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Spent Fuel and Fuel Pool Component Integrity

September 1981

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Pacific Northwest Laboratory
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ANNUAL REPORT - FY 1980
SPENT FUEL AND FUEL POOL COMPONENT INTEGRITY

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SUMMARY

During program FY 1980 staff members of the Spent Fuel and Fuel Pool Component Integrity Program at Pacific Northwest Laboratory (PNL) completed the following major tasks:

- represented DOE on the international BEFAST^(a) Committee
- the program manager, A. B. Johnson, Jr., participated in an International Survey of Water Reactor Spent Fuel Storage Experience, which was conducted jointly by the International Atomic Energy Agency (Vienna) and the Nuclear Energy Agency (Paris); the survey report will be published in 1981.
- provided written testimony and cross statement for the Proposed Rulemaking on Storage and Disposal of Nuclear Waste
- acquired and began examination of the world's oldest pool-stored Zircaloy-clad fuel from the Shippingport reactor, stored ~21 years in deionized water
- acquired and began examination of stainless-clad spent fuel from the Connecticut Yankee Reactor (PWR); the fuel had been stored five years in borated water.
- negotiated for specimens from components stored in spent fuel pools at the following fuel storage facilities:
 - Receiving Basin for Offsite Fuel (RBOF) at the Savannah River Plant, Aiken, South Carolina
 - Zion (PWR) spent fuel pool, Zion, Illinois
 - La Crosse (BWR) spent fuel pool, La Crosse, Wisconsin.
- planned for examinations in FY 81 of specimens from the three spent fuel pools
- investigated a low-temperature stress corrosion cracking mechanism that developed in piping at a few PWR spent fuel pools.

This report summarizes the results of these activities and investigations. Details are provided in the presentations and publications generated under this program and summarized in Appendix A.

(a) BEhavior of Fuel Assemblies in Storage, operating under the Nuclear Energy Agency of the Organization for Economic Cooperation and Development, Paris, France.

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INTRODUCTION

The Pacific Northwest Laboratory (PNL), operated by Battelle Memorial Institute for the US Department of Energy, is conducting a program entitled "Spent Fuel and Fuel Pool Component Integrity." The program, which has both domestic and international involvements, conducts examinations on spent fuel and specimens from spent fuel pool components. It monitors information on the behavior of spent fuel and of pool components that function in spent fuel pools.

This report briefly summarizes the program activities for FY 1980 and directs the reader to publications for details of the investigations. Presentations and publications generated under the program are also summarized.

Storage of spent nuclear fuel in water has been and will continue to be an important factor in nuclear fuel management. The following is an overview of the status of spent fuel management in the US:

- Currently (1980) water storage of spent nuclear fuel is the only licensed option.
- The last commercial reprocessing of spent fuel occurred in 1971. Since then the commercial spent fuel inventory stored in US water pools has risen to about 28,000 assemblies in 1980 (Carter 1980).
- The longest pool residence for Zircaloy-clad fuel is 20 years (Shippingport fuel - examined in 1980).
- Dry storage is being investigated as an interim fuel management measure (Anderson and Meyer 1980). However, spent fuel would continue to be discharged to water storage for a cooling period before transfer to dry storage.
- Several countries are reprocessing or preparing to reprocess spent nuclear fuel. The future of commercial reprocessing in the US continues to be uncertain, however. A decision to return to commercial reprocessing would require several years to implement. A decision to proceed with permanent disposal of unreprocessed fuel also would require considerable time to implement (U.S. Department of Energy 1980). Therefore, interim storage will continue to be an important factor in spent fuel management in the US for an indefinite period. In any mode, interim water storage will continue to be an essential factor.

The BEFAST committee, an international spent fuel storage committee, met in Paris on July 2 & 3, 1980. Representatives from the following countries attended: Austria, Belgium, Canada, Finland, France, Federal Republic of Germany (FRG), Italy, Norway, The Netherlands, Spain, Sweden, Switzerland, United Kingdom, and United States. Observers from International Atomic Energy Agency (IAEA) and the Organization for Economic Cooperation and Development (OECD) also attended. A. B. Johnson, Jr. of PNL was the US representative. Each national representative reviewed the scope of pool storage investigations under way or planned.

An international survey of Spent Fuel Storage Experience was conducted jointly by IAEA, Vienna and the Nuclear Energy Agency (NEA, a component of OECD, Paris). A consultant group with members from IAEA, NEA, Austria, Belgium, the United Kingdom and the United States met in

Vienna in April and in Paris in July to draft the international survey report. A. B. Johnson of PNL was the Lead Consultant. The final report will be issued by IAEA after the nineteen contributing countries have commented on the draft.

The following bilateral information exchanges occurred between the US program and programs in other countries.

- Norway, including visits to the Kjeller Laboratory and the Halden reactor (OECD-sponsored BWR sited in Norway), in March 1980
- Kraftwerk Union, FRG, including a visit to Erlangen, FRG, in April 1980.

SPENT FUEL POOL COMPONENT BEHAVIOR

Specimens from stainless steel components stored in three PWR spent fuel pools were acquired and examined in FY 1979 (Johnson, et al. 1980). During FY 1980 negotiations were conducted to obtain fuel pool component specimens from three other storage facilities:

- The Receiving Basin for Off-Site Fuel (RBOF) at the Savannah River Plant, Aiken, South Carolina; specimens promised included weld areas from a stainless steel table that resided for 15 years in deionized water and an aluminum spacer that also was exposed to deionized water.
- The Zion Reactor (PWR) spent fuel pool, Zion, Illinois; specimens promised were sections from stainless steel spent fuel storage racks stored 4 3/4 years in borated water.
- Several specimens were promised from the La Crosse (BWR) spent fuel pool, La Crosse, Wisconsin: sections of a boronated steel storage rack (14 years in a pool), sections of regular stainless steel (14 years in a pool), and sections of boron sheet (3 3/4 year in a pool).

