cycle 1 (PH 006) had a burnup analysis made on a rod that had been gamma scanned. The peak burnup in this bundle was 13,000 MWd/t,[12] which is in close agreement with the GE value. A burnup analysis on a PH 462 rod is in order during DE.

The daily time-dependent average core power is available[11] in graphic form. The core power is given on a monthly basis in Table 7. The power peaked at ~240 W/cm and was within 5% of that power level except for a month at the start of the run and three months near the end of the run when it dropped as low as 120 W/cm. No data has been located on the fluence tilt in the reactor or calculations of power for this particular bundle.

The bundle was discharged from the reactor during the refueling outage between March 27 and June 24, 1976. The rod failures in the bundle were associated with control rod manipulations.[11]

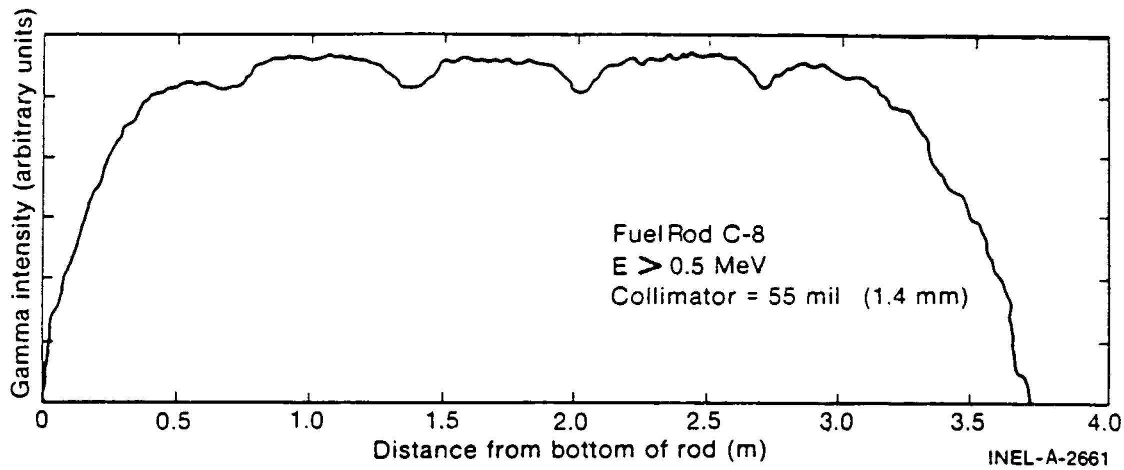


FIGURE 7. Gamma Scan of Fuel Rod C-8 from H. B. Robinson Unit 2 Assembly BO-5. (Ref 3)

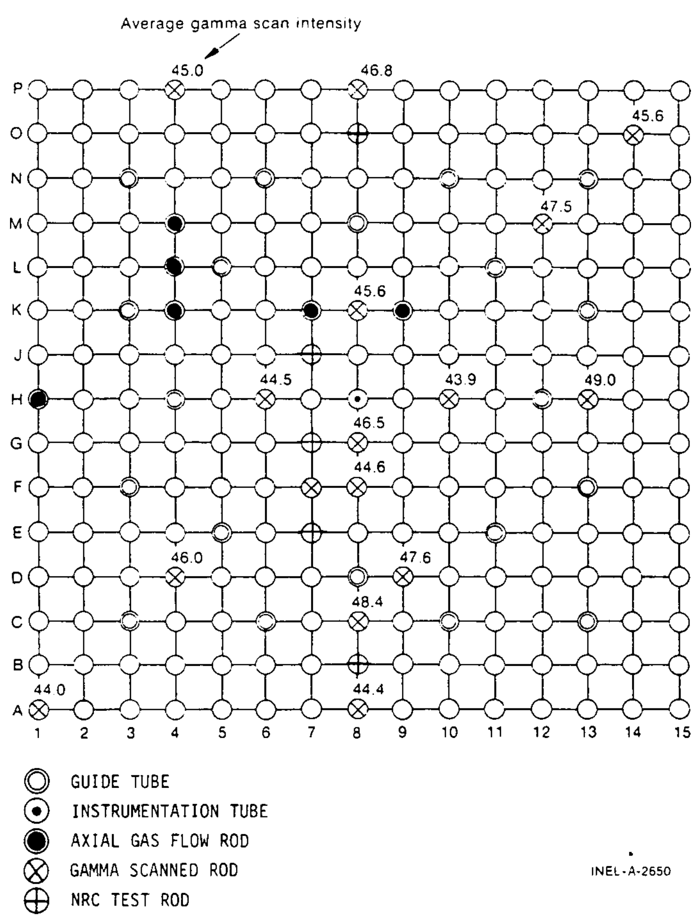


FIGURE 8. Diagram of Assembly BO-5 from H. B. Robinson Unit 2 with Average Rod Gamma Scan Intensity. (Ref 3)

TABLE 7

AVERAGE CORE HEAT RATING DURING RESIDENCE OF PEACH BOTTOM-II ASSEMBLY PH 462*

Date	Average Core Power (MWt)	Average Heat Rating (wt/cm)	Burnup (MWd/t)
Apr. 4, 1974	1835	134	230
Apr. 25, 1974	2603	190	390
May 12, 1974	2513	184	648
May 26, 1974	3164	231	741
June 19, 1974	3261	238	1010
July 15, 1974	3280	240	1585
Aug. 17, 1974	3292	240	2080
Sep. 10, 1974	3265	238	2555
Oct. 4, 1974	2856	209	2920
Nov. 21, 1974	3271	239	3542
Jan. 6, 1975	3280	240	4364
Feb. 3, 1975	3277	239	4697
Mar. 13, 1975	3293	241	5262
Apr. 2, 1975	3283	240	5640
Apr. 24, 1975	3215	235	6106
May 13, 1975	3172	232	6470
July 25, 1975	1649	120	7000
Aug. 16, 1975	1855	135	7300
Sep. 27, 1975	1882	137	7712
Oct. 31, 1975	1858	136	8100
Dec. 24, 1975	3285	240	8430
Jan. 15, 1975	3292	240	8766
Feb. 14, 1975	3255	238	9295
Mar. 26, 1975	3001	219	10100

*From Reference 11.

"During the refueling outage of March 27 to June 24, 277 fuel bundles were sipped using the out-of-core wet technique. Of these 277 bundles, 231 were of the improved 7 x 7 type. Of this group, 19 were determined to be leaking. Ten of these were given a detailed inspection of each fuel rod; 15 perforated (short tight cracks) rods were found. The primary mode of rod perforation was attributed to pellet-clad interaction. Visable evidence of minor secondary hydriding was observed on most rods.

Four leaker bundles of the unimproved 7 x 7 type were also found by sipping. A detailed examination of two of these bundles revealed 3 leaker rods. Two of the leaker rods (both in one bundle) sustained open long splits characteristic of the pellet-clad interaction mechanism. The third leaker rod (in the second bundle inspected) exhibited a typically hydrided appearance. The apparent severity of the pellet-cladding interaction cracks in the unimproved 7 x 7 fuel was significantly worse than that observed in the improved 7 x 7 fuel.

Most of the failures described above were apparently associated with control rod manipulations on January 13, which were not in accordance with the fuel vendor's fuel preconditioning recommendations."

C. TRANSPORTATION HISTORY

1. H. B. Robinson Unit 2 Assembly BO-5

During transportation of the H. B. Robinson Unit 2 assembly from the reactor to the hot cell and subsequent unloading of the cask, a series of events took place (see Table 8) that may have caused the assembly to overheat. As standard procedure, 25 gallons of water were drained from the transportation cask to allow for thermal expansion. As a result, the assembly was transported partially uncovered in a horizontal position. When the cask arrived at the hot cell, it was placed in a vertical position where it was also partially uncovered. After removal from the cask, the assembly was held in air before it was placed in the pool. When placed in the pool, large amounts of steam formed and an alarm sounded -- possibly due to the release of crud from the rod surfaces.[14] Later the assembly was held in a horizontal position for 7 hours during rod removal, but no steam was released when the assembly was put back into the pool.[14]

TABLE 8

TRANSPORTATION HISTORY FOR H. B. ROBINSON UNIT 2 FUEL ASSEMBLY BO-5

Configuration	Time or Event	Cladding Temp (°C)	Method of Temperature Determination
Storage in Pool	13 mo[13]	∿21 (Actual)	Measured
Horizontal Shipping in Half-Water-Filled Cask	<1 wk	343 (Maximum Possible) <300 (Maximum Possible)	Calculations Hardness Measurements
Vertical Hold in Half-Water-Filled Cask	6 h[8,13]	466 (Maximum Possible) <350 (Maximum Possible)	Calculations Hardness Measurements
Vertical Dry Hold in Cell	1-1/2 to 2 h(13,14)	466 (Maximum Possible) <350 (Maximum Possible)	Calculations Hardness Measurements
Transfer to Storage Pool	Steam & Alarm[14]	---	---
Storage Pool Residue	24 h[14]	∿21 (Actual)	Measured
Vertical Dry Hold in Cell	1 h[13]	100 to 150 (Maximum Possible)	Thermocouple Test[13]
Horizontal Dry Hold in Cell	6 h[13]	100 to 150 (Maximum Possible)	Thermocouple Test[13]
Transfer to Storage	Negligible Steam[14]	---	---

Two tests were conducted to determine the fuel rod cladding temperatures since calculations indicated that it might have been as high as 466°C:

1) EG&G inserted a thermocouple into the assembly in a horizontal position to measure the temperature.(13) When correcting for possible air films, temperatures were believed to be ~102°C but in no case >148°C.

2) BCL made a series of cladding microhardness measurements(15) at five different axial locations on six different rods. The locations of the measurements are shown in Figure 9. Since one side was uncovered in the horizontal transportation configuration and the top of the assembly was uncovered during vertical hold, the series of measurements ensured that all regions of the assembly would be tested. There is no statistically significant variation in the measurement results among the rods or along the length of the rods.(15) The hardness was 250 ± 10 kHN.* For comparison, Turkey Point fuel, also made by Westinghouse [operated to almost the same burnup (28.5 GWd/MTM) at nearly the same average power level (182 W/cm), and discharged from the reactor at nearly the same time (November 1975)] had a cladding hardness of 274 ± 30 kHN as-irradiated and 190 ± 15 when annealed at 482°C.(16,17) Hence, it is concluded that little, if any, annealing of the irradiation hardening took place in the Assembly BO-5 fuel rod cladding.

The correlation of time-at-temperature and irradiation recovery of Zircaloy is not well established but the best work has been done by Kemper and Zimmerman.(18) Using the data of Kemper and Zimmerman for the 7-hour vertical hold, the temperature would have had to be above 350°C to have 15% recovery. During horizontal transportation, the temperature would have had to be above 300°C to get 20% recovery. Since the BCL hardness measurements indicated that the cladding probably did not exceed the recovery levels, the corresponding temperatures of 350°C and 300°C can be regarded as upper temperatures for the respective time periods.

---

*Knoop hardness.

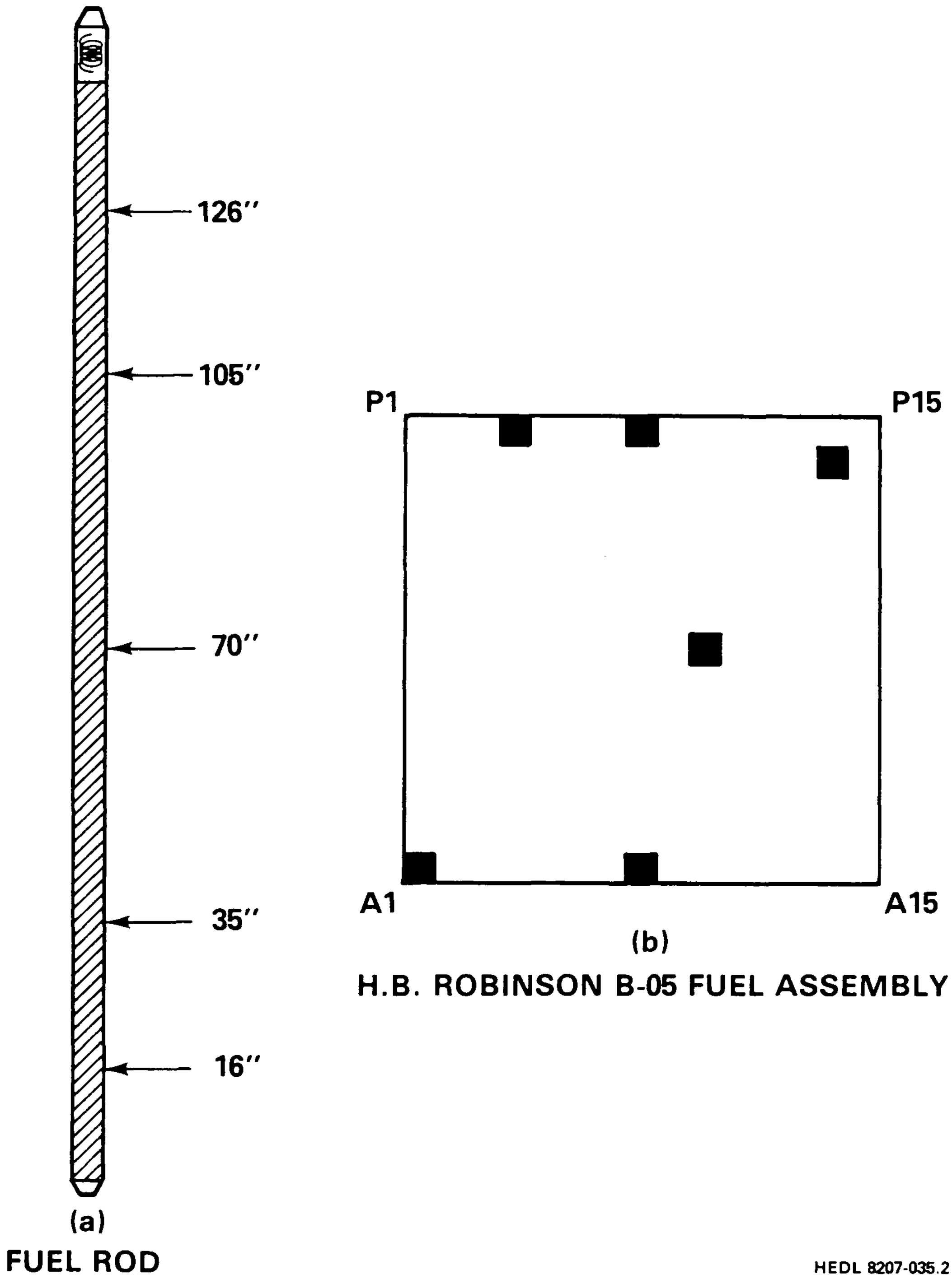


FIGURE 9. BCL Annealing Studies on As-Irradiated Fuel from H. B. Robinson Unit 2 Assembly BO-5.
a) Axial location of hardness measurements.
b) Planar location of rods that provided hardness specimens.

Although the Assembly BO-5 may have experienced a temperature excursion during transportation, it was not long enough nor high enough to change the condition of the fuel rod significantly.

2. Peach Bottom-II Assembly PH 462

In April 1977, the fuel was shipped from Peach Bottom-II to EG&G in a National Lead Industries dry shipment cask.[7] The cask was designed for dry shipment, and no incidents have been reported with the shipment.

IV. H. B. ROBINSON UNIT 2 AND PEACH BOTTOM-II CHARACTERIZATION

Normal pretest procedure is to nondestructively characterize those rods actually being tested, and destructively characterize companion rods from the same assembly. For the present test, as much existing data as possible are used to establish the test rod condition without actual test rod characterization. To do this, the data in this section are in two parts: NDE and DE.

An overall view of the characterization of fuel rods and the relationship between characterized rods and actual test rods with the assemblies is given in Figures 10 and 11. It is obvious that the H. B. Robinson Unit 2 fuel is characterized much more extensively than the Peach Bottom-II fuel . In fact, the H. B. Robinson Unit 2 fuel is characterized as well as any assembly that has been examined. Whether this degree of characterization is suitable for our purposes or additional characterization is necessary will be discussed in Section V, which examines the condition of the test rods based on the available NDE and DE.