Specimens were received and examined at PNL during FY-1981. Results were incorporated into a report entitled Investigation of Conditions of Spent Fuel Pool Components, PNL-3515.

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CANDIDATE FUELS SELECTION FOR NEAR-TERM STUDIES

Candidate fuels selection for near-term studies focused on identifying three types of sound and defective fuel assemblies:

- BWR fuel assemblies with Zircaloy-clad fuel rods
- PWR fuel assemblies with Zircaloy-clad fuel rods
- PWR fuel assemblies with stainless steel-clad fuel rods.

The literature survey to locate such fuel assemblies is continuing. The search for defective fuel emphasizes locating fuel assemblies with verified defects (e.g., by visual examination).

As stated in the FY 1979 annual report (PNL-3171), the guidelines for fuel assembly selection are as follows:

- extended fuel pool residence or high burnup
- prior examination to define reactor effects
- fuel assemblies available for long-term surveillance.

Tradeoffs will be necessary in selecting the best available fuel for examination and long-term surveillance because not all of the factors will be optimized in a single fuel assembly.

In the continuing effort to select the fuel assemblies for examination and long-term surveillance, particular emphasis is placed on choosing assemblies of the following types:

- a Zircaloy-clad fuel assembly with fuel rods having the optimum combination of extended pool exposures and high burnup
- a PWR fuel assembly with stainless steel-clad fuel rods having an attractive combination of burnup and pool residence
- fuel with known defects.

To identify suitable defective fuel, the search emphasizes locating fuel assemblies with fuel rod defects that have previously been visually confirmed (e.g., small hole in cladding, crack in cladding, missing section of fuel rod). Of particular interest are fuel assemblies having defects associated with peripheral fuel rods (i.e., defects are visible without disassembly of the fuel assembly).

The fuel assemblies that were acquired and examined in FY 1980 are described below.

SHIPPINGPORT SPENT FUEL EXAMINATIONS

Fuel rods from two Shippingport PWR Core 1 blanket fuel bundles, 0551 and 0074, are included in the current examination program, which is assessing the effects of long-term water storage on Zircaloy-clad fuel rods.

Justification for Fuel Selection

The blanket fuel bundles from the Shippingport PWR reactor were selected for examination in the Spent Fuel and Fuel Pool Component Integrity Program because of their long storage times and the availability of data on the condition of similar fuel rods immediately following discharge from the reactor. The specific justifications for their selection are given below.

- Pool Storage Time - Fuel bundle 0551 contains the world's oldest pool-stored Zircaloy-clad fuel rods. The average burnup for this bundle is 346 GJ/kgU (4000 MWd/MTU); the fifteen fuel rods included in the current examinations have been stored in deionized

TABLE 1. Summary of Experimental Program for Shippingport Fuel Examinations

Type of Examination	Fuel Bundle No. 0551	Fuel Bundle No. 0074
Visual	4 Rods	4 Rods
Gamma Scanning	4 Rods	4 Rods
Eddy-Current Testing	4 Rods	4 Rods
Profilometry	2 Rods	2 Rods
Burnup Analysis	1 Rod	1 Rod
Leak Testing and Fission Gas Release	4 Rods	4 Rods
Burst Testing	2 Rods	2 Rods
Metallography	2 Rods (4 Sections)	2 Rods (4 Sections)
Analysis of Cladding	2 Rods (4 Samples)	2 Rods (4 Samples)

Upon arrival at the BCL hot cells, the two bundles were photographed and the exposed fuel rods were examined visually through a stereomicroscope at low magnifications. The general appearances of the seven-rod and eight-rod clusters from bundle 0551 and the fuel rods in bundle 0074 are shown in Figure 1. The cladding on fuel rods from both bundles was covered with an adherent black oxide and a light grey deposit, which could easily be wiped off with a paper towel. Many scratches and handling marks were evident on the cladding surfaces.

A steel lifting cable attached to bundle 0074 (not a part of the fuel bundle) had severely corroded during the sixteen years of storage. This corrosion resulted in a reddish deposit on the fuel rods and tube sheets, which also was easily removed with a paper towel. The possibility of enhanced hydrogen pickup in the Zircaloy due to a galvanic couple with the corroding cable was checked by comparing the hydrogen content of a tube sheet sample in contact with the cable to two adjacent samples taken outside the contact area. The results showed no significant differences in hydrogen content among the three samples. Also, metallographic examination of the tube sheet at the contact area showed no evidence of galvanic corrosion or hydriding.

Eight fuel rods were selected for detailed examination and were separated from the other rods with a circular saw in a milling machine. The information obtained from each type of examination is summarized below.

- Visual Examination: A detailed visual examination of each rod was made after the right rods had been separated from the bundles. No evidence of localized corrosion or defective cladding was seen on any of the fuel rods. The thin deposits on the surfaces could easily be wiped off with a paper towel, revealing an adherent black oxide. The characteristics of the oxide were the same as those reported at the time of discharge for fuel rods having similar irradiation exposures. Also, the tube sheet surfaces of Bundle 0551 that were produced by cutting during the hot cell examination in 1960 showed no evidence of reaction with the water environment over the nearly 21-year span of water storage (Figure 2).

(a) Bundle 0551 - Appearance of 7-rod cluster after nearly 21 years of pool storage (Neg. No. C8312)

(b) Bundle 0551 - Appearance of 8-rod cluster after nearly 21 years of pool storage (Neg. No. C8309)

(c) Bundle 0074 - Appearance of 120-rod bundle after 16 years of pool storage (Neg. No. C8519)

FIGURE 1. The General Appearance of Zircaloy-Clad Fuel Rods in Two Shippingport Fuel Bundles After Extended Water Storage

- **Axial Gamma Scanning:** The results from the axial gamma scans showed no unusual or unexpected behavior. The activity in all of the rods was quite low, with the 0551 rods being only slightly above the background of the hot cell.
- **Eddy-Current Testing:** The eddy-current examinations showed no strong indications of defective cladding. Weak signal distortions, numbering from two to six, were observed in seven of the eight rods examined. Visual examination of the fuel rods showed that the majority (over 80%) of these distortions were associated with scratches or marks produced by handling. Transverse metallographic sections were taken at some of the remaining locations but no definite cause for the signal distortions could be identified, which suggested that they were not associated with major defects.
- **Profilometry:** The outside diameters along four of the fuel rods were measured by spiral profilometry. The average diameters were 10.44 and 10.41 mm (0.411 and 0.410 in.), respectively, for the low and high burnup rods. These average values are within the original manufacturing specifications of 10.44 ± 0.05 mm (0.411 $\pm$ 0.002 in.). The maximum ovality for the low burnup rods was 0.08 mm (0.003 in.), whereas 0.15 mm (0.006 in.) was the maximum ovality in the high burnup rods.

FIGURE 2. Appearance of Tube Sheet Surfaces on Shippingport Fuel Bundle 0551 [Cut in 1960 and Exposed to the Water Environment in the Expanded Core Facility (ECF) for Nearly 21 Years (Neg. No. C8297)]

- **Leak Testing and Fission Gas Release:** The gases inside the fuel rods were collected and measured by drilling through one fuel rod end cap and measuring the pressure rise in an evacuated chamber of known volume. The fission gas releases estimated from these measurements ranged from 0.2 to 0.5% of the total fission gas generated in the rods for the low burnup rods and from 0.3 to 0.9% for the high burnup rods. Fuel rod integrity was determined by: 1) evacuating the fuel rod and measuring the pressure change during the gas collection operation, 2) pressurizing the fuel rods with helium to 0.28 MPa (40 psi) and measuring the pressure change as a function of time, and 3) analyzing the gases collected from the fuel rods. No evidence of defective fuel cladding was observed by evacuating or pressurizing the eight fuel rods, but water vapor and/or air was detected in several gas samples. The air and water vapor are believed to have been introduced from external sources during the drilling operations and/or leakage into the vials following collections and prior to the analyses. The small gas volumes in these rods make them extremely sensitive to small leaks.
- **Burst Testing:** Two fuel rods from each bundle were tested using the same procedures as had been used in the previous investigations. This process involved slowly pressurizing the fuel rods with water through a small hole drilled in one fuel rod end cap. The burst pressures measured in the four rods ranged from 99.3 to 105.5 MPa (14,400 to 15,300 psi) and were independent of burnup.
- **Burnup Analysis:** Samples for burnup analysis were taken from the center of one fuel rod in each bundle. The results show burnups of 420 GJ/kgU (4900 Mwd/MTU) for the fuel rod in bundle 0551 and 3100 GJ/kgU (36,000 Mwd/MTU) for the fuel rod in bundle 0074.

- Hydrogen Analysis: Samples for determining the hydrogen content of the cladding were taken adjacent to each of the eight metallographic sections. The measured values ranged from 50 to 90 ppm and were independent of burnup.
- Metallography: Two fuel rods from each bundle were sectioned for metallographic examination. One transverse section near the rod center and either a longitudinal section through the bottom end cap or an additional transverse section were taken from each fuel rod. For each transverse section, up to five surfaces were examined by progressive grinding and polishing.

Figure 3 shows typical oxide films observed on fuel rods from bundles 0551 and 0074. The measured thickness of the oxide films from the eight metallographic sections ranged from 0.7 to 2.8 μm for bundle 0551 and 1.3 to 3.3 μm for bundle 0074. These measurements agreed very well with those taken from similar rods immediately after discharge from the reactor and thus, no significant changes found in the external oxide films and caused by extended water storage were evident.

The internal cladding surfaces of fuel rods from bundle 0551 showed no measurable oxide films or bonding between the fuel and cladding. As shown in Figure 4, patches of oxide and fuel-cladding bonding were observed on the internal surfaces of the fuel rods from bundle 0074. Because the condition of the internal cladding surfaces at comparable exposures after discharge was not reported, direct comparison with present work is not possible. However, patches of a grey phase at the internal surfaces were reported for fuel rods having less exposure (Lynam 1964), which is consistent with current observations.

Although special attention was given to finding any defects or areas showing cladding degradation, very few specific areas could be identified. Figure 5 shows what appears to be a small crack at the internal surface of one high burnup rods from bundle 0074. This rod also exhibited more fuel-cladding bonding and the internal cladding surface was rougher than that on any of the other fuel rods examined. This apparent crack represents the worst defect found in any of the rods and extends through only 5% of the cladding.

Table 2 compares the results of the current examinations to the results obtained from similar Shippingport fuel after reactor discharge.^(a) All data were obtained from blanket fuel rods that were included in the initial Core 1 loading. Therefore, the results of the previous examinations show the effects of reactor exposure, whereas the effects of pool storage are seen by comparing the present results to those obtained at the end of the first and fourth seeds.

Examination of Table 2 data shows that during storage no measurable or significant changes occurred in several areas:

- external oxide thickness on the cladding exterior

(a) Henke 1966; Hillner 1974; Larson 1960; Lynam 1961; and Lynam et al. 1962
Lynam 1963; Lynam 1964; Rubin 1961a; Rubin 1961b; Rubin and Lynam 1966;
Sphar 1962.