A. NONDESTRUCTIVE EVALUATION

NDE is conducted much more often than DE for a number of reasons: 1) it is significantly less expensive, 2) many times it can be conducted at poolside, and 3) it can be used to detect reactor breaches and other rod changes (such as bowing, growth, and creepdown), which affect reactor operations. In the reactor industry, NDE usually refers to eddy current (EC) and/or ultrasonic examination; sometimes visual, profilometry, and gamma scanning are also included. In a hot cell, complete NDE includes all the above tests.

The uses of the various NDE techniques are listed in Table 9. Visual examination is quite subjective but with proper lighting, meaningful observations can be made. Many times the observations can be misleading. Only large cracks can be observed. Pinhole breaches are rarely discovered visually.

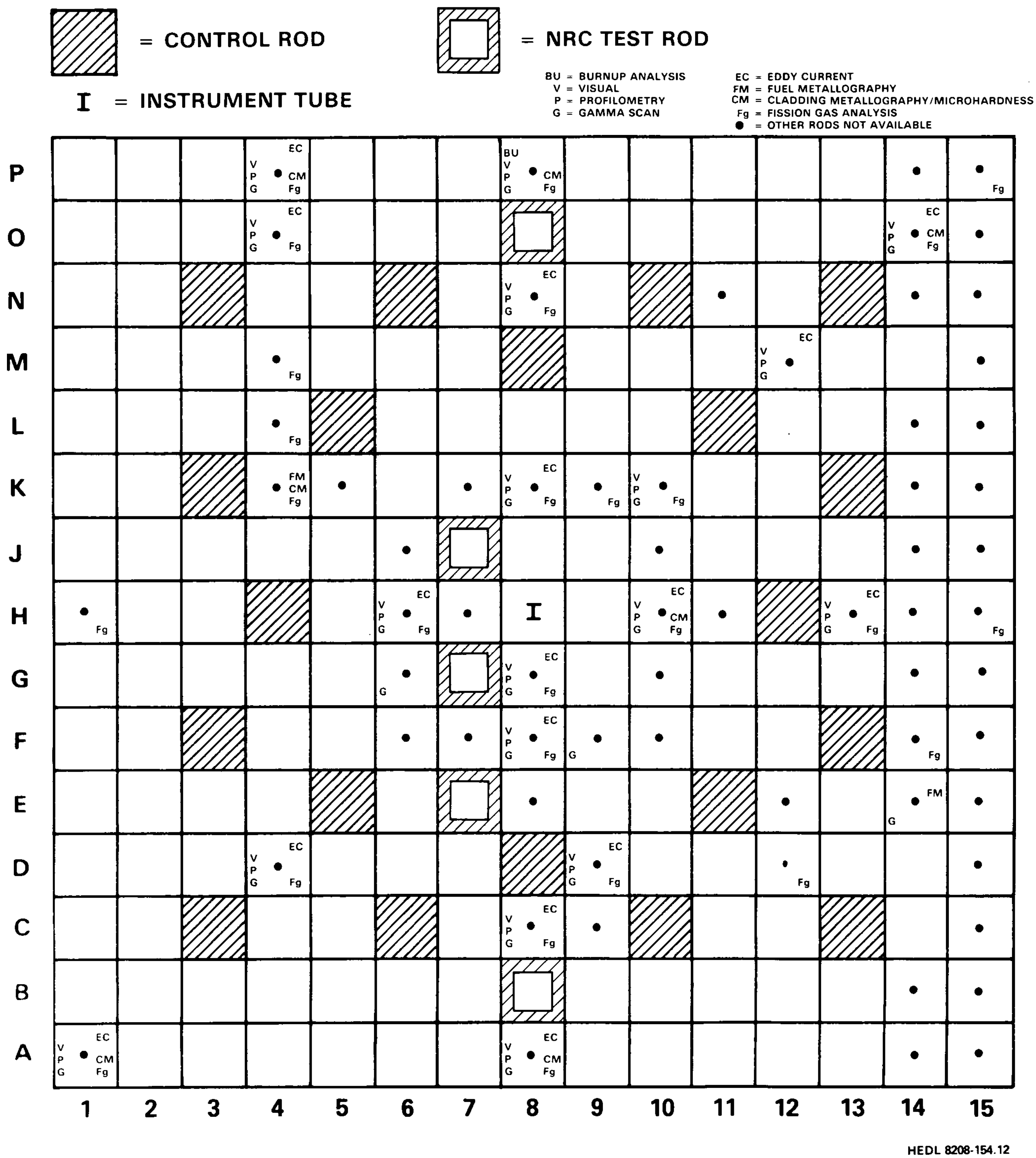


FIGURE 10. Rod Characterization of H. B. Robinson Unit 2 Assembly BO-5.

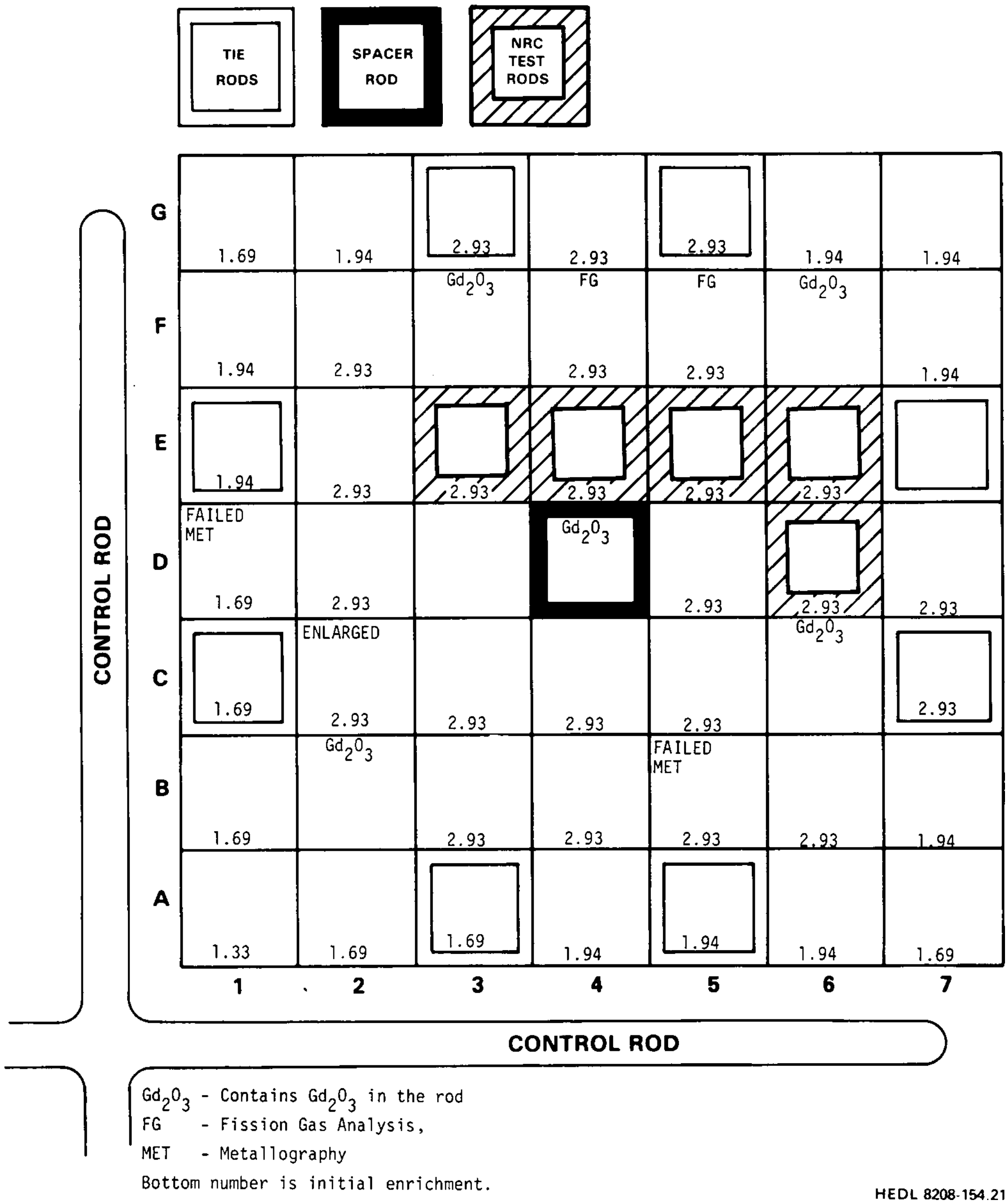


FIGURE 11. Rod Characterization of Peach Bottom-II Assembly PH 462.

TABLE 9

USES OF NONDESTRUCTIVE EXAMINATION TECHNIQUES

Visual	Eddy Current
Gross Cracks	Large Incipient Cracks[21]
Crud	Breaches on Water-Inundated Rods[21]
General Rod Condition	Gross Pellet Cladding Interaction
Rod Length	Oxide Thickness with Specialized EC[20]

Profilometry	Gamma Scan
Cladding Dilation	Fluence Profile
Pellet-to-Cladding Mechanical Interaction	Gross Pellet Condition
	Fuel Column Height
	Fuel Pellet Gaps
	Crud Profile with Some Scanners

Profilometry is used to determine dimensional stability during reactor residence. It can also be used to infer stress state and stress level if only prepressurization is known. While the best accuracy is usually quoted at 0.2 mil,[19] when cladding ovality, rod positioning, and the difference between pre- and post-test measurements are considered, measurement of cladding strain by the use of profilometry requires a strain increment $>3.3 \times 10^{-3}$ to be meaningful (based on 0.2-mil accuracy and a 0.5-mil positioning and ovality uncertainty). This is a gross change.

While attempts have been made to use the EC technique to look at small cracks, the results are misleading and unreliable. In a recent study on incipient crack detection,[21] the accuracy of the EC technique was only ∿50%, even with through-the-wall cracks. The technique is enhanced when the breach is contaminated with water. Unfortunately, the technique also gives indications from pellet-cladding interaction, large ovality, and other anomolies that confuse interpretation. Efforts have been successful in specialized situations to determine oxide thickness using the EC technique.[20]

The accuracy of the gamma scan is dependent on the particular feature being measured. Fluence profiles, stack height, and pellet-to-pellet gaps can be measured quite accurately since they are positioned features requiring only a change in the profile at a particular level. The pellet condition can be determined in a gross manner. Distinct pellet to pellet interfaces can be observed on gamma scans. Any change, such as oxidation in a breached rod, that produces a gross distortion of the interface and hence a change in the shape of the profile will be detected. The actual gamma count rate is dependent on the combination of the beam and positioning of the rods. Therefore, this method is not good for small changes in pellet condition since it requires comparison of traces taken at two different times. Small pellet condition changes that only change the count rate but not the shape of the profile will be difficult or impossible to detect. Special systems with multichannel analyzers can be used to determine crud behavior since they can be tuned to the crud spectrum. EG&G is the only location with such a system; most hot cells have single channel analyzers.

NDE is good for establishing the condition of a rod and looking for gross changes. Small changes are not accurately detected and should be followed with DE techniques that are more accurate.

1. Visual Cracks and Crud

H. B. Robinson Unit 2 Assembly BO-5 -- Visual examinations were conducted on the rods shown in Figure 12.[15-22] Still photographs of all rods are available at BCL hot cells and were inspected by WHC personnel. No video tapes are available. Below ~130 in., only light axial scratches were seen during the dismantling process. The crud was very light and almost unnoticeable. Above 130 in., there was a heavy crud deposit on all rods similar to that shown in Figure 13. Little variation was observed from rod to rod.

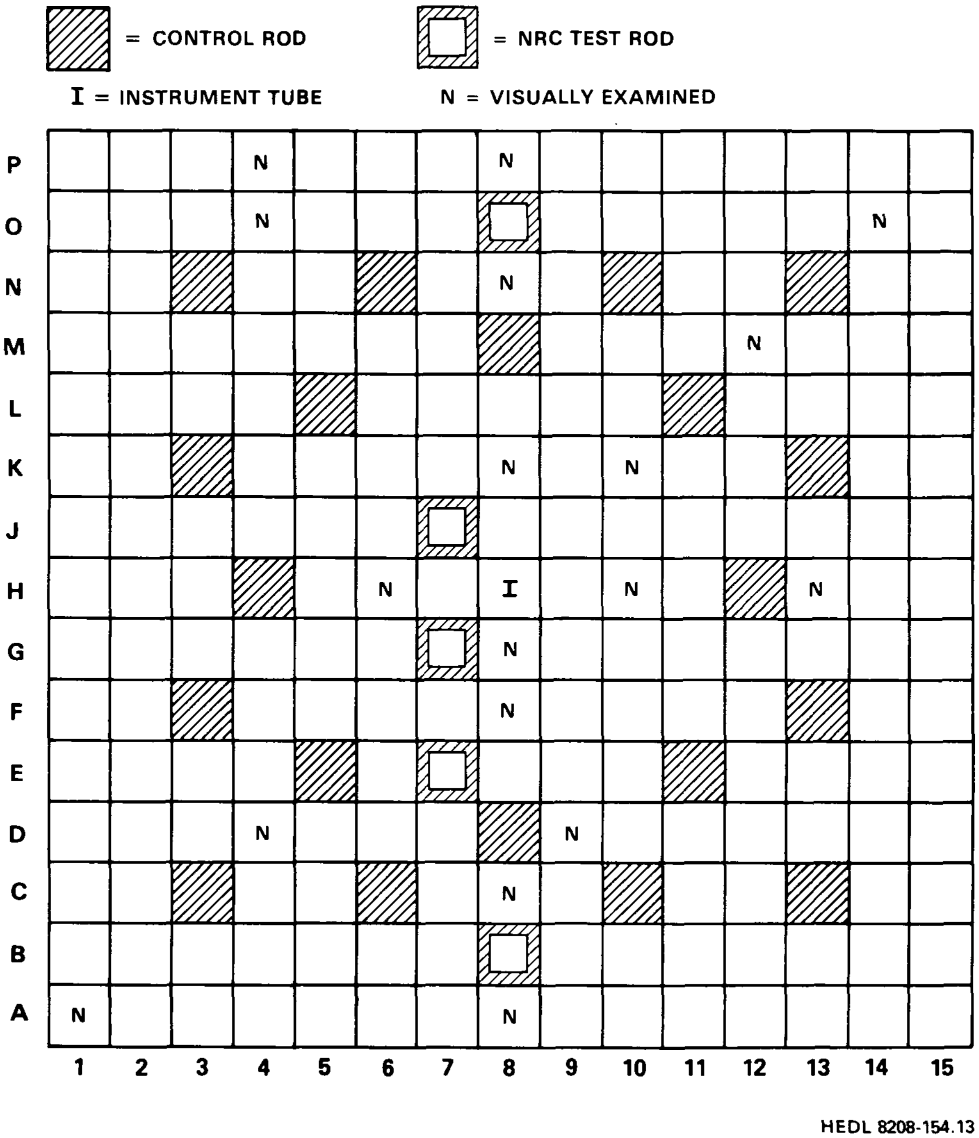
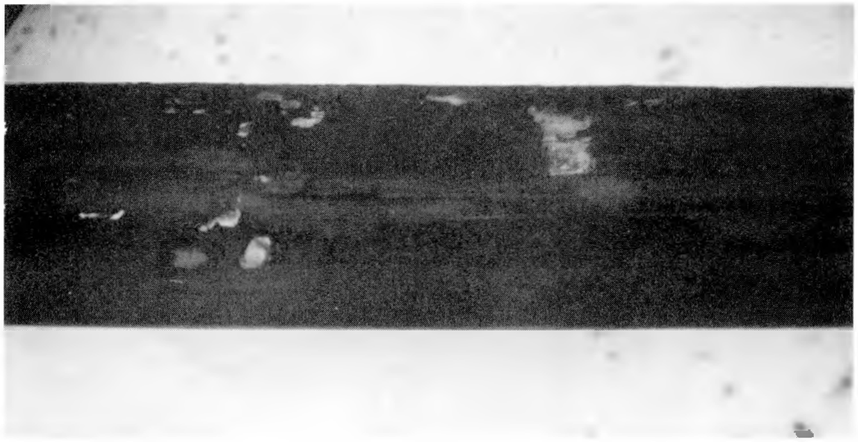
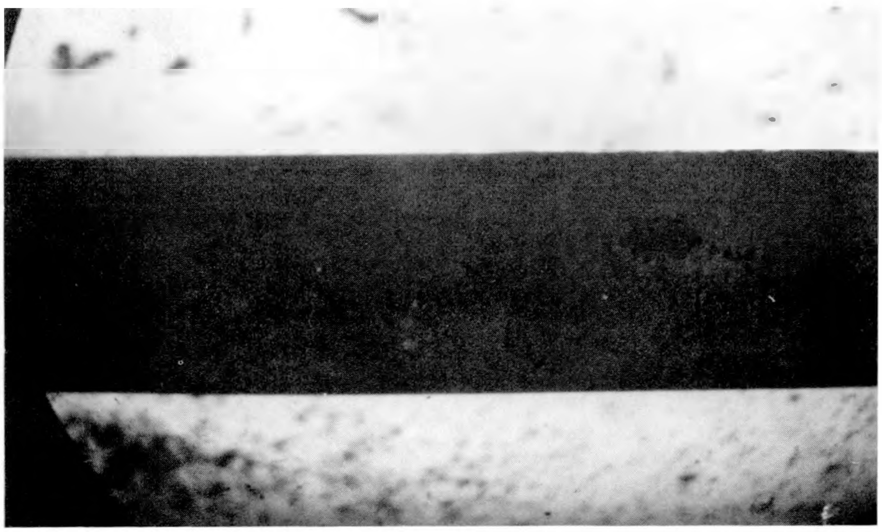


FIGURE 12. Visual Examination of H. B. Robinson Unit 2 Assembly BO-5.