Bundle 0551 (Neg. No. HC50115)

Bundle 0074 (Neg. No. HC50104)

FIGURE 3. Comparison of Oxide Films on Fuel Rods from Shippingport Fuel Bundles 0551 and 0074 After Extended Water Storage (nearly 21 years and 16 years of pool storage, respectively)

- hydrogen content of the cladding
- burst strength of the cladding
- diameter or ovality of the fuel rods
- fission gas release.

In addition, the results of the detailed visual examinations, eddy current tests, and metallography have not shown any evidence of cladding degradation caused by the extended water storage.

Rod 110 (Neg. No. HC50214)

Rod 120 (Neg. No. HC50209)

FIGURE 4. Fuel-Cladding Bonding in Fuel Rods from Bundle 0074

In summary, the results from profilometry, burst testing, fission gas release, hydrogen analysis, metallography, and visual examination compared favorably with those obtained immediately after discharge from the reactor. These results indicate that storage of Zircaloy-clad fuels is a safe and viable option. Absence of detectable degradation after two decades of water storage suggests that water storage of Zircaloy-clad fuel for four or five decades is an excellent prospect.

CONNECTICUT YANKEE FUEL ASSEMBLY EXAMINATION

A search was made to locate fuel assemblies with stainless steel-clad fuel rods that displayed certain characteristics: 1) had attained high burnup, 2) had been stored for a significant time in water, 3) had been examined on prior occasions, and 4) were characterized (e.g.,

FIGURE 5. Apparent Crack at Internal Cladding Surface in Rod 120, Bundle 0074 (Neg. No. HC 50246)

preirradiation, interim, and postirradiation examination data available). As of 1979, the highest burnups for fuel assemblies with stainless steel-clad fuel rods discharged from domestic and foreign commercial reactors were 3240 GJ/kgU (37,500 MWd/MTU)^(a) and 2765 GJ/kgU (32,000 MWd/MTU), respectively (Johnson et al. 1980). Some domestic fuel with batch-average burnups to the former level was discharged during the 1972-1976 period. Irradiated experimental stainless steel-clad fuel has been stored in water for as long as 12 years, while domestic commercial fuel assemblies with stainless steel-clad fuel rods have been stored up to 10 years (Johnson 1977).

Connecticut Yankee (Haddam Neck) Qualification Fuel Assembly S004 was selected from three candidate qualification assemblies for this program because it had the highest burnup. Standard commercial fuel assemblies stored at the Connecticut Yankee spent fuel pool have higher burnups [to 3180 GJ/kgU (36,800 MWd/MTU)] or longer times (to 10 years versus 5 years for S004) in storage in the boric acid pool water, but were not characterized to the same detail as S004.

The United States' examination program will eventually include results from a few stainless steel-clad (Type 304) fuel rods from each of two other Connecticut Yankee standard commercial fuel assemblies that have higher burnups but shorter storage times than S004. The cladding on the S004 fuel rods is Type 304L stainless steel.

Selection of Fuel Rods for Nondestructive Examination

Rod selection was based on three criteria:

- rod power (calculated)
- fuel burnup (calculated)
- rod precharacterization.

The examination's objectives were to characterize the range of fuel conditions that exist in the assembly and to determine if any significant change had occurred in the precharacterized condition. The assembly contained fuel rods with a range of rod powers and fuel burnups, which can be categorized into three types:

(a) Batch-average.

TABLE 2. Comparison of Results from Hot Cell Examinations of Zircaloy-Clad Shippingport PWR Core 1 Blanket Fuel Rods

Examination	(a) Discharged from Reactor, Year	Pool Storage, Years	Fuel Rod Diameter, mm	Maximum Ovality, mm	Burst Pressure, MPa	Fission Gas Release, %	Hydrogen Content of Cladding, ppm	Oxide Film Thickness, m
End of First Seed(b)	1959	--	10.41 to 10.64	0.02 to 0.09	91 to 102	0.2 to 0.7	49 to 101	1.3 to 2.5
Present Work (Fuel Bundle 0551)	1959	nearly 21	10.44 & 10.44	0.08 & 0.10	102 & 105	0.2 & 0.5	53 to 86	0.7 to 2.8
End of Second Seed	1961	--	10.43 & 10.47	0.04 to 0.12	91 to 114	0.2 to 0.6	38 to 63	0.6 to 1.3
End of Third Seed	1962	--	10.33 & 10.39(c)	0.03 & 0.06	--	0.8 & 0.9	57 to 73	1.2 to 2.5
End of Fourth Seed	1964	--	--	--	98 & 102(d)	--	52 to 93	1.8 to 3.3
Present Work (Fuel Bundle 0074)	1964	16	10.42 & 10.42	0.10 & 0.15	99 & 103	0.3 & 0.8	58 to 89	1.0 to 3.3
MELBA I(e)	1969	--	--	--	--	--	108 to 199	7.4 to 11.9(f)
MELBA II	1974	4	--	--	--	--	188 to 454	7.8 to 19.6(f)

(a) Core 1 started operation December 1957.

(b) The 1959 examination involved fuel rods from Fuel Bundle 0551; rods from the same bundle were examined in 1980.

(c) Small diameters attributed for final pickling operation (WAP0-TM-433) (Lynam 1964)

(d) Unpublished data.

(e) Multipurpose Extended Life Blanket Assembly.

(f) Averaged values from metallographic sections.

1. fuel rods that operated at relatively high powers through all three irradiation cycles, accumulating high fuel burnup.
2. fuel rods that operated at relatively low powers through all three irradiation cycles, accumulating relatively low fuel burnup.
3. fuel rods that operated at relatively low power in the first irradiation cycle, followed by a step increase to high power in the subsequent cycle and continuing at high power for the duration of the irradiation, accumulating relatively high fuel burnup.

Eight fuel rods, which included at least one representative of each of the three types and some of the precharacterized rods, were selected for nondestructive examination. Two of these eight were selected for further destructive examination later.

Fuel Characteristics

Gulf General Atomic contracted with British Nuclear Fuels Limited (BNFL) for the design and fabrication of the fuel assembly. The fuel rods contained UO_2 fuel pellets enriched to 4.0% ^{235}U by weight, in 10.76 mm (0.4235 in.) OD, seam-welded 304L stainless steel cladding with a 0.42 mm (0.0165 in.) wall thickness. The fuel rod overall length was 3.2172 m (126.66 in.). The fill gas composition was greater than 95% He plus 0.1-1.5 $\mu\text{Ci}/\text{cm}^3$ of ^{85}Kr ; the pressure was one atmosphere. The control rod guide tubes and the instrument tube were fabricated from 12.2 mm (0.480 in.) OD, 304L stainless steel tubing with a 0.34 mm (0.0135 in.) wall thickness. The grid spacers and springs were fabricated from Inconel 718, and the upper and lower nozzles were fabricated from 321 stainless steel. The upper nozzle was not designed as a removable nozzle. Additional design parameters are listed in Table 3.

Reactor, Shipping and Storage Conditions

Table 4 summarizes the examination history of S004. S004 attained a burnup of 2778 GJ/kgU (32,151 MWd/MTU). When S004 was readied for shipment to the hot cell, it had been stored in water in the spent fuel storage pool for 60 months. The S004 fuel assembly was shipped dry on May 14, 1980, and arrived at the hot cell facility at BCL on May 15, 1980.

Scope and Results of Hot Cell Examination

The objective of the hot cell examination was to characterize spent fuel that has been (or will be) in pool storage for a long time. The fuel was characterized to the extent that any significant changes in the fuel condition caused by extended pool storage would be discernible during interim examinations. Data from other fuel assemblies containing stainless steel-clad fuel rods from the same PWR will provide additional information. Some stainless steel-clad fuel rods from other Connecticut Yankee fuel assemblies may be reconstituted with those of S004 to compare effects of extended pool storage.

The nondestructive examination of S004 included visual inspection and sipping. The non-destructive examination of the stainless steel-clad fuel rods included visual inspection, profilometry, gamma scanning, eddy-current testing, and weighing. The destructive examinations on two fuel rods involved fission gas analysis, cladding metallography, cladding mechanical property determinations, and fuel studies (e.g., burnup analyses, density determinations, ceramography).

TABLE 3. Fabrication Data for Connecticut Yankee (Haddam Neck)
Qualification Fuel Assembly S004

Fuel Vendor	Gulf General Atomic
Fuel Designer and Fabricator	British Nuclear Fuels Limited
Type (Rod Array)	15 x 15
Guide Tubes:	
Number	20
Upper OD	12.2 mm (0.480 in.)
Wall Thickness	0.34 mm (0.0135 in.)
Material	304L Stainless Steel
Instrument Tubes:	
Number	1
OD	12.2 mm (0.480 in.)
Wall Thickness	0.34 mm (0.0135 in.)
Material	304L Stainless Steel
Spacer Grids:	
Number	7
Material	Inconel 718
Spring Material	Inconel 718
Fuel Rods (not prepressurized):	
Number	204
Length	3.2172 m (126.66 in.)
OD	10.76 mm (0.4235 in.)
Wall Thickness	0.42 mm (0.0165 in.)
Material	304L Stainless Steel
Fuel Length	3.0798-3.0925 m (121.25-121.75 in.)
Orifice Plate Material	321 Stainless Steel
Plenum Springs:	
Working Length	103 mm (4.06 in.)
Material	Inconel 718
Fuel Pellet:	
Geometry	Solid Right Circular Cylinder with Dished Ends
Diameter	9.73-9.75 mm (0.383-0.384 in.)
Material	UO ₂
Enrichment	4.00 wt% ²³⁵ U
Density	10.215 g/cm ³ (93.2% of theoretical density)
Weight/Rod	2.264 kg (4.987 lb)
Fuel Rod Identification	Fuel rod identification letters are marked on bottom end cap (e.g., two of the fuel rods are AHR and ABG)

TABLE 4. Examination of Connecticut Yankee Qualification Fuel Assembly S004

Examination	Time	Location	Information
Preirradiation	1971, Prior to shipment	British Nuclear Fuels Ltd., England	Fuel Weight, Fuel Stack Length, Rod Diameter
Interim	1972, End of Cycle 3(a)	Poolside, Connecticut Yankee Reactor	Rod Length, Rod Diameter, Video-tape Visual
Discharge	1975, End of Cycle 5	Poolside, Connecticut Yankee Reactor	
Preshipment	1980, April/May	Poolside, Connecticut Yankee Reactor	Periscope Visual
Postirradiation	1980, July	Battelle-Columbus Laboratories, Ohio	Visual, etc.(b)

(a) End of Cycle 3 operation in the Connecticut Yankee (Haddam Neck) plant.

(b) See Table 5.

The S004 fuel assembly underwent a poolside (underwater periscope) examination at the Connecticut Yankee spent fuel storage pool prior to being shipped to the hot cell facility at BCL for postirradiation examination. A typical view of the fuel rods is shown in Figure 6. Several anomalies were observed; for example, Figure 7 shows a fuel rod with a small elliptical black spot. Results from this examination will be discussed later.