(a) Rod H-10 at 133¼" Elevation and 0° Orientation.



(b) Rod A-1 at 133¼" Elevation and 90° Orientation.

FIGURE 13. Typical Crud Patterns on the Upper 1/4 of Rods from H. B. Robinson Unit Assembly BO-5.

Peach Bottom-II Assembly PH 462 -- Rods D-1 and B-5 have been visually examined by periscope at EG&G-Idaho.[7] Other than the fact that the examined rods were breached, there were no unusual findings. Some fine cracks were evident along with light oxide layers. The photographs are on record at EG&G. Since these rods were from the other side of the assembly from the test rods, visual examination of the test rods themselves will be needed.

2. Profilometry

H. B. Robinson Unit 2 Assembly BO-5 -- Eighteen rods throughout the assembly were profilometered at BCL.[8,22,23] (See Figure 14.) The main findings were the diameters, ovality, and location of maximum ovality. No actual traces are in this report since they are of little use in a reduced form. The original traces are on file at BCL and have been examined by WHC personnel. The diameter at selected locations is plotted in Figure 15 for all eighteen rods. All the traces can be bounded by a curve of $D \pm 1$ mil where D is the average axially dependent diameter. There is no reason to believe that the test rods are not within the bound. Similarly, the maximum ovality and position of maximum ovality is tightly bound as shown in Figure 14. No indications of ridging were found in the original traces.

Peach Bottom-II Assembly PH 462 -- Profilometry was conducted on rods B5 and D1 but no traces have been located at EG&G.

3. Gamma Scanning

H. B. Robinson Unit 2 Assembly BO-5 -- Axial gamma scans were run on 22 rods or parts of rods as shown in Figure 16. The scans at BCL[8,22] were total gamma which give a gross count. The scans on rod sections at ANL and Los Alamos National Laboratory (LANL) were for gross counts as well as specific isotopes; ^{137}Cs and ^{106}Rh at ANL[10,24] and ^{106}Rh, ^{134}Cs, ^{137}Cs and ^{154}Eu at LANL.[25] The gamma scans from BCL and ANL are available at those laboratories.

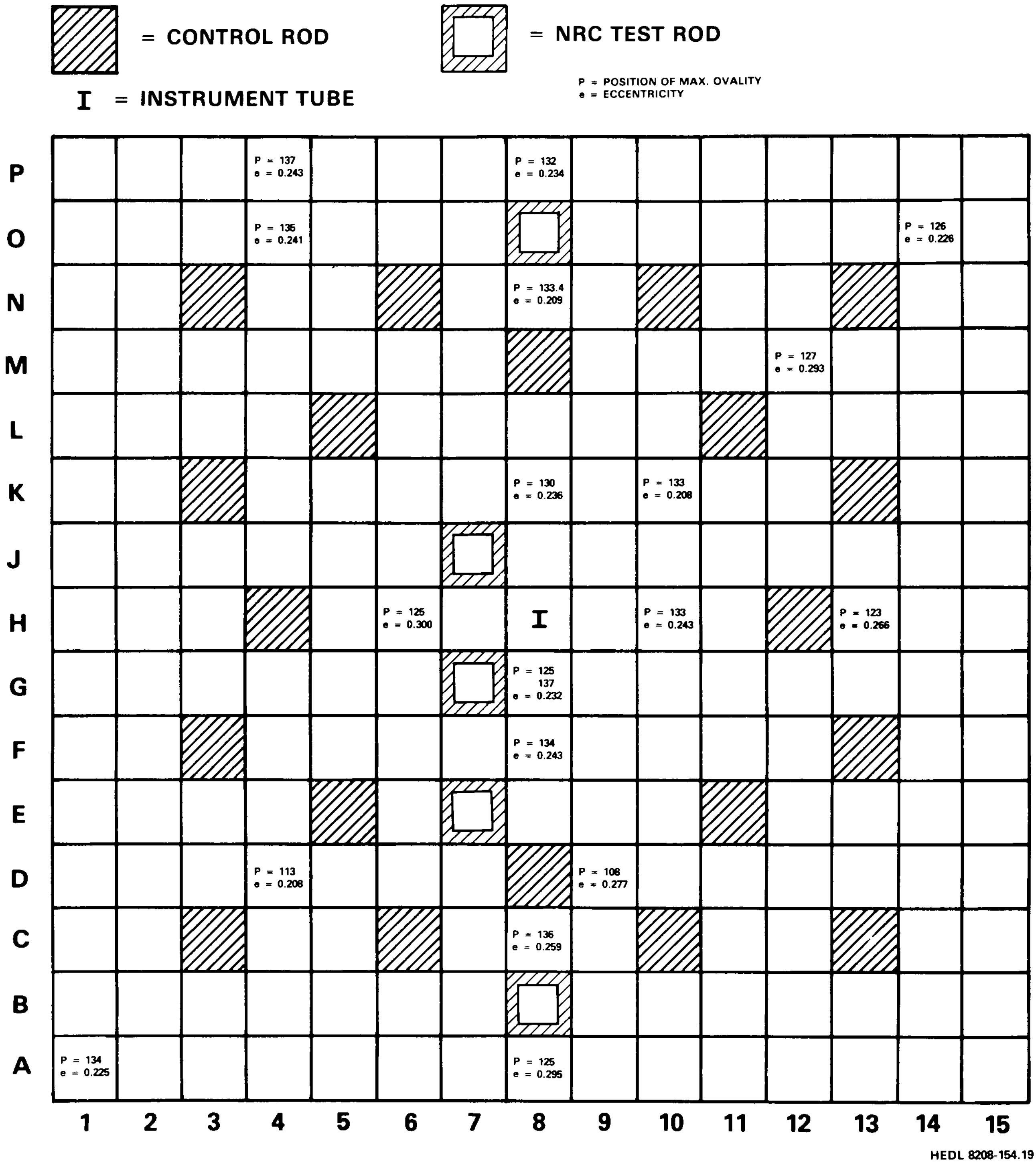
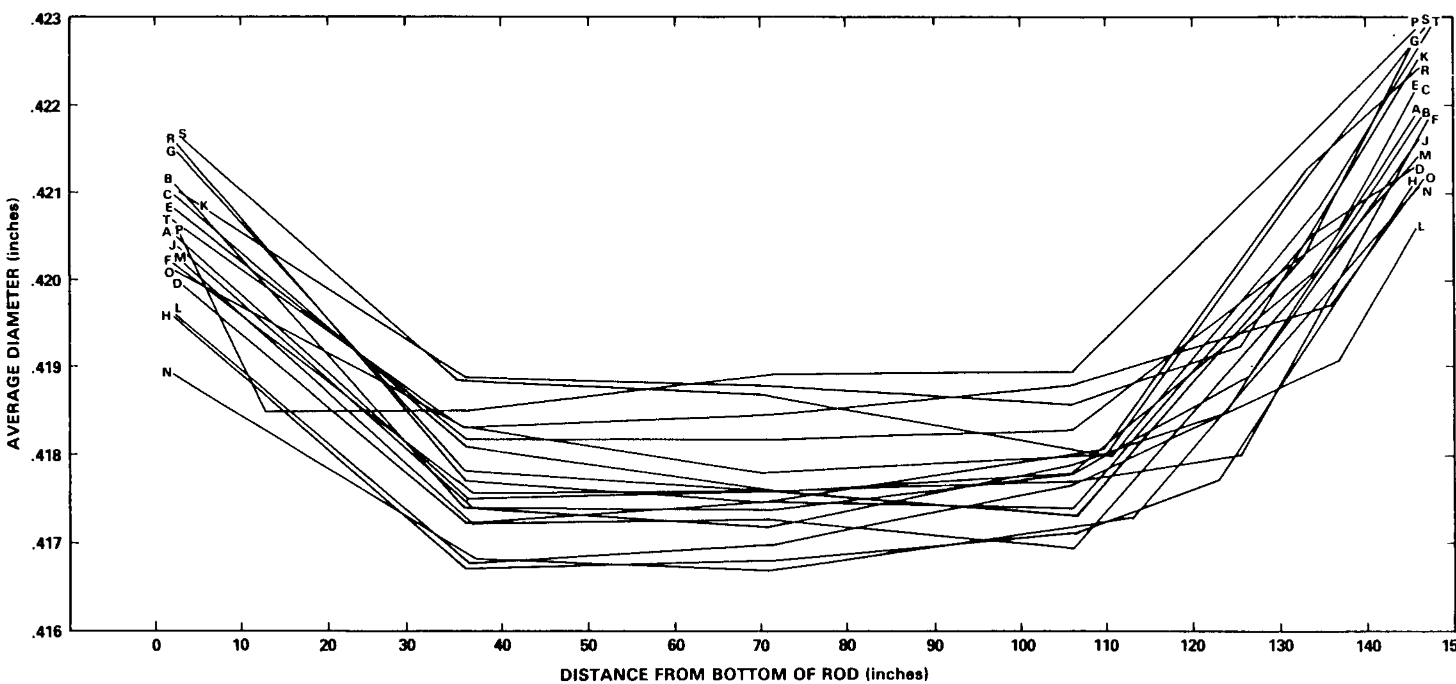


FIGURE 14. Profilometry Measurements from H. B. Robinson Unit 2 Assembly BO-5.



ROD	DESIGNATE	REF
A-1	A	8
A-8	B	8
D-9	C	8
F-8	D	8
G-8	E	8
H-6	F	8
H-10	G	8
H-13	H	8
M-12	J	8
O-14	K	8
P-4	L	8
P-8	M	8
D-4	N	23
C-8	O	23
K-8	P	23
K-10	R	22
N-8	S	22
O-4	T	22

FIGURE 15. Spiral Profilometry of Rods from H. B. Robinson Unit 2 Assembly BO-5 (no indication of ridging).

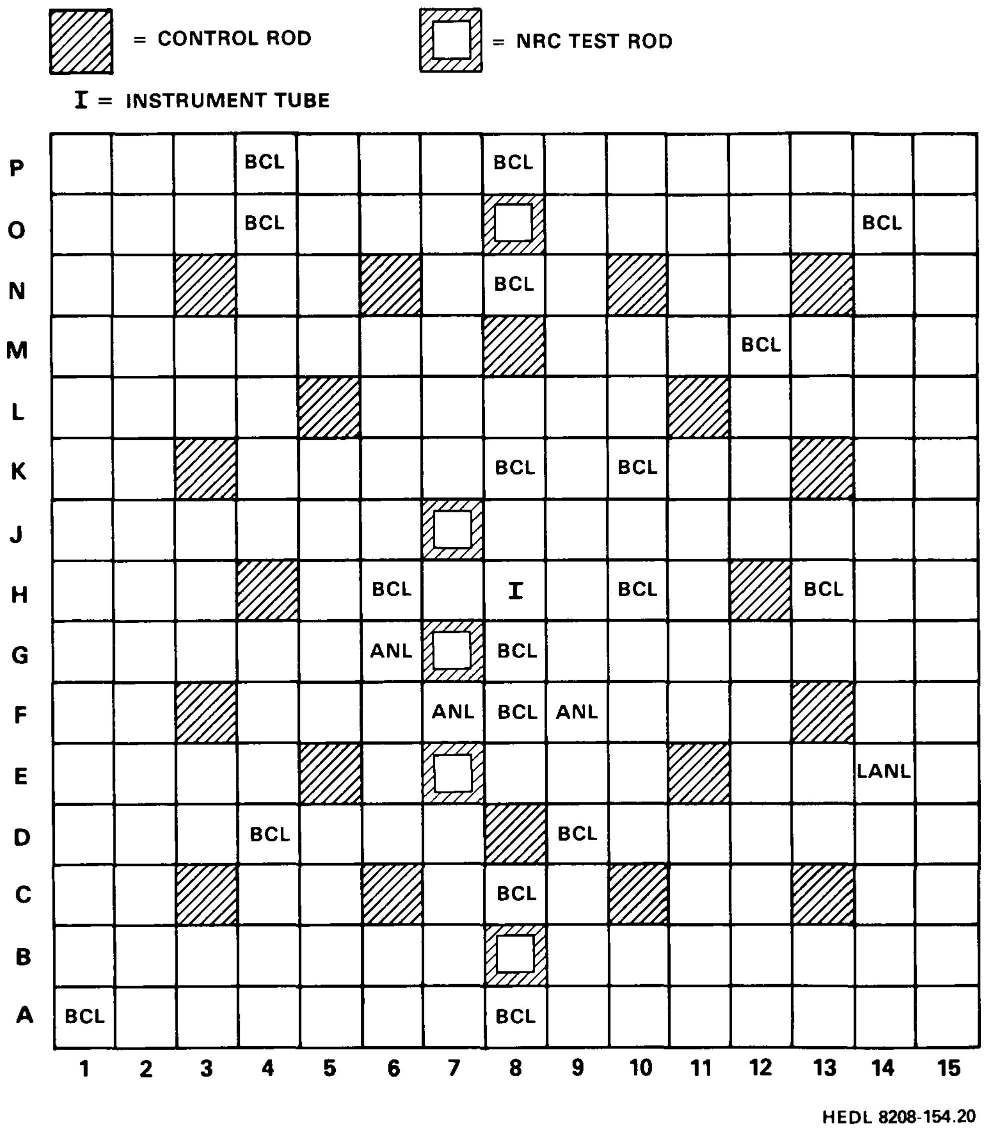


FIGURE 16. Rods from H. B. Robinson Unit 2 Assembly BO-5 Gamma Scanned at Various Laboratories.

As can be seen from Table 10, the stack height is very uniform at 143.5 ± 0.4 in. and contains on the average 6 ± 3 large pellet-to-pellet gaps. Discounting the dips in the traces where the structural members changed the fluence (see Figure 6), the profiles were very flat as indicated by the burnup measurements (Table 6). Gaps between pellets were also found on the ANL[10,24] and LANL[25] gamma scans, although the separations in the LANL gamma scans are probably not meaningful due to the small size of the segments (∿2-1/2 in.). The ANL gamma scans also showed a very flat profile as indicated in Figure 6.

TABLE 10

GAMMA SCAN RESULTS OF FUEL RODS* FROM H. B. ROBINSON UNIT 2 ASSEMBLY BO-5

Rod	Fuel Height	Max./Min. Intensity**	No. Pellet/Pellet Gaps >30 mil
A-1	143.63	1.02	8
A-8	143.4	1.06	8
C-8	143.7	1.04	6
D-4	144.1	1.02	8
D-9	142.9	1.02	5
F-8	143.9	1.02	7
G-8	143.6	1.02	3
H-10	143.5	1.04	3
H-13	142.7	1.05	4
K-8	143.1	1.02	2
M-12	143.8	1.02	11
0-14	143.1	1.02	8
P-4	143.0	1.02	8
P-8	144.1	1.00	4
K-10	143.1	1.04	4
N-8	143.4	1.02	4
0-4	143.6	1.04	5
H-6	143.7	1.02	9

*From References 8 and 22.
**Over central 20 to 115-in. length of rod measured from bottom.