When fuel assembly S004 arrived at BCL, the shipping cask underwent a wet sipping test prior to fuel removal. The sipping test results were negative, which indicated that the fuel rods did not release fission products during shipping. The cask and fuel were transferred into the BCL fuel pool on May 16, and the fuel assembly was removed from the cask.

A video system was used to examine the fuel assembly in the BCL pool. The fuel rods appeared clean; no obvious assembly distortion was seen. The system image was of limited clarity and none of the anomalies of the previous periscope examination were observed.

The fuel assembly's top nozzle (Figure 8) was visually examined dry prior to removal from the assembly. Some orange colored spots that were observed near the tops of some guide tubes may have been dried water spots. The top nozzle was removed by cutting off the other guide tubes just below the nozzle and drilling out the guide tube-to-nozzle welds on four guide tubes (Figure 8). Eight fuel rods were removed for nondestructive examination (see Table 5); two of the eight were selected for destructive examination. The principal criterion for selection was calculated fuel burnup. Rod burnups ranged from a high of 2978 GJ/kgU (34,470 MWd/MTU) to a low of 2139 GJ/kgU (24,750 MWd/MTU). The information obtained from each type of examination is summarized below.

- Weight: No significant weight change occurred. A weight change could have been caused by water ingress or fuel loss.

TABLE 5. Postirradiation Examination Results for Connecticut Yankee
(Haddam Neck) Qualification Fuel Assembly S004

Type of Examination	Number of Examinations	Examination Result Survey
• Fuel Assembly		
Nondestructive:		
Shipping	1 assembly	Negative; no fission products released during shipping.
Visual	1 assembly	Clean; no obvious distortion.
• Fuel Rods		
Nondestructive:		
Weight	8 rods	No significant change.
Gamma Scan	5 rods	No axial gaps in the fuel column or significant axial migration of ^{137}Cs .
Visual	8 rods	Nothing unusual; generally clean. No defects. Possible thin layer of crud or oxide. Axial scratches (from handling) on fuel rod exterior surfaces. Some burnished spots evident at spacer grid spring-to-fuel rod cladding contact areas.
Eddy Current	5 rods	No strong indications.
Profilometry	7 rods	Ovality, 0.30-0.36 mm (12 to 14 mils), detected at top and bottom ends of rods. Maximum ovality of up to 0.46 mm (18 mils) was detected. Less ovality noted in the middle region of the fuel rods. Cladding creep-down was up to about 0.03 mm (about 1 mil) on rod ABG and up to about 0.04 mm (about 1.5 mils) on rod AHR; most creep-down occurred in the upper half of the rods. No obvious indication of cladding defects.
Destructive:		
Fission Gas	2 rods	Hydrogen present in both rods; fission gas release relatively high in the high burnup rod (AHR) in comparison to that in the low burnup rod (ABG).
Void Volume	2 rods	Nothing unusual.
Cladding		
Metallography	2 rods	No crud or oxide layer observed.
Mechanical Properties		
Tensile Test (Longitudinal)	1 rod 6 samples	Reasonable ductility, some irradiation strengthening.
Ring Crush Test (Transverse)	1 rod 6 samples	Rings show ductile behavior.
Fuel		
Burnup	1 rod 1 sample	Result of chemical fuel burnup determination agreed well with calculated burnup.
Density	1 rod 4 samples	Samples from lower end of fuel column showed over 2% densification, but exhibited less swelling than fuel from near the center of the rod.
Ceramography	2 rods 7 samples	---
Autoradiographs	1 rod 1 sample	Fission products located near specimen OD.

FIGURE 6. Periscope Photograph Illustrating Typical Appearance of Fuel Rods in Fuel Assembly S004 at Connecticut Yankee During Preshipment Poolside Examination (Neg. No. 43834-11)

FIGURE 7. Small Elliptical Black Spot Observed on Fuel Rod in Fuel Assembly S004 at Connecticut Yankee During Preshipment Poolside Examination (Neg. No. 43835-3)

FIGURE 8. Fuel Rod and Guide Tube Locations in Top Nozzle in Connecticut Yankee Qualification Fuel Assembly S004. (Specific locations of fuel rods selected for examination and guide tubes with tube-to-nozzle welds drilled out are noted.)

- Gamma Scan: No significant axial fuel gaps or axial migration of ^{137}Cs occurred. The gamma scan trace for ^{137}Cs for four of the five rods examined showed regular sharp dips in intensity at fuel pellet interface locations. The scan of the high burnup rod showed much less pronounced intensity decreases at pellet interfaces.
- Visual Examination: Fuel rod surfaces, as viewed through an underwater periscope, were golden brown, indicative of an oxide film. Later metallurgical examinations indicated that surface deposits were very thin. The fuel rod end cap weld showed no indication of exterior corrosive attack either in the visual examinations (Figure 9a and b) or in the metallographic examinations (Figure 10a and b). Generally, the fuel rods were clean. Axial scratches were evident from abrasion at spacer contacts when the fuel rods were removed from the assembly, and some burnished spots at points of fuel rod cladding-grid spacer spring contact were observed (see Figure 9). None of the anomalies observed during the preshipment periscope examination were noted. The fuel rods were dry during the visual examination. Drying out the cladding surface possibly could have altered the appearance of stains that were evident under wet conditions. No cladding defects were noted.
- Eddy Current: An encircling coil was used for the eddy-current examinations. No strong indications were observed. A weak indication was observed 2.502 m (98.5 in.) above the bottom end of the high burnup fuel rod. Defect indications from traces of the vertical and horizontal components of the eddy-current signal and the differential of the horizontal component were qualitative. In general an indication only suggests the possibility of a cladding defect at a given axial location. Indications from other complementary examinations can then be used to determine the presence or absence of a cladding defect at that location.