One of the ANL gamma scans[10] is shown in Figure 17. Distinct pellet-to-pellet interfaces are seen in the gross traces. The irregularity is due to pellet fragmentation. Similar features are found in the isotopic traces but with a much higher noise ratio due to the lower count rates.

Peach Bottom-II Assembly PH 462 -- No gamma scans have been conducted on rods from this assembly.

4. Eddy Current Examination

H. B. Robinson Unit 2 Assembly BO-5 -- Eighteen rods shown in Figure 18 were EC scanned at BCL.[15,22] Indications were found on four rods: P-4, O-4, N-8 and K-10. The traces are available at BCL and have been examined by WHC personnel. Only Rod O-4 had a completely uncorrelated indication, and only Rod P-4 was thought to exhibit any type of defect.[15] Detailed EC analysis in Reference 15 states:

> *"Only one fuel rod (P-4) provided an eddy current signal response that could be considered to be caused by a defect. The eddy current scan on this rod indicated a relatively long anomaly located in the fuel cladding about 213 cm from the bottom of the rod. The anomaly appeared to cover about 7.6 cm along the length of the rod in this area. This relatively gradual signal indication was not observed on fuel rods very frequently and was much different from the abrupt indication caused by pits, holes, and internal hydride cracking in the cladding between the fuel pellets. However, the relatively strong signal voltage in the horizontal output suggested that the defect was on the ID of the cladding.*
>
> *Rod K-10 exhibited an eddy current indication at 80 cm from the rod bottom that correlated well with a gap in the fuel column, as indicated by the gamma scan. Rod N-8 exhibited eddy current indications at 61 and 79 cm that also corresponded to fuel column gaps detected in the gamma scan. Rod O-4 exhibited an eddy current indication at 79 cm, although no correlation with gamma scan or profilometry data was possible. This may indicate a defect region.*
>
> *All three rods showed eddy current indications in the region of 335 to 340 cm. These indications are believed to be caused by the increase in rod diameter and oxide or crud deposits in this region."*

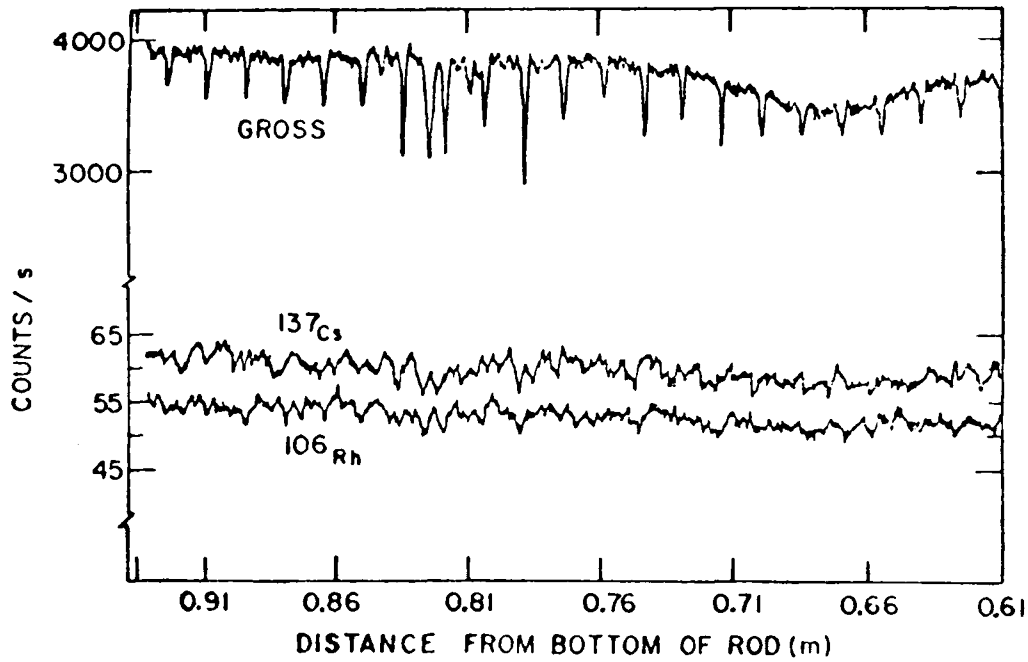


FIGURE 17. Comparison of Gross and Isotopic Activity Traces for Part of Rod F-7 from H. B. Robinson Unit 2 Assembly BO-5. (Ref 10)

Peach Bottom-II Assembly PH 462 -- All the rods from the assembly were EC examined at the Peach Bottom-II pool.[7] All rods had no EC-indicated defects except for two breached rods and one "possible" breached rod (see Figure 11). A letter detailing the methods, standards, and results of this unpublished work is provided in the Appendix.

B. DESTRUCTIVE EXAMINATION

DE is most useful when looking for small changes in properties, verifying or determining reactor operating conditions, such as burnup, or obtaining input for mechanistic modeling codes. To measure changes, companion rods are usually destructively examined prior to testing. The actual test rods are examined after testing is complete. By its very nature, DE can only be done once; therefore, if more than an average rate for a phenomenological occurrence is desired, multiple duplicate experiments must be run. The advantage is the sensitivity to materials conditions revealed by DE techniques as compared to NDE.

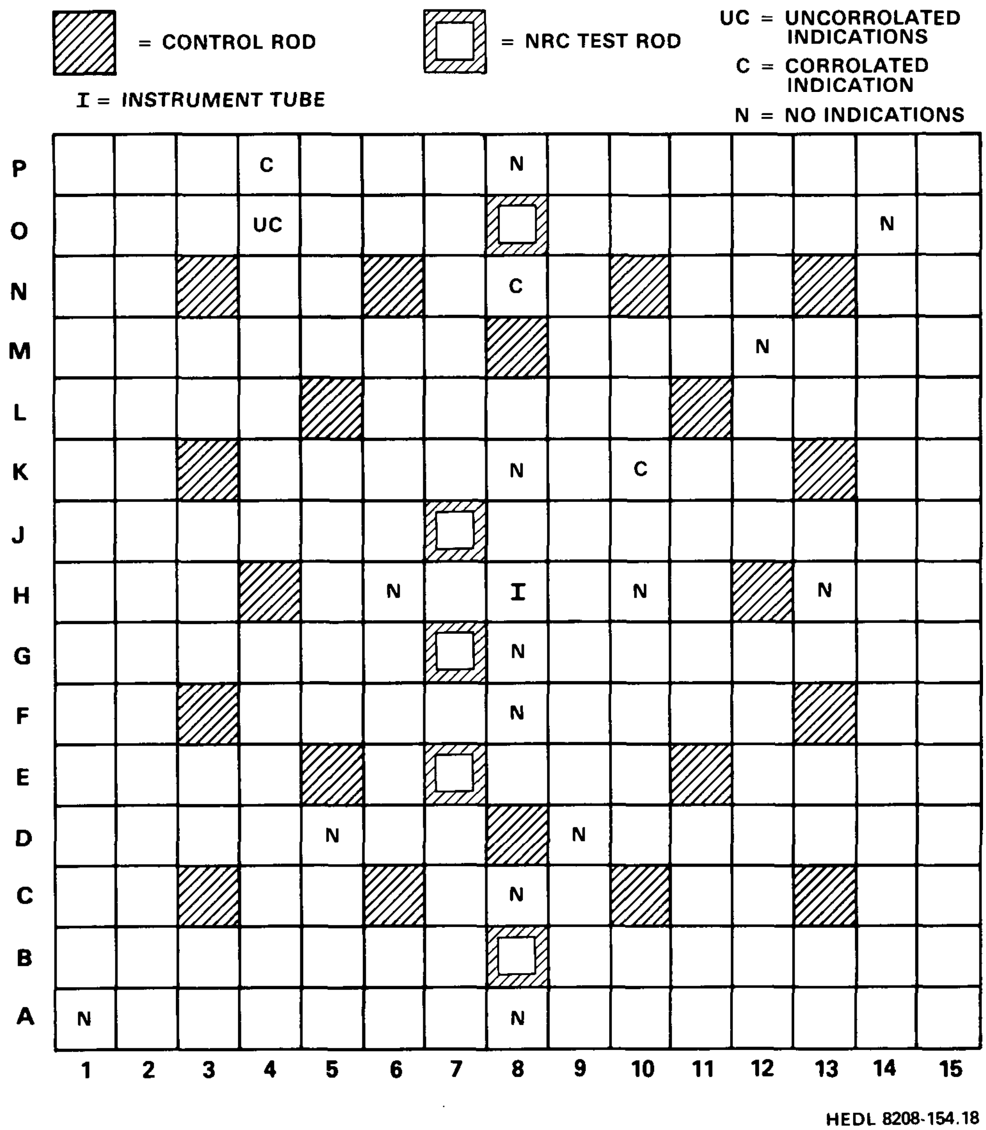


FIGURE 18. Eddy Current Examination of H. B. Robinson Unit 2 Assembly BO-5. (Refs 15 and 22)

The data obtained from a DE usually falls into one of three categories: 1) properties related to the internal gas pressure, 2) cladding properties, and 3) fuel properties. For the purpose of this report, fuel cladding interaction will be listed under fuel properties. The various types of data that may be obtained from the destructive examination are given in Table 11. Usually any particular examination does not obtain all the data listed in the table.

The main methods for obtaining the destructive examination data are: 1) rod puncturing and backfilling, 2) gas analysis, 3) metallography (etched and polished), 4) microhardness, 5) electro-optical (SEM, TEM, Auger, Ion Microprobe, etc.), 6) tensile, burst and stress rupture tests, etc., 7) fission gas analysis, and 8) radiochemical analysis.

TABLE 11

POSSIBLE DATA FROM DESTRUCTIVE EXAMINATION

Gas Analysis	Cladding Properties	Fuel Properties
Internal Void Volume	Hardness	Pellet Cracking
Internal Gas Pressure	Hydride Volume and Orientation	Fuel Porosity
Gas Volume	Oxide Thickness	Grain Size
Gas Composition	Cladding Thickness	Chemical Composition
Fission Gas Release	Crud Thickness	Fuel-Pellet Gap Size
Fission Gas Isotope Composition	Grain Size	Fuel-Cladding Chemical Interaction (FCCI)
	Cracks	Fuel-Cladding Mechanical Interaction (FCMI)
	Mechanical Properties	Fission Product Deposition
		Fission Product Chemical State

1. Gas Analysis

H. B. Robinson Unit 2 Assembly BO-5 -- Gas analysis was conducted on 28 rods at BCL,[8,22,23] EG&G,[3] and LANL.[26,27] The void and gas volumes, gas pressure, and fission gas release are given in Table 12. The average void volume of 24.7 ± 1.5 cc in the H. B. Robinson Unit 2 rods agrees well within a standard deviation with the void volume of 22.8 ± 1.2 cc measured for the Turkey Point fuel.[16] The internal pressure (220 ± 14 psia) for the H. B. Robinson Unit 2 rods is about 70% lower than the internal pressure (373 ± 25 psia) in the Turkey Point rods. This is due to a lower pre-irradiation gas fill pressure (528 ± 20 cc for Turkey Point vs 370 ± 10 cc for H. B. Robinson Unit 2). The fission gas release for both types of fuel were within one standard deviation (0.27 ± 0.08 for Turkey Point vs 0.21 ± 0.05 H. B. Robinson Unit 2).[17] Only Rod H-6, has gas volume and hence gas pressure significantly lower than a standard deviation for the average. No indication is given in the reference as to whether there was an apparatus problem with this measurement. The fission gas release, internal pressure, void and gas volumes, are shown in Figures 19 through 22 as a function of position in the assembly. There is no systematic variation within the assembly indicating that average values should be suitable as initial pretest conditions for the present test.

Table 13 gives the molecular gas analysis and both krypton and xenon isotopic compositions. Once again, there is very little variation among the 13 rods analyzed. These analyses also compare favorably with the Turkey Point gas analyses.[16]

Peach Bottom-II Assembly PH 462 -- Gas analysis was made on two rods, F-4 and F-5, from Assembly PH 462. The results from V. Storhok at EG&G are presented in Table 14. These two Peach Bottom-II rods have very similar gas analysis results. The void volume is higher than in the H. B. Robinson Unit 2 fuel due to the larger dimensions of the BWR fuel. The gas volume and pressure are lower since the Peach Bottom-II fuel was not pressurized prior to irradiation. The high fission gas release of 2.94% at this fairly

TABLE 12

GAS ANALYSIS OF RODS FROM H. B. ROBINSON UNIT 2 ASSEMBLY BO-5

Rod	Void Volume (cc)	Gas Volume (cc at STP)	Gas Pressure (psia)	Fission Gas Release (%)	Ref
A-1	26.1	386	209	--	8
H-1	26.2	376	220	0.20	3
D-4	27.2	370	200	--	23
K-4	20.6	366	261	0.18	3
L-4	24.5	380	228	0.28	3
M-4	22.7	368	238	0.18	3
O-4	24.5	367	221	0.19	22
P-4	24.2	378	222	--	8
H-6	26.3	160	86.5	--	8
K-7	23.7	374	232	0.17	3
A-8	25.0	358	203	--	8
C-8	24.1	379	228	--	23
F-8	22.7	355	221	--	8
G-8	23.5	372	224	--	8
K-8	25.3	381	220	--	23
N-8	24.3	363	219	0.25	22
P-8	24.5	387	224	--	8
D-9	25.0	354	201	--	8
K-9	23.3	375	236	0.14	3
H-10	24.7	355	201	--	8
K-10	25.4	360	209	0.22	22
D-12	26.5	---	---	0.26	26,27
H-13	25.0	361	204	--	8
F-14	25.8	361	206	0.17	27
O-14	24.6	373	215	--	8
H-15	27.4	---	---	0.30	26,27
P-15	24.0	385	236	0.21	27
M-12	--	377	211	--	8
Avg	24.7	370	220	0.21	
σ	1.5	10	14.5	0.05	

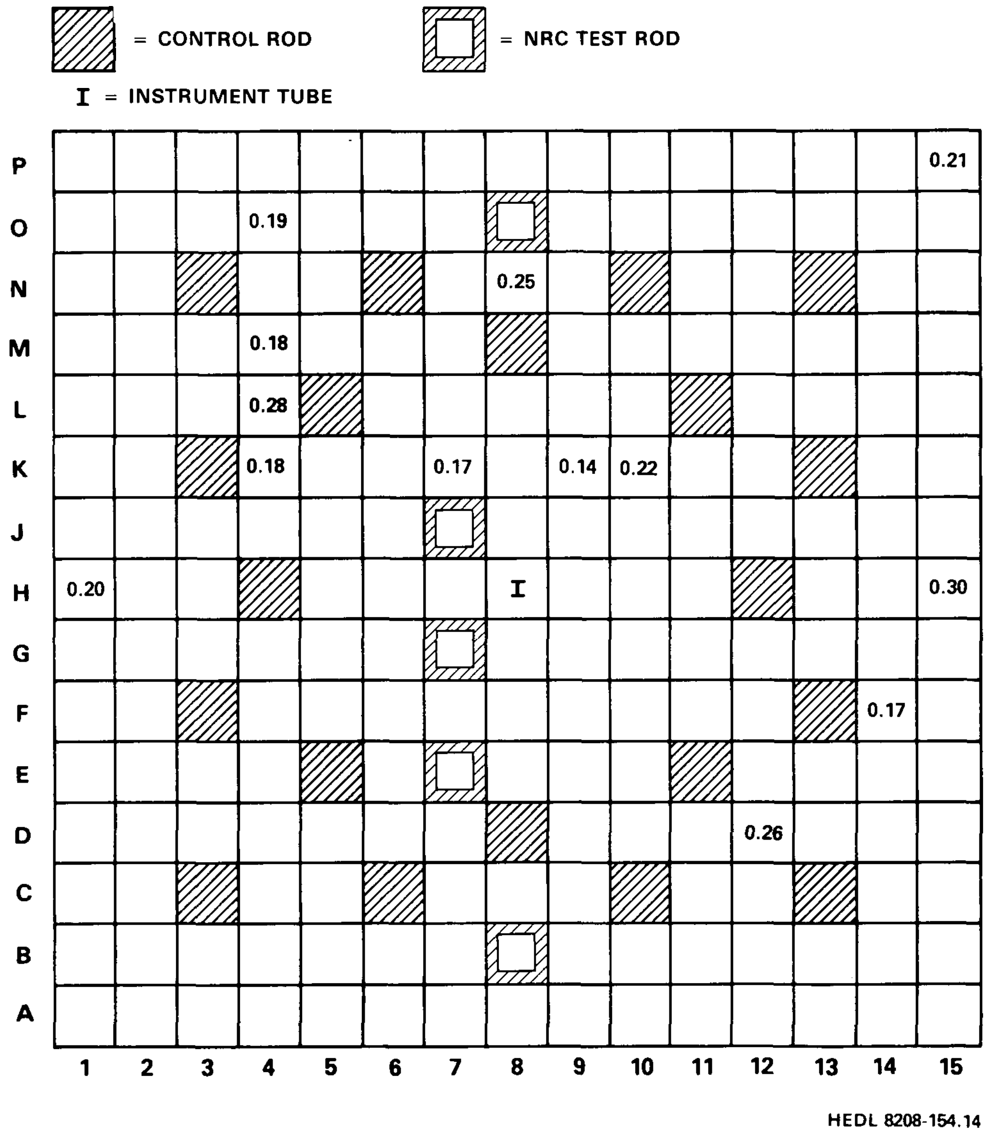


FIGURE 19. Fission Gas Release (%) for H. B. Robinson Unit 2 Assembly BO-5.

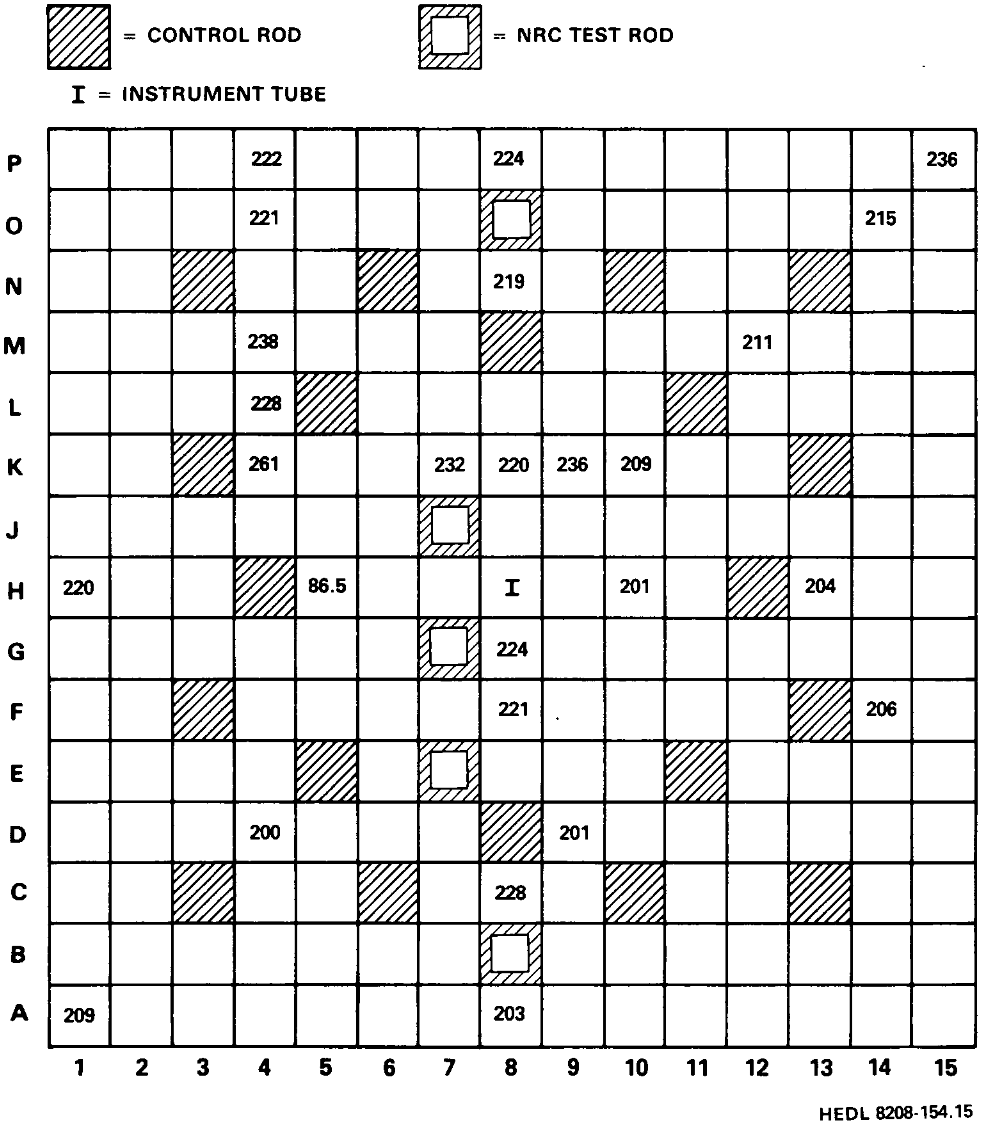


FIGURE 20. Internal Rod Pressure (psia) for H. B. Robinson Unit 2 Assembly BO-5.

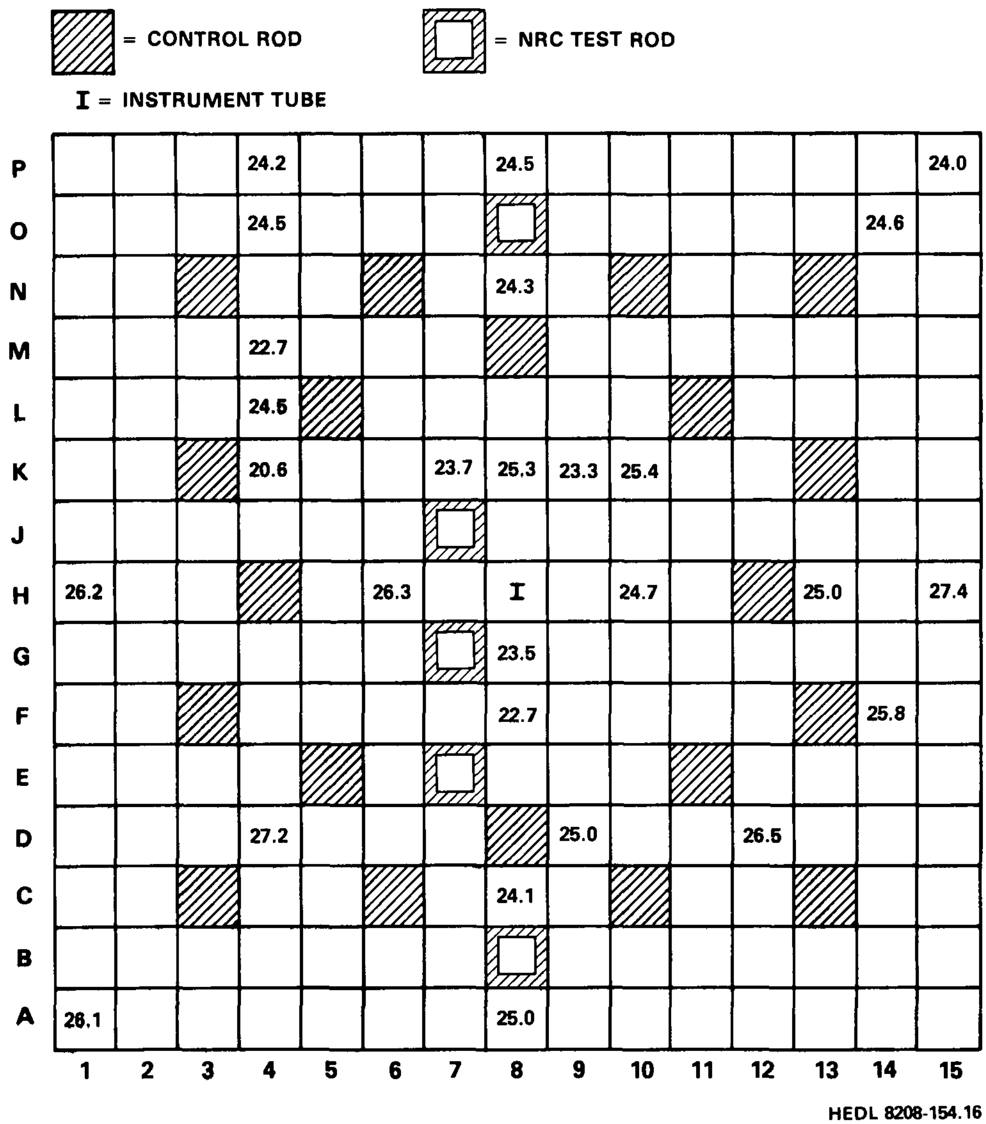


FIGURE 21. Rod Void Volume for H. B. Robinson Unit 2 Assembly BO-5.

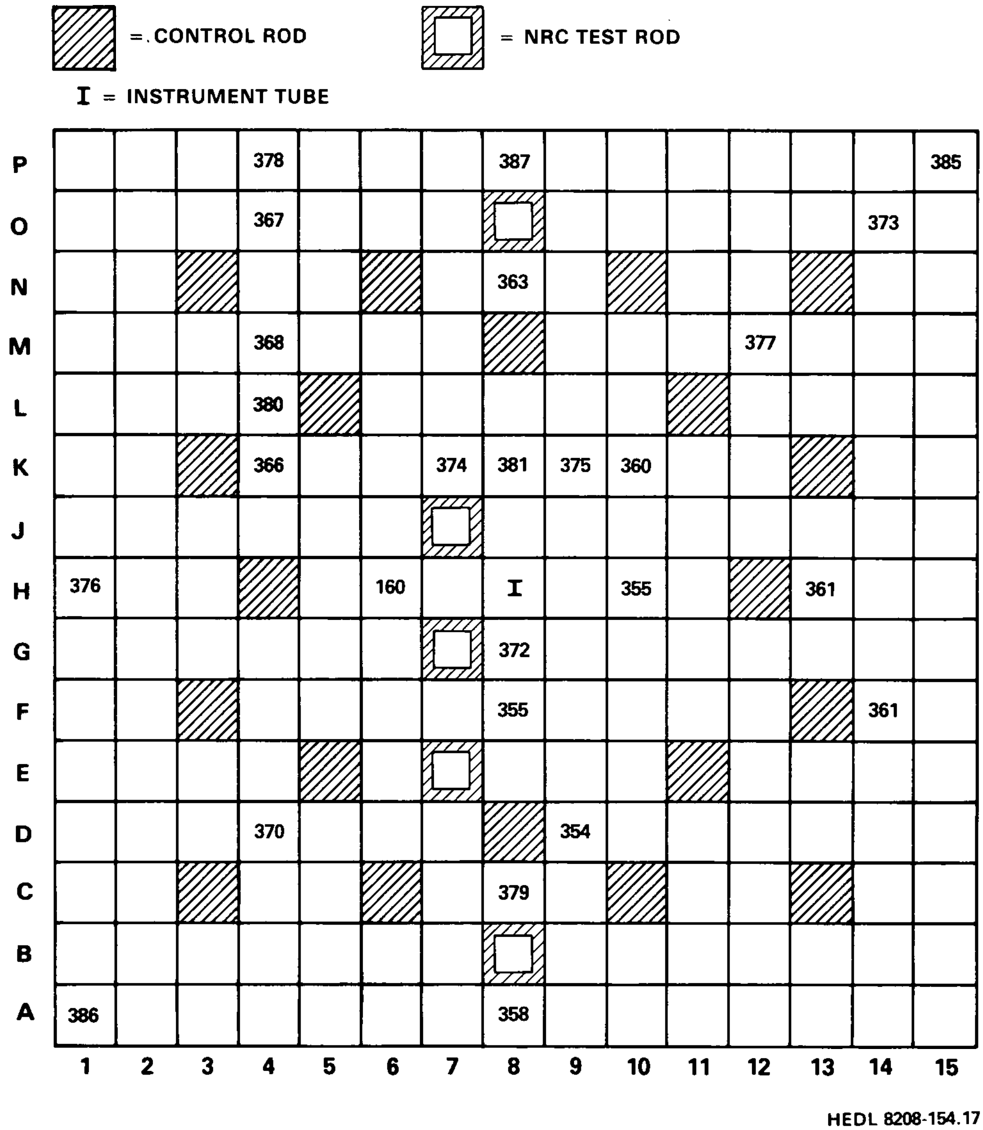


FIGURE 22. Gas Volume (cc @ STP) for H. B. Robinson Unit 2 Assembly BO-5.

TABLE 13a

MOLECULAR FISSION GAS COMPOSITION ANALYSIS OF H. B. ROBINSON UNIT 2 ASSEMBLY BO-5

Molecular Gas Analysis (%)

Rod	H_2	He	CH_4	H_2O	O_2	N_2	Ar	CO_2	Kr	Xe	Xe/Kr	Reference
H-1	<0.01	98.3	---	---	<0.01	0.01*	0.67	<0.01	0.11	0.87	7.91	3
K-4	<0.01	97.7	---	---	<0.01	0.02*	1.31	0.01	0.10	0.89	8.90	3
L-4	0.03	97.0	---	---	0.04	0.10*	1.33	0.03	0.16	1.31	8.19	3
M-4	<0.01	97.6	---	---	<0.01	0.05*	1.29	<0.01	0.10	0.89	8.90	3
O-4	<0.01	98.6	<0.01	<0.1	<0.01	0.09	0.33	<0.01	0.10	0.81	8.10	22
K-7	<0.01	98.1	---	---	<0.01	0.06*	0.98	0.01	0.09	0.76	8.44	3
N-8	<0.01	96.5	<0.01	<0.1	0.10	0.74	1.39	<0.01	0.13	1.09	8.38	22
K-9	<0.01	98.3	---	---	0.01	0.04*	0.71	0.01	0.10	0.86	8.60	3
K-10	<0.01	98.1	<0.01	<0.1	<0.01	0.09	0.72	<0.01	0.12	0.97	8.08	22
D-12	<0.01	95.2	0.01	<0.1	<0.27	1.27	2.05	<0.01	0.133	1.04	7.82	26,27
F-14	--	98.2	---	---	<0.05	0.1	0.7	0	0.1	0.9	9.0	Unpublished Data
H-15	<0.01	95.2	0.01	<0.1	<0.49	2.04	0.91	<0.01	0.158	1.18	7.47	26,27
P-15	--	98.4	---	---	<0.05	0.1	0.2	0	0.1	1.1	11.0	Unpublished Data

*Contains some CO also.

TABLE 13b

ISOTOPIC FISSION GAS COMPOSITION ANALYSIS OF H. B. ROBINSON UNIT 2 ASSEMBLY BO-5

Krypton and Xenon Isotopic Composition (%)

Rod	Krypton				Xenon						Reference
	83	84	85	86	131	132	134	136	130	128	
H-1	11.5	31.1	6.2	51.2	7.37	20.91	28.69	42.90	0.13	----	3
K-4	11.7	31.1	6.1	51.1	8.33	20.61	28.02	42.91	0.13	----	3
L-4	11.5	31.3	6.5	50.7	7.81	20.66	28.44	42.86	0.16	0.07	3
M-4	11.6	30.9	6.3	51.2	8.40	20.48	28.25	42.74	0.13	----	3
O-4	13.1	30.2	6.61	50.1	8.74	20.8	28.5	42.0	----	----	22
K-7	11.7	31.3	6.2	50.8	8.47	20.54	28.41	42.42	0.15	----	3
N-8	12.4	31.9	5.78	49.9	8.63	20.9	28.4	42.1	----	----	22
K-9	12.7	31.2	5.8	50.3	8.44	20.55	28.33	42.49	0.14	0.05	3
K-10	13.1	31.9	5.11	49.9	8.73	20.4	29.7	41.2	----	----	22
K-12	13.7	31.1	7.17	48.0	7.86	21.0	28.9	42.2	----	----	26,27
F-14	11	32	5	52	7.8	21.1	28.6	42.5	----	----	Unpublished Data
H-15	13.7	30.8	7.67	47.9	8.10	21.0	28.9	42.0	----	----	26,27
P-15	12	32	5	51	7.8	21.0	28.4	42.8	----	----	Unpublished Data

TABLE 14

PEACH BOTTOM-II ASSEMBLY PH 462

Rod	Void Volume (cc)	Gas Volume (cc @ STP)	Gas Pressure (psia)	Fission Gas Release (%)
F-4	75.1	136	24.0	2.94
F-5	72.2	118	26.5	2.94

low burnup is not unexpected. Unpressurized fuel tends to have greater gas release than pressurized fuel[28] due to the lower thermal conductivity across the fuel-cladding gap. Pressurized BWR fuel would have gas releases lower than 1% up to approximately the current burnup limits of 25 GWD/MTM.