(a) Bottom End Cap (Neg. No. C8498)

(b) Top End Cap (Neg. No. C8487)

FIGURE 9. Bottom and Top End Caps and End Cap Weld Areas from Stainless Steel-Clad Fuel Rods from Connecticut Yankee Qualification Fuel Assembly S004 (Showing Machined Area on Bottom End Cap and Handling Marks)

Stainless Steel Backing Plate
for Metallographic Preparation

Mounting Material

(a) Longitudinal Cross Section of Bottom End Cap (Neg. No. HC 50135)

(b) Cladding-to-End Plug Weld Area (Neg. No. HC 50136)

FIGURE 10. Metallography of Bottom End-Cap Weld from Stainless Steel-Clad Fuel Rod from Connecticut Yankee Qualification Fuel Assembly S004

- **Profilometry:** Spiral profilometry results indicated that cladding ovality was typically 0.30-0.36 mm (12-14 mils) in the upper and lower regions of the rods and ranged to a maximum of about 0.46 mm (18 mils). Ovality was less in the center region of the rods. Cladding creep down was to 0.025 mm (1 mil) on the lower burnup rod and 0.038 mm (1-1/2 mil) on the high burnup rod. Most of the creep down occurred in the upper half of the rods. Linear (axial) profilometry traces were run at 0°, 45°, 90°, and 135° orientations on the high burnup rod in the region of the eddy-current indication. No obvious indications of cladding defects were noted in any of the profilometry traces.
- **Fission Gas:** The high and low burnup fuel rods were punctured for fission gas collection and analysis. Hydrogen was present in the collected gases of both rods, but none was measured in fuel rods from another similar assembly from the same reactor. The Xe/Kr ratios were 7.6 and 8.2 for the low and high burnup rods, respectively. The fission gas releases from the low burnup and high burnup rods were 0.2 and 8%, respectively. Discussions of the dependence of fission gas release on fuel temperature indicated that the fission gas release from these rods did not occur while they were in the pool (Bouffieux and De Meulemeester 1979).
- **Void Volume:** The void volumes determined for these rods were 22.55 cm³ for the high burnup rod and 24.71 cm³ for the low burnup rod.
- **Metallography:** At magnifications up to 600X the crud or oxide layers were not detectable on metallographic cross sections of the cladding. However, observation of very thin surface layers, such as the crud or oxide layers that could be present on this cladding, is strongly dependent on the sample preparation and the optical resolution limits of the metallograph. Sample preparation with specimen edge retention of the caliber required to show thin surface films is very difficult, especially when the work is being done in a hot cell.

Examination of cladding seam and end cap welds showed no evidence of intergranular corrosion or stress-corrosion cracking.
- **Mechanical Properties:** Longitudinal tensile tests were completed on six 13-cm (5-in.) lengths [5-cm (2-in. gauge length)] of cladding from the high burnup fuel rod at three cross-head speeds. Ring compression tests (side pressing) were completed on six 1.3-cm (1/2-in.) lengths of cladding from the same rod at the same cross-head speeds. The results from these tests are being analyzed.
- **Fuel Burnup:** A fuel burnup analysis yielded a value of 3495 GJ/kgU (40,410 MWd/MTU or 4.209 at.%) for a sample taken from the high burnup area of rod AHR. This value compares well with the peak burnup value calculated for rod AHR of 3578 GJ/kgU (41,368 MWd/MTU). The peak value is calculated by multiplying the calculated value for average burnup for rod AHR (2982 GJ/kgU or 34,473 MWd/MTU) by the peaking factor (1.2).
- **Fuel Density:** Two pairs of fuel density samples were measured. The pair of samples with low burnup (i.e., adjacent to the lower end cap) had an average fuel density of 10.44 g/cm³ (95.3% of theoretical density); the pair of samples with high burnup from near the core midplane had an average fuel density of 10.24 g/cm³ (93.5% of theoretical density).

- Ceramography: The fuel structure was examined in five transverse sections from rod AHR and two from rod ABG. Alpha and beta-gamma autoradiographs were made using the as-polished surface of the fuel section taken from just below the middle grid spacer.

Examination Results

Results from the examinations listed in Table 5 showed assembly S004 to be well-behaved. The interim examination at a fuel burnup of 700 GJ/kgU (8100 MWd/MTU) showed essentially no change in fuel rod length in the rods measured.

The preshipment poolside examination of S004 at a burnup of 2778 GJ/kgU (32,151 MWd/MTU) was performed using an underwater periscope at Connecticut Yankee. The assembly exhibited circular and elliptical crud deposit patterns. These deposits were much less obvious in the post-irradiation examination, which was performed at the BCL hot cell facility.

Other anomalies noted during the preshipment examination, but not observed during the postirradiation examination at the BCL hot cell facility, were a rod with a small elliptical black spot (possibly a hole) and two possible cladding cracks in another rod. Another rod appeared to be bulged. Subsequent spiral profilometry, eddy-current testing, and visual examinations at BCL could not confirm the presence of any bulged areas in that fuel rod.

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MECHANISM STUDIES

Intergranular stress corrosion cracking (IG SCC) discovered in a weld heat-affected zone (HAZ) of piping in the Spent Fuel Pool piping system at Three Mile Island (TMI) Unit No. 1 was evaluated by a series of analytical tests (Jones, Johnson and Giacobbe 1981). The IG nature of the cracks was identified by optical metallography and scanning electron microscopy, as shown in Figures 11(a) and (b), respectively. Electrochemical potentiokinetic reactivation (EPR) analysis and the observation of grain boundary precipitation of chromium carbides [Figure 11(c)] indicated that significant sensitization had occurred in the HAZ. Cracking was found to have initiated at the ID surface where weld preparation grinding caused considerable deformation and stress concentration sites for crack initiation. Energy Dispersive X-ray Analysis and Auger Electron Spectroscopy revealed sulfur and chlorine on the fracture surface. Sulfur was found to be enriched both in particles and uniformly on the surface.