The isotopic and molecular gas analyses are given in Table 15. The molecular analysis shows a higner percentage of krypton and xenon since the released fission gas is a larger percentage of the initial gas fill in an unpressurized rod. The isotopic composition is nearly the same as the H. B. Robinson Unit 2, as expected.

TABLE 15

GAS COMPOSITION ANALYSIS OF PEACH BOTTOM-II ASSEMBLY PH 462

Molecular Composition (%)

Rod	H_2	He	N_2	O_2	Ar	CO_2	Kr	Xe	Xe/Kr
F-4	<0.1	58.0	0.1	<0.1	2.9	<0.1	4.5	34.5	7.67
F-5	<0.1	60.1	0.2	<0.1	3.2	0.1	4.2	32.2	7.67

Isotopic Composition

	Krypton				Xenon					
Rod	83	84	85	86	128	130	131	132	134	136
F-4	13.55	28.67	6.10	51.68	0.01	0.7	10.25	19.45	30.05	40.17
F-5	13.48	28.77	6.11	51.64	0.01	0.6	10.12	19.57	30.13	40.11

2. Cladding Characteristics

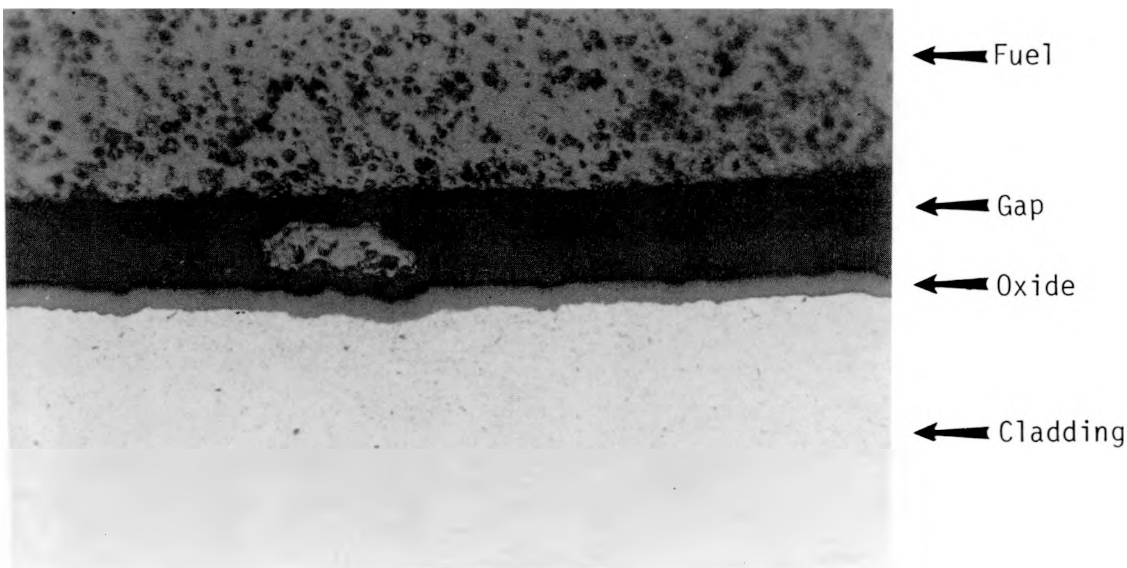
H. B. Robinson Unit 2 Assembly BO-5 -- Metallography was done along the length of Rods K-4[3] and O-14.[15] An extensive series of photographs is on file at EG&G-Idaho. Fuel Rod K-4 showed hydrides that were primarily circumferentially oriented. No fusion analysis for hydride content was conducted. Exterior and interior cladding oxide was measured on both rods. Typical examples of each are shown in Figure 23. The oxide in both cases was very uniform. No globular oxidation was observed. As shown in Figure 24, the oxide thickness increases with increasing distance up the rod. This is to be expected due to the coolant axial temperature gradient. The exterior oxide layer showed no azimuthal dependence.[29] Rod K-4 had a slightly thinner oxide layer than Rod O-14. The data are not sufficient to determine if this difference is significant. The internal oxide layer is also slightly thinner than the external oxide layer.

Microhardness measurements[8] were taken at the locations indicated in Figure 9. The sampling adequately represents the assembly. As indicated in Table 16, there is no significant cladding hardness differences in the assembly. The cladding hardness, which would provide an indication of irradiation damage annealing, should be monitored at intervals during the whole rod tests.

Tensile specimens were tested over a range of temperature as indicated in Table 17.[8,23] The strength decreases linearly from 80 to 700°F while the ductility remains essentially constant.

Peach Bottom-II PH 462 -- Rods B-5 and D-1 from fuel Assembly PH 462 were metallographically examined.[7] Since the sections were taken from near the breach location in these two rods, most of the examination concentrated on the characteristics of the breach. Some samples were taken away from the breach zone. Data on hydride orientation and oxide thickness were obtained.

(a) Exterior Oxide



(b) Interior Oxide

FIGURE 23. Oxide Layers on Rod K-4 from H. B. Robinson Unit 2 Assembly BO-5 at 500X.
a) Exterior oxide 3.52 m from the rod bottom.
b) Interior oxide 3.52 m from the rod bottom.
(Courtesy of EG&G-Idaho).

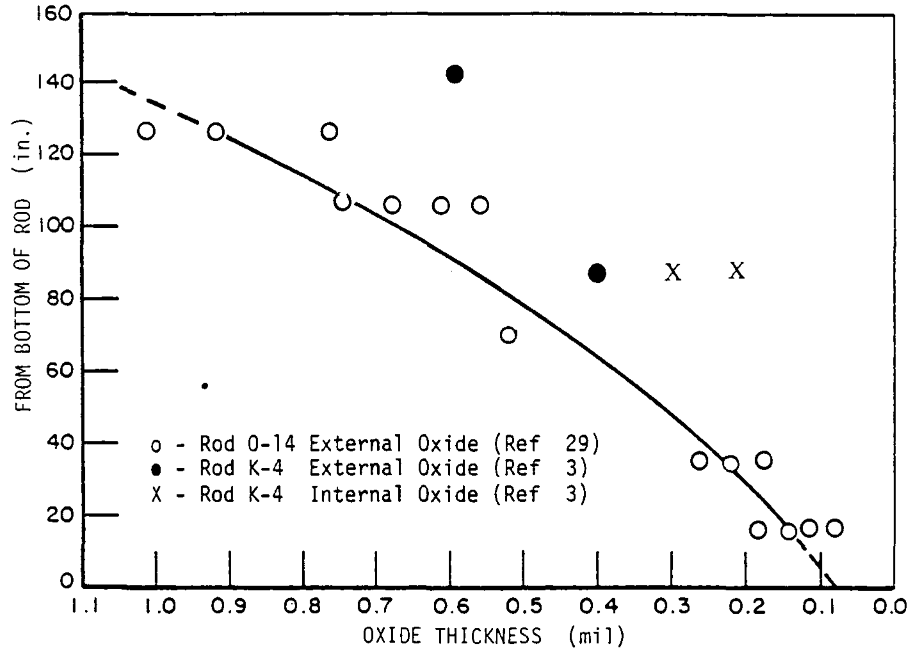


FIGURE 24. Oxide Thickness Variations Along Zircaloy Fuel Rod from H. B. Robinson Unit 2.

TABLE 16

HARDNESS OF AS-RECEIVED, AS-IRRADIATED ZIRCALOY CLADDING FROM H. B. ROBINSON UNIT 2 ASSEMBLY BO-5 RODS (Knoop hardness with 1-kg load)[8]

Rod Number	Distance from Bottom of Rod (in.) 16	35	70	105	126
A-1	253	250	240	250	249
	243	249	240	258	246
A-8	240	242	253	239	246
	248	263	253	252	256
0-14	259	245	244	240	246
	264	250	239	247	253
P-4	251	237	236	240	244
	264	240	244	245	242
P-8	243	261	272	242	253
	261	268	262	260	261
H-10	270	253	253	241	250
	256	250	252	250	238

Upper number has indentation parallel to radius; lower number has indentation perpendicular to radius.

TABLE 17

TENSILE TEST RESULTS AS A FUNCTION OF TEMPERATURE FOR AS-IRRADIATED ZIRCALOY CLADDING* FROM H. B. ROBINSON UNIT 2 ASSEMBLY BO-5

Specimen No.	Location (in.)**	Temp °C±3°	Strength (psi)***		Elongation (%)		Ref
			Ultimate	0.2% Offset	Uniform	Total	
P8-7		27	135,833	117,500	2.06	3.0	23
P8-21		27	142,833	117,000	4.20	8.02	23
P8-9		27	145,000	120,333	3.85	7.00	23
P8-19		93	133,000	110,667	3.76	5.46	23
P8-23		93	131,167	108,333	3.80	7.17	23
P8-37		93	130,333	107,333	3.62	7.09	23
P8-8		204	119,167	96,667	3.85	5.54	23
P8-38		204	116,000	97,500	4.23	7.34	23
P8-47		204	112,667	92,667	3.96	7.34	23
P8-51		316	99,333	83,000	2.69	6.33	23
P8-52		316	97,167	81,333	2.31	5.70	23
P8-50		316	102,167	87,167	2.77	6.84	23
P8-20	42-47	371	93,450	72,760	4.1	13.2	8
P8-34	73-78	371	97,590	78,620	4.0	14.0	8
P8-46	94-99	371	95,860	78,280	2.8	8.6†	8

*Strain rate 0.005/min. to yield stress, 0.025/min to fracture.
**Measured from bottom of fuel rod.
***Based on tube area of 0.029 in^2.
†Defect possible cause of fracture.

Figure 25 shows a typical hydride concentration. The inner 40% has radial hydrides and the outer 60% has circumferential hydrides. This is in contrast to the H. B. Robinson Unit 2 rods, which had only circumferential hydrides. The radial hydrides that weaken the cladding(30) could be caused either by the particular texture introduced during the manufacturing process or by excessive cladding stress(31) during operation. This hydride distribution may not be typical of BWR rod cladding. Since the test rods are from the opposite side of the assembly (away from the control rod blade), a metallographic sample of the hydride orientation and fusion analyses of the hydride content should be conducted on companion material if post-test examination also shows radial hydriding.

Typical internal and external oxide layers are shown in Figure 26. In contrast to the H. B. Robinson Unit 2 rods that showed a smooth uniform oxide layer, the outer oxide on the Peach Bottom-II rods was globular. The inner surface oxide was uniform. Oxide thicknesses are ~0.4 mil on the ID and range up to a maximum of 1.3 mil on the OD surface.(7) Since these samples were near the bottom of the rod, the oxide thicknesses are significantly greater than found on the H. B. Robinson Unit 2 rods.

3. Fuel Pellet Characteristics

H. B. Robinson Unit 2 Assembly BO-5 -- Ceramographic examination of the H. B. Robinson Unit 2 fuel assembly was conducted at both EG&G(3) and ANL.(9,10) Two rods were examined (K-4 between the 2.26- and 3.51-m elevation and F-7 between the 0.05- and 0.89-m elevation). Micrographs are on file at both ANL and EG&G. Typical transverse sections are shown in Figure 27. The pellet is cracked in the radial direction. As tabulated in Table 18, usually 4 to 13 fragments formed in a cross section. The one longitudinal sample, which was 2 diameters long, had twice as many fragments; therefore, ~16 to 170 fragments formed per pellet diameter(2) of fuel volume. The length and width of the major cracks were also measured. The area from the cracks can be compared to the area from the measured fuel pellet-to-cladding gap (see Table 19). Except at the 3.27-m elevation, the

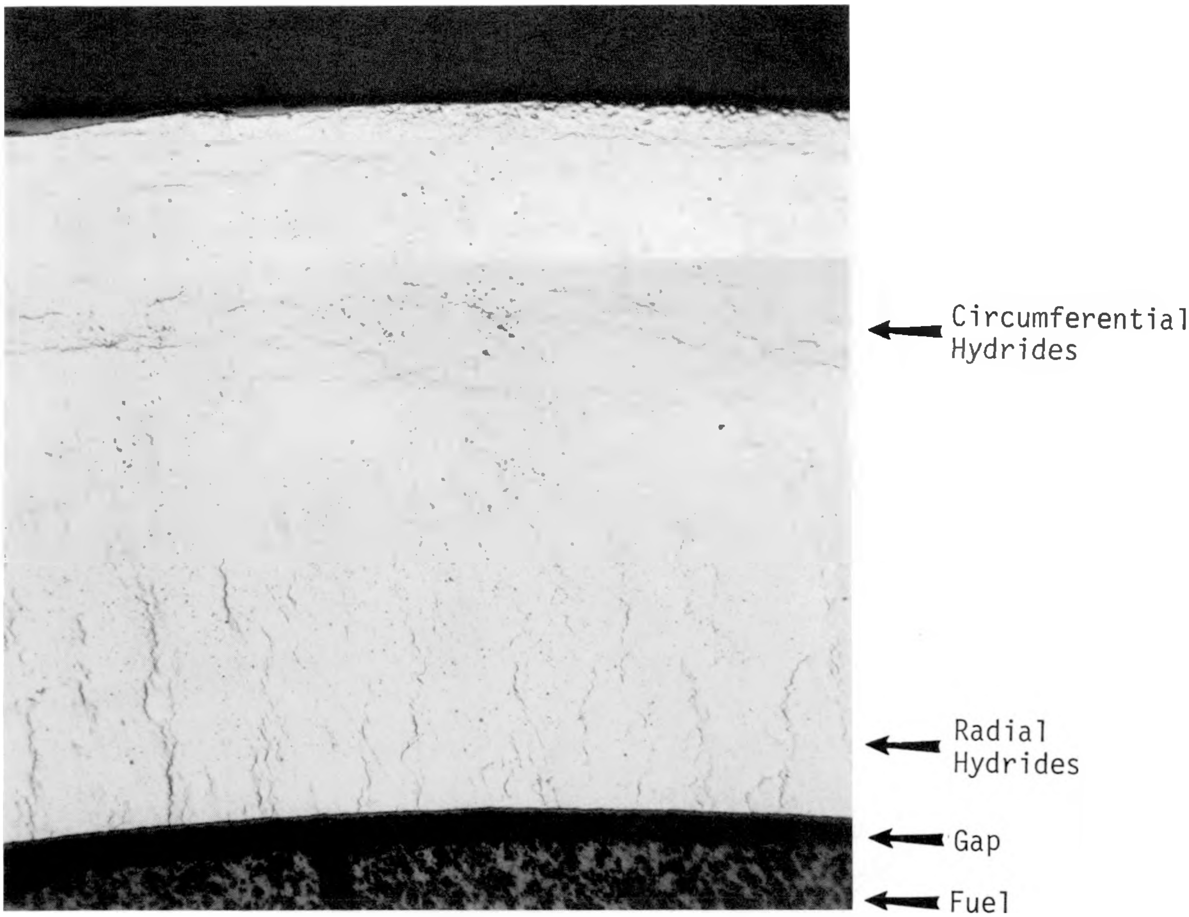
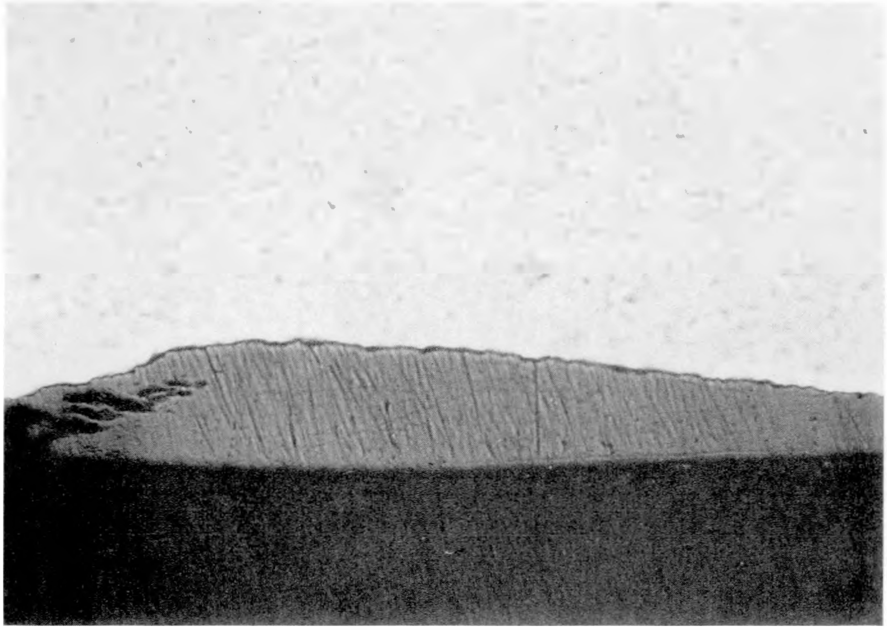