Constant extension rate tests (CERT) were performed on specimens removed from the spent fuel pool pipe containing uncracked areas of the weld HAZ. IG SCC was conclusively identified only for a welded pipe specimen tested in a solution of 13,000 wt ppm boric acid (2,100 wt ppm B) and 15 wt ppm Cl^- at 90°F. Ductile failure was the predominant fracture mode in a specimen tested in the same solution but with the original ID and OD surface of the pipe removed and in a specimen with the original thickness but without Cl^- . Shallow intergranular cracks were observed on the inner diameter of a welded specimen tested in 13,000 wt ppm boric acid, but ductile failure was the predominant failure mode. Furnace sensitized pipe specimens (EPR values of 8 C/cm^2) showed no signs of intergranular fracture in tests with up to 50 ppm Cl^- in boric acid and with low oxygen concentration (<1 ppm) in boric acid. The details for these tests are given in Table 6.

The results given in Table 6 strongly suggest that crack initiation in Type 304 stainless steel in boric acid at 40°C is the rate determining step or at least a significant step in the fracture process. This conclusion is supported by Tests 2 and 3 in which intergranular fracture occurred in the CERT tests only when the original inner diameter surface was present. The furnace-sensitized pipe specimens may also support this conclusion; however, their low sensitization of 8 C/cm^2 as compared to the maximum value of 22.8 C/cm^2 for the welded specimens suggests that sensitization and not surface defects was a controlling factor in their failing by ductile rupture and not IG SCC.

The results of the TMI pipe cracking study suggest several factors that should be avoided in the use of Type 304 stainless steel in spent fuel pool systems: (1) carbon concentrations should be kept below 0.07 wt%; (2) weld preparation grinding should be done with care; (3) weld heat input should be minimized; and (4) system impurities such as chlorides and sodium thio-sulfate should be minimized.

An experimental program has been initiated to determine the parameters responsible for the IG SCC of Type 304 stainless steel in low-temperature water environments. This research will also define fabrication limits such as material chemistry, welding conditions and water chemistry for the present and future application of Type 304 stainless steel. To study material chemistry, seven heats of Type 304 stainless steel are being evaluated. Constant axial load experimental cells have been constructed to determine a material's susceptibility to IG SCC and standard ASTM corrosion cells have been set up to measure sensitization.

FIGURE 11. Cracking in Spent Fuel Pool Piping from Three Mile Island Unit No. 1: (a) Optical Micrograph Showing Through Wall Crack in HAZ, (b) Scanning Electron Micrograph of Fracture Surface Near Mid-Thickness, (c) Optical Micrograph Illustrating Grain Boundary Precipitation in the HAZ of the Pipe

A paper at the Corrosion 81 conference in Toronto (Divine, et al. 1981) summarizes corrosion tests that are under way at PNL in two series of spent fuel pool environments.

TABLE 6. Summary of CERT Parameters and Results

Test Number	Specimen Type	Test Solution	Temp. °F	Cross-Head Rate	Ultimate Stress MPa	Fracture Strain, %	Reduction of Area, %	Time of Test hrs	Fracture Mode
1	Full Thickness Welded(a)	Air Sat'd 13,000 ppm H ₃ BO ₃	90	10 ⁻³ mm/min (7 x 10 ⁻² in/s)	--	Test was not to Failure	--	285	IG Cracks on ID Surface 1-9 mm from Weld
2	Full Thickness Welded(a)	Air Sat'd 13,000 ppm H ₃ BO ₃ + 15 ppm Cl ⁻	90	10 ⁻³ mm/min	657	63	41	265	75% IG to 0.2 mm from ID Mixed IG and Ductile Remainder
3	Milled Welded(a)	Air Sat'd 13,000 ppm H ₃ BO ₃ + 15 ppm Cl ⁻	90	10 ⁻³ mm/min	554	53	63	222	Ductile
4	Full Thickness Welded(a)	Air Sat'd 13,000 ppm H ₃ BO ₃	90	10 ⁻³ mm/min	614	67	69	285	Ductile
5	Full Thickness Welded(a)	Deaerated 13,000 ppm H ₃ BO ₃ + 15 ppm Cl ⁻	90	3 x 10 ⁻³ mm/min	620	64	53	90	Mixed IG and Ductile
6	Furnace Sensitized Pipe(b) 620°C/40 min	Air Sat'd 13,000 ppm H ₃ BO ₃	90	10 ⁻³ mm/min	607	98	69	418	Ductile EC = + 180 ± 20 mV
7	Furnace Sensitized Pipe(b) 620°C/40 min	Air Sat'd 13,000 ppm H ₃ BO ₃	90	3 x 10 ⁻³ mm/min	616	--	75	145	Ductile
8	Furnace Sensitized Pipe(b) 62°C/40 min	Air Sat'd 13,000 ppm 50 ppm Cl ⁻	90	3 x 10 ⁻³ mm/min	624	109	78	143	Ductile

(a) Maximum EPR sensitization - 22.8 coulombs/cm².(b) EPR sensitization - 8 coulombs/cm².

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APPENDIX A

PNL SPENT FUEL POOL PRESENTATIONS AND PUBLICATIONS

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PNL SPENT FUEL POOL PRESENTATIONS AND PUBLICATIONS

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