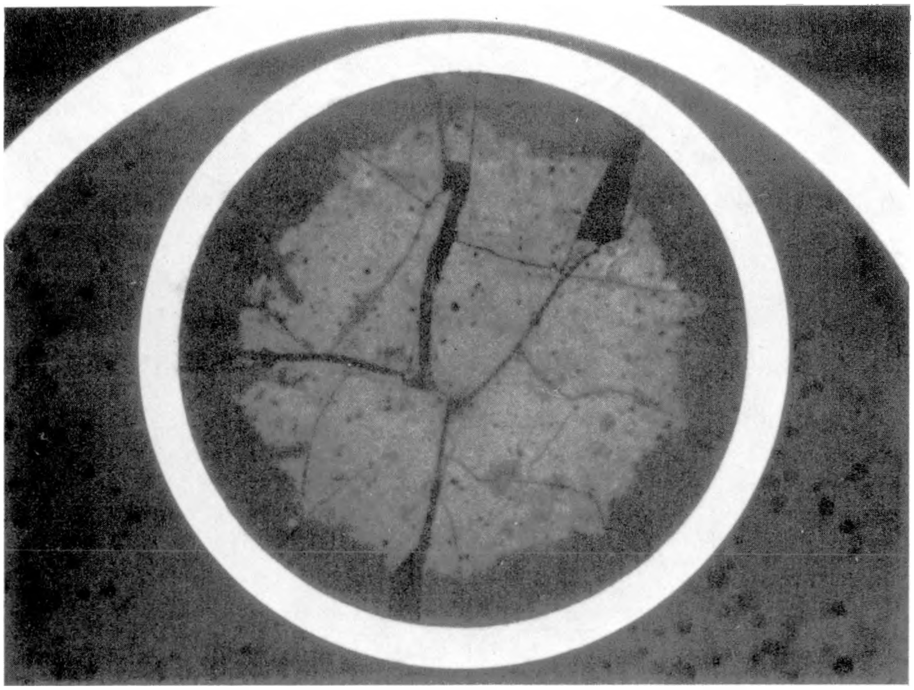
FIGURE 25. Hydride Structure at 0.55-m Elevation of Rod B-5 from Peach Bottom-II at 100X. (Courtesy of EG&G-Idaho)



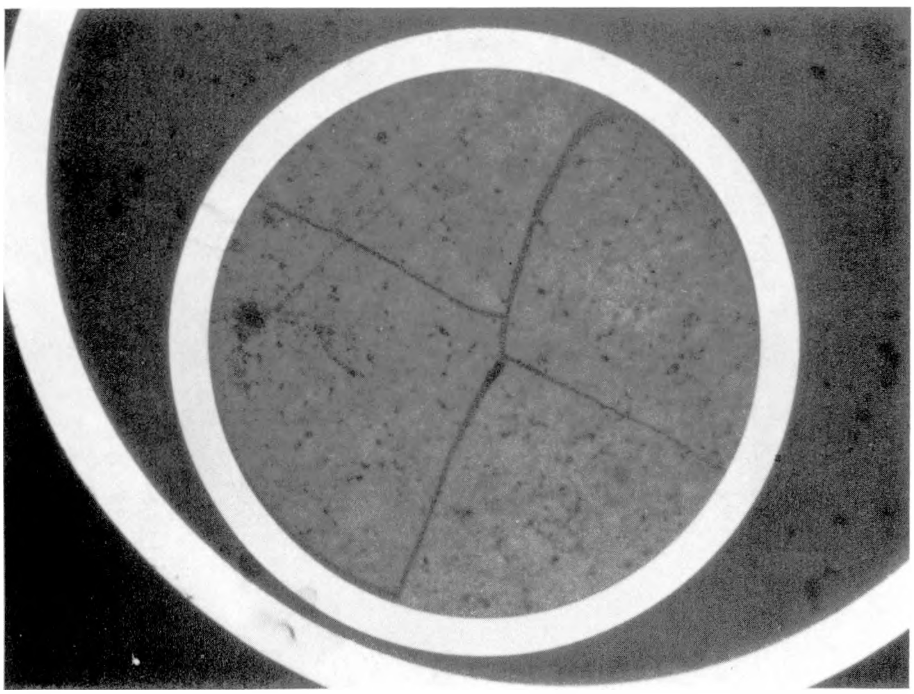
(a) Outer

(b) Inner

FIGURE 26. Typical Oxide Formations on Fuel Rod B-5 from Peach Bottom-II at 0.55-m Elevation from the Bottom at 500X. (Courtesy of EG&G-Idaho)



(a) 2.26-m Elevation



(b) 3.52-m Elevation

FIGURE 27. Transverse Sections of Fuel from Rod K-4 for H. B. Robinson Unit 2 Assembly BO-5. (Ref 3)

TABLE 18

FRAGMENTATION OF PELLETS FROM H. B. ROBINSON UNIT 2 ASSEMBLY BO-5

Rod	Elevation* (m)	Type of Section	No. Large Fragments	Ref	Equivalent Crack Area (mm^2)
K-4	3.22	Transverse	7-9	3	0.90
K-4	2.26	Transverse	9-13	3	2.11
K-4	2.46	Transverse	10-12	3	1.47
K-4	2.88	Longitudinal	23-24**	3	--
K-4	2.36	Transverse	8	3	--
K-4	2.66	Transverse	8-12	3	--
K-4	3.51	Transverse	4-5	3	1.37
K-4	3.27	Transverse	10-11	3	0.68
F-7	0.89	Transverse	3-13	10,9	--
F-7	0.05	Transverse	7	10	--

*Measured from bottom of fuel rod.
**Length of sample was approximately two diameters.

TABLE 19

METALLOGRAPHIC MEASUREMENT OF AVERAGE RADIAL GAP[3] IN FUEL RODS FROM H. B. ROBINSON UNIT 2 ASSEMBLY BO-5

Rod	Elevation*	Average Radial Gap (μm) Orientation (degrees) 0	30	60	90	120	150	180	210	240	270	300	385	Average for Sample	Avg Gap Area (mm^2)
K-4	3.22	18.0	25.0	37.5	**	**	23.5	24.5	35.0	**	37.5	47.5	32.5	31.2	0.93
K-4	3.27	32.5	20.0	20.0	40.0	55.0	20.0	20.0	27.5	22.5	**	35.0	27.5	29.1	0.87
K-4	3.52	10.0	15.0	25.0	22.0	52.5	9.0	12.5	20.0	25.0	14.0	70.0	27.5	25.2	0.75
K-4	2.26	37.5	25.0	25.5	25.0	35.0	32.5	30.0	30.0	30.0	35.0	35.0	25.0	30.5	0.91
K-4	2.36	15.0	18.0	10.0	17.5	20.0	15.0	10.0	10.0	15.0	10.0	13.0	20.0	14.5	0.43
K-4	2.46	25.0	27.5	27.5	27.5	30.0	18.5	30.0	20.0	20.0	30.0	40.0	30.0	27.2	0.81
K-4	2.66	20.0	20.0	10.0	10.0	28.0	10.0	35.0	28.0	18.0	22.5	15.0	7.0	18.6	0.56
F-7***	0.89	110 to 22 range													

*Measured from bottom of rod (meters).
**Missing fuel prevented gap measurement.
***Ref. 10.

cracks provide a larger cross section area for flow of gas through the rod than through the gap. The large cross section for gas flow would indicate that a grossly breached rod stored in an air atmosphere would have its full fuel column exposed to the oxidizing atmosphere. Hence, if the UO_2 converts to U_3O_8 during storage, it may be more than a locally initiating phenomenon.

From the 2.26- to 3.51-m elevations, little grain growth occurred (see Figure 28). Average fuel grain size varied from 7 µm in the peak power range (2.26 m) to 4 µm near the top of the rod.[3] At the 0.89-m elevation, the grain size had a radial gradient with a mean diameter of ~6 µm in the central region and ~3 µm near the outer surface.[10] The porosity in Figure 27b at the 3.52-m elevation is very similar to that at the 0.89-m elevation. The coarse porosity ranges from 30 to 250~m in diameter and occupies a volume fraction of 0.03.[10] According to ANL, the gross porosity is due to pore formers added during fabrication.[10]

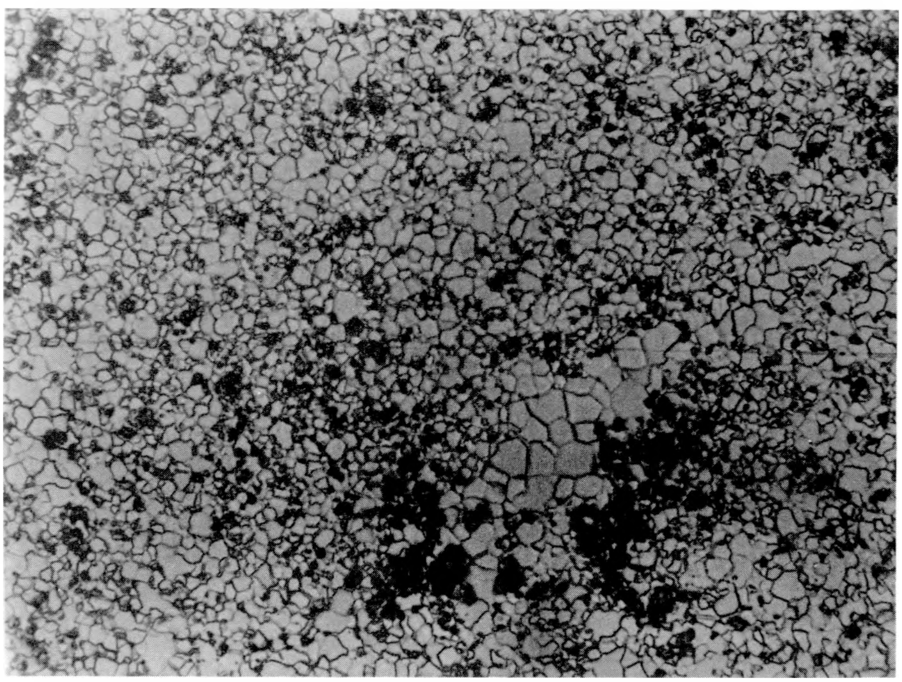


FIGURE 28. Typical Fuel Structure at Center of Pellet from Fuel Rod K-4 for H. B. Robinson Unit 2 Assembly BO-5 at 3.52-m Elevation. 500X. (Ref 3)

Both PNL[32,33] and LANL[34] conducted radial microprobe measurements on unidentified fragments of H. B. Robinson Unit 2 fuel. The results were essentially the same. PNL traces are shown in Figure 29. Other than Pu and Ru, no elements showed radial gradients. [Besides the elements shown in Figure 29, LANL conducted Sr and Sb analyses[34] and found no radial gradients.] The Pu concentration is ∿70% higher on the surface, which can be probably attributed to its mode of formation; fissionable plutonium preferentially forms near the outer surface of the fuel pellet due to resonance absorption of thermal neutrons by ^{238}U.[38] No reason has been postulated for the higher ruthenium concentration near the surface.

Peach Bottom-II Assembly PH 462 -- Ceramographic studies of the fuel from the area near the breaches in Rods D-1 and B-5 have been conducted.[7] Even though the examination is exhaustive with information on grain size, void volume, and pellet cracking, the information may not be representative of the fuel structure in an unfailed rod. It is, therefore, recommended that a few ceramographic sections be examined from unfailed Peach Bottom-II fuel.

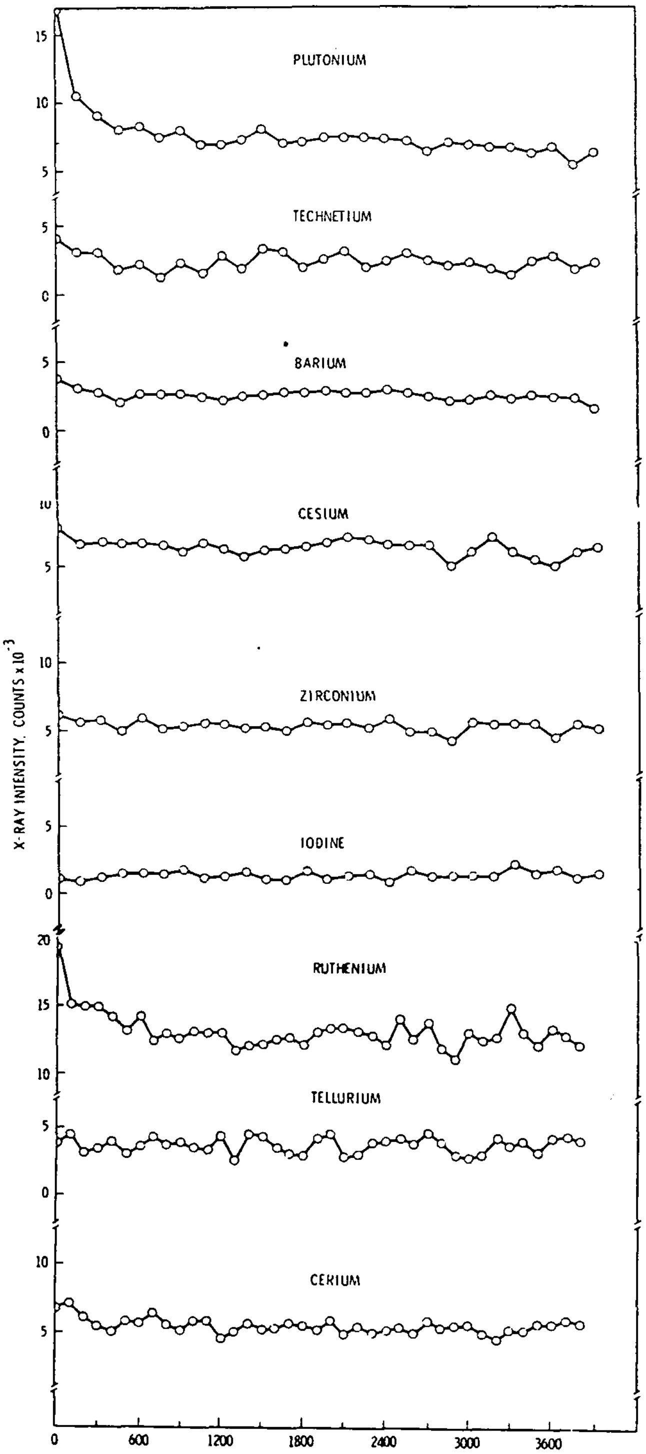


FIGURE 29. Radial Microprobe-Measured X-Ray Intensity Across Fuel Pellet from H. B. Robinson Unit 2 Assembly BO-5. Zero corresponds to outside pellet diameter. (Ref 32 & 33)

V. EXPECTED CONDITIONS OF TEST RODS BASED ON AVAILABLE CHARACTERIZATION

As stated in Section I, there are three main reasons for characterization: 1) to select test rods, 2) to establish the initial fuel rod condition so that changes that take place during the test may be determined, and 3) to compare the test rods with the general population of spent fuel available for storage. The characterization is broken into two categories: 1) fabrication and environmental information and 2) NDE data. Ideally, complete characterization on all test rods is desired prior to the experiment; this is not necessary nor do available resources permit it. Thus, the preponderance of pretest characterization data for the H. B. Robinson Unit 2 and the Peach Bottom-II rods used in the present experiment come from other programs.

A. H. B. ROBINSON UNIT 2 ASSEMBLY BO-5 RODS

The H. B. Robinson Unit 2 Assembly BO-5 rods are as well characterized as any rods examined. The fabrication and environmental history is adequate. Fabrication is similar to the new 17 x 17 assemblies, and the burnup of 30 GWd/MTM is typical of present day fuel. No pool storage history is available. There was some uncertainty on the rod temperatures during the transfer procedure at EG&G but the excursion was shown by hardness testing to have been insufficient to cause annealing of the irradiation damage.

The only planned NDE of the actual test rods is a visual examination. This should be adequate since over 20% of the rods in the assembly have been nondestructively examined by gamma scan, eddy current, profilometry, and visual with consistent results. The test rods were chosen next to and surrounded by, if possible, well characterized rods. Further, the most pertinent and useful data for purposes of this test are from DE, and the DE of other rods in the assembly has been extensive. This assembly represents one of the few cases when as-irradiated tensile properties of the cladding are available so environmental effects of whole rod testing on cladding strength can be determined.

While additional pretest NDE on the actual test rods might be desireable for completeness, it is not necessary for a successful test. No additional DE of comparison rods is needed. The H. B. Robinson Unit 2 rods compare very closely with the Turkey Point rods, which were of the same vintage and burnup.

B. PEACH BOTTOM-II ASSEMBLY PH 462 RODS

The Peach Bottom-II Assembly PH 462 rods are not nearly as well characterized as the H. B. Robinson Unit 2 rods. Adequate fabrication and reactor environmental history are available. The assembly did undergo an abnormal reactivity insertion, which caused the breach of two rods. Care was taken to choose test rods away from the control rod. Eddy current and ultrasonic examination at poolside showed all the rods except three to be intact. The assembly has three zones of enrichment. Since the fission gas analysis examination was done in the region of highest enrichment, the test rods were also chosen from this region. Little is known about the transportation or storage history of the rods except that there are no reports of unusual occurrences.

As indicated, EC examination has been conducted on the rods. Pretest visual examination of the test rods is planned as part of the present test. Neither profilometry nor gamma scanning information is available. Both would be interesting to have but are not critical to the present test. Due to the low temperature of the test, profilometry would not be accurate enough to detect cladding diameter changes. Gamma scanning is also only useful when looking for gross changes, such as fuel oxidation, in which case comparison is made within a scan and not between pre- and post-test scans. While the NDE on the rods is not complete, it is adequate with the addition of the scheduled visual examinations, for the purposes of this test. A step-wise scan on the crud grinding rod should be made to determine the fluence profile.

Small changes in the fuel and cladding conditions will be determined by DE. An initial examination will be made of the Peach Bottom-II test rod after two furnace runs (20 months), and all test rods will be examined at the end of the test. Fission gas analyses on two rods were very consistent with each other and, since they had the same enrichment as the test rods, adequately represent the fission gas release in the test rods. The DE on the breached rods may have been sufficiently influenced by the breach to render the results invalid for intact rods. A full metallographic examination consisting of metallography, ceramography, burnup, fission gas analysis, and hardness is planned on two samples from the rod being cut for crud evaluation. This may be done as resources permit if the samples are stored in an inert atmosphere. Since only small changes in the cladding and fuel structure are expected, DE will be a primary tool. Care should be taken to set aside DE samples, so that a better DE data base can be established when necessary.

The power levels on the Peach Bottom-II fuel are nearly the same as more recent fuel. The burnup of the Peach Bottom-II fuel is lower than more recent fuel (12 GWd/MTM vs 25 GWd/MTM), but the fission gas release of the Peach Bottom fuel is higher (2.9% vs 1%). This combination of burnup and fission gas release indicates that the internal rod fission product atmosphere in the Peach Bottom-II fuel is probably comparable to modern vintage BWR fuel. Therefore, the results of the test with this fuel will be a good representation of the expected results from modern BWR fuel.

VI. REFERENCES

1. R. E. Einziger, R. L. Fish and R. L. Knecht, A Technical Description of the NRC Long-Term Whole Rod and Crud Performance Test, NUREG/CR-2889, HEDL-TME 82-32, Hanford Engineering Development Laboratory, Richland, WA, September 1982.

2. J. H. Goode and R. G. Story, Head-End Reprocessing Studies with H. B. Robinson Unit 2 Fuel, ORNL/TM-6037, Oak Ridge National Laboratory, Oak Ridge, TN, June 1978.

3. S. J. Dagbjartsson, B. A. Murdock, D. E. Owens and P. E. MacDonald, Axial Gas Flow in Irradiated PWR Fuel Rods, TREE-NUREG-1158, Nuclear Regulatory Commission, Washington, DC, September 1977.

4. H. B. Robinson Unit 2 FSAR - Docket 50261-104, Carolina Power & Light Co., Raleigh, NC.

5. D. O. Campbell, S. R. Buxton, R. L. Beatty and W. L. Pattison, LWR Fuel Reprocessing and Recycle Program Quarterly Report for Period July 1 to September 30, 1976, p. 320, ORNL/TM-5060, Oak Ridge National Laboratory, Oak Ridge, TN, November 1976.

6. P. H. Larson, Core Design and Operating Data for Cycles 1 & 2 of Peach Bottom-II, EPRI NP-563, Electrical Power Research Institute, Palo Alto, CA, June 1978.

7. D. E. Owen, Interim Report on the Postirradiation Examination of Peach Bottom-II BWR Fuel Rods, TFBP-TR-264, EG&G, Idaho Falls, ID, March 1978.

8. Progress on Evaluating Strength and Ductility of Irradiated Zircaloy During July through September 1975, BMI-1938, Battelle Memorial Institute, Columbus, OH, p.3, 1975.

9. S. M. Gehl and M. G. Seitz, "Burnup Analysis," Light Water Reactor Safety Research Program: Quarterly Report, Oct - Dec 1975, ANL-76-15, Argonne National Laboratory, Argonne, IL, p. 50, 1975.

10. S. M. Gehl and L. R. Kelman, "Fuel Characterization," Light Water Reactor Safety Research Program: Quarterly Report July - Sept 1975, ANL-75-72, Argonne National Laboratory, Argonne, IL, p. 42, 1975.

11 Nuclear Power Plant Operating Experience 1976, NUREG-0366, Nuclear Regulatory Commission, Washington, DC, 1976.

12. R. A. Lorenz, J. L. Collins, M. F. Osborne, R. L. Towne and A. P. Malinauskas, Fission Product Release from BWR Fuel Under LOCA Conditions, NUREG/CR-1773, Nuclear Regulatory Commission, Washington, DC, July 1981.

13. K. R. Perkins, Thermal Transients of the Carolina Power and Light Fuel Bundle in Air, KRP-4-75, Carolina Power & Light Co., Raleigh, NC, July 14, 1975.

14. A. A. Bauer, L. M. Lowry and J. S. Perrin, Progress on Evaluating Strength and Ductility of Irradiated Zircaloy During April through June 1975, BMI-1935, Battelle Memorial Institute, Columbus, OH, June 1975.

15. L. M. Lowry, A. J. Markworth, J. S. Perrin and M. P. Landow, Evaluating Strength and Ductility of Irradiated Zircaloy Task 5, Experimental Data Final Report, NUREG/CR-1729, BMI-2066, Vol. 1, Battelle Memorial Institute, Columbus, OH, May 1981.

16. R. B. Davis and V. Pasupathi, Data Summary Report for the Destructive Examination of Rods G7, G9, J8, I9, and H6 from Turkey Point Fuel Assembly B17, HEDL-TME 80-85, Hanford Engineering Development Laboratory, Richland, WA, April 1981.

17. R. E. Einziger, S. D. Atkin, D. E. Stellrecht and V. Pasupathi, "High Temperature Postirradiation Materials Performance of Spent Pressurized Water Reactor Fuel Rods Under Dry Storage Conditions," Nucl. Technol. 57, April 1982.

18. R. S. Kemper and D. L. Zimmerman, Neutron Irradiation Effects on the Tensile Properties of Zircaloy-2, HW-5232, Hanford Engineering Laboratory, Richland, WA, August 1957.

19. D. G. Franklin, J. T. A. Roberts and H. Ocken, "Poolside Inspection of Core Components in Light Water Reactors," ANS Trans. 38, p. 718, June 1981.

20. H. D. Goddard and R. G. Weber, "Nondestructive Measurement of Zirconium Oxide Corrosion Films on Irradiated Zircaloy-Clad Fuel Rods," ANS Trans. 38, p. 721, June 1981.

21. V. Pasupathi et al., Determination and Microscope Study of Incipient Defects in Irradiated Power Reactor Fuel Rods, EPRI NP-812, Electrical Power Research Institute, Palo Alto, CA, July 1978.

22. A. A. Bauer, L. M. Lowry and J. S. Perrin, Evaluating Strength and Ductility of Irradiated Zircaloy, BMI-NUREG-1976, Battelle Memorial Institute, Columbus, OH, July 1977.

23. A. A. Bauer, L. M. Lowry and J. S. Perrin, Evaluating Strength and Ductility of Irradiated Zircaloy, BMI-1942, Battelle Memorial Institute, Columbus, OH, December 1975.

24. Light-Water Reactor Safety Research Program: Quarterly Progress Report for Jan - March 1976, ANL-76-49, Argonne National Laboratory, Argonne, IL, 1976.

25. A. E. Norris, Status Report Concerning Empirical Measurements of Fission Product Release from Nuclear Reactor Spent Fuel, ONWI/ Sub/79/E511-01200-11, p. 2. Battelle Memorial Institute, Office of Nuclear Waste Isolation, Columbus, OH, June 19, 1979,

26. R. A. Lorenz, J. L. Collins and S. R. Manning, Quarterly Progress Report on Fission Product Release from LWR Fuel for the Period Oct - Dec 1975, ORNL/TM-290, Oak Ridge National Laboratory, Oak Ridge, TN, p. 30, March 1976.

27. R. A. Lorenz, J. L. Collins, A. P. Malinauskas, O. L. Kirkland and R. L. Towns, Fission Product Release from Highly Irradiated LWR Fuel, NUREG/CR-0722, ORNL/NUREG/TM-287/R1, Oak Ridge National Laboratory, Oak Ridge, TN, p. 2, February 1980.

28. J. T. A. Roberts, F. E. Gelhaus, H. Ocken, N. Hoppe, S. T. Oldberg, G. R. Thomas and D. Franklin, LWR Fuel Performance Program Progress in 1978, EPRI NP-1024-SR, Electrical Power Research Institute, Palo Alto, CA, February 1979.

29. A. A. Bauer, L. M. Lowry and J. S. Perrin, Evaluating Strength and Ductility of Irradiated Zircaloy, BMI-NUREG-1967, Battelle Memorial Institute, Columbus, OH, January 1977.

30. R. P. Marshall and M. R. Loutham, "Tensile Properties of Zircaloy with Orientated Hydrides," Trans. Am. Soc. Met. 56, p. 693, 1963.

31. D. O. Pickman, "Properties of Zircaloy Cladding," Nucl. Eng. Des. 21, p. 212, 1972.

32. Y. B. Katayamo, D. J. Bradley and C. O. Harvey, Status Report on LWR Spent Fuel LAEA Leach Tests, PNL-3173, Pacific Northwest Laboratory, Richland, WA, March 1980.

33. Y. B. Katayama, Spent LWR Fuel Leach Tests, PNL-2982, Pacific Northwest Laboratory, Richland, WA, April 1979.

34. A. E. Norris, (Ed.), Fission Product Release - January 1 through March 31, 1980, LA-8389-PR, Los Alamos National Laboratory, Los Alamos, NM, p. 3, June 1980.

35. C. A. Friskny and J. A. Turnbull, "The Characteristics of Fission Gas Release for Uranium Dioxide During Irradiation," J. Nucl. Mater. 79, p. 184, 1979.

A P P E N D I X

GENERAL ELECTRIC

NUCLEAR ENERGY

BUSINESS OPERATIONS

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125

June 10, 1982

Dr. Robert Einziger
Westinghouse Hanford Company
P. O. Box 1970
Mail Stop W/A-53
Richland, Washington 99352

Dear Bob:

In response to your telephone request of June 7, 1982 concerning the condition of specific "sound" fuel rods in the Peach Bottom-2 fuel assembly PH-462, a brief description of the techniques used is appropriate.

Fuel rod removal from the fuel bundle for NDT is accomplished by first removing the upper tie plate. Individual rods are then removed from the bundle by using specially designed remote fuel rod grapples. The rods are held by the grapple while the NDT fixture, which is attached to the Fuel Preparation Machine carriage, traverses the active portion of the rod and a portion of the plenum. The NDT fixture performs three basic functions in series: (1)brushes loose crud from the rod, (2) eddy current testing, and (3) ultrasonic testing. After NDT, the rod is returned to its original bundle location and the process repeated in a special sequence for all the rods in the bundle except the spacer capture rod.

The two basic nondestructive test (NDT) methods used during the fuel inspection were continuous wave eddy current (E/C) and ultrasonic (UT). The purpose of the E/C was to determine the location and relative magnitude of possible cladding discontinuties, whereas the UT was used to determine the presence of water within the fuel rods (indicating a cladding perforation). The basic equipment set consisted of a NDT head containing an E/C coil and UT transducer, ultrasonic tester with associated electronics, and oscilloscope for presentation of the UT output, E/C tester with associated electronics, and a strip chart recorder for the E/C readout. The UT transducer and the E/C coil were contained in the NDT head through which the fuel rod passed during the actual testing.

The E/C instrumentation detects cladding flaws, discontinuities, and defects by continuous wave E/C technique. Output signal is automatically recorded in a manner that provides the necessary information regarding the magnitude of the signal and the axial location on the fuel rod of any cladding defect detected. The E/C standard is used to establish the base (35%) E/C signal. The standard contains precision machined flaws to simulate the amount of cladding material removed. The E/C signals from tested rods are then reported as a percentage based on the 35% flaw. The E/C instrumentation calibration is checked at the beginning, mid-way, and at the completion of each bundle or every four hours at a minimum. Fuel rods with E/C signals > 35% are visually examined, those with E/C signals < 35% are considered sound and no further examination is required.

Using this criteria only rods D1 and B5 were identified as failed. One rod, C2, yielded an E/C signal of 57% at 60 inches. Visual examination at this location revealed no discernible flaws in the cladding.

The fuel rods which are of interest to you, i.e., C5, D5, D6, E3, E4, E5, E6 and G4, did not exhibit E/C signals > 35% and the UT test indicated no water in the rods. They were thus assumed sound and no additional examination performed.

If you have any additional questions or concerns please give me a call.

Sincerely,

John E. Gonser

John E. Gonser, Sr. Engineer
Site Fuel & Component Inspection

JEG:bm

Do Not Microfilm Distr.

NUREG/CR-2871
HEDL-TME 82-27
RW

DISTRIBUTION

RW (105

DOE-RL/Office of Asst Manager
for Advanced Reactor Programs
P.O. Box 550
Richland, WA 99352

KR Absher, Chief
Technology Dev Branch

HEDL (50)

LD Blackburn	W/A-40	W. Mills	W/A-40
DA Cantley	W/A-62	WF Sheely	W/A-62
RE Einziger (2)	W/A-53	D. Wray	W/A-40
RL Fish (2)	W/A-53	HH Yoshikawa	W/A-44
LA James	W/A-40	Spent Fuels Files (32)	W/A-40
NE Kenny	W/C-115	Central Files	W/C-110
RL Knecht (2)	W/A-40	Publ Services	W/C-115