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SPENT-FUEL BEHAVIOR IN WATER POOLS

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ABSTRACT

This paper deals with the major question, "How long can water reactor spent fuel be stored in water?", from the standpoint of fuel rod cladding integrity. Evidence to date (1980) from the United States and other countries on spent fuel with Zircaloy and stainless steel cladding is described. That evidence includes findings from theoretical studies, data from operating experience, and results from detailed examinations of irradiated fuel rods. Current efforts at the Pacific Northwest Laboratory under the Spent Fuel and Fuel Pool Component Integrity Program, which is sponsored by United States Department of Energy, are discussed. Hot cell examinations are currently underway in that program; initial results from two Shippingport fuel bundles (Zircaloy cladding) and one Connecticut Yankee fuel assembly (stainless steel cladding) are presented. The Shippingport fuel being examined is the world's oldest pool-stored Zircaloy-clad fuel.

INTRODUCTION

A major question in the United States, Canada and most other nuclear countries is: "How long can spent fuel be stored in water?" That question arises from the following circumstances:

- Nuclear fuel becomes spent (burned out) after producing power in a reactor for several (three to five) years.

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- After discharge, the fuel is radioactive and continues to generate some heat. Water is an attractive medium to shield from radiation and to dissipate the heat. Nearly all nuclear fuel has been discharged to water pools since nuclear reactors began operation in the 1940s.

- Until recently, the nuclear fuel was intended for reprocessing after several months (up to two years) in water. In 1977, a reprocessing moratorium in the United States deferred that option and left water storage as the only near-term fuel management option in some countries. In other countries, notably France, the Federal Republic of Germany (FRG), Japan, the United Kingdom and the Soviet Union, reprocessing is either underway or is being developed. However, water storage remains an important fuel management link in these countries as well.

- Dry storage concepts have been developed and are being demonstrated with irradiated fuel,[1] so they offeran option in the unlikely event that problems eventually develop with water storage.

- Permanent disposal of spent fuel is the subject of a major hearing in the United States.[2] However, the first licensing of a disposal facility probably would not occur before the year 2000.

Thus, water storage is the only currently licensed option in the United States and several other countries, and could remain the principal fuel management option for several decades.

This paper deals with the question posed at the beginning from the standpoint of water reactor fuel rod cladding integrity.

WATER REACTOR SPENT FUEL INTEGRITY - THE EVIDENCE TO DATE

Water reactor fuel has two types of cladding materials: Zircaloy[a] or stainless steel[b]. Stainless steel-clad fuel currently constitutes less than ten percent of the stored commerical light water reactor (LWR) fuel inventory in the United States and the percentage is even lower world-wide. To date, there have been no problems with storage of water reactor fuel in spent fuel pools and no evidence, either theoretical or actual, that the fuel cladding is degrading.

Theoretical Evidence

Because Zircaloy and stainless steel have been used extensively in nuclear reactors, their corrosion charactertistics have been studied extensively. Thus, there is a large data base for assessing the expected behavior of the cladding materials under spent fuel pool conditions.

(a) A zirconium-base alloy, nominal composition (wt %): Zr-1.5 Sn-0.2 Fe - 0.15 Cr-0.05 to 0.005 Ni.
(b) Either 304 or 348 alloys.

Several investigators have independently assessed the array of potential degradation mechanisms without finding a basis to expect fuel cladding degradation in pool storage.[3-9] The principal uncertainty involved stress corrosion cracking of stainless steel, and that aspect is being addressed in fuel examinations to be discussed later in the paper.

Operating Experience

The observations of spent fuel pool operators is important to the assessment of spent fuel integrity. Not all commercial LWR fuel is inspected regularly: typically, sipping is done at BWRs if on-line monitors have indicated that fuel failures are present and visual inspection is typically done at PWRs if the radioactivity in the effluents is high.[10] Fuel assemblies are handled individually, so there are occassional opportunities to focus attention on the visual appearance. Also, the fuel is visible through the water during storage. Spent fuel types that are subject to water corrosion have signaled that they were degrading by evolution of hydrogen from the reaction:

$$xM + yH_2O \rightleftharpoons M_xO_y + yH_2$$

where M is a metal atom and x and y are small whole numbers.

Magnesium-clad[a] gas reactor fuel and metallic uranium fuel with cladding defects are the most notable fuel types which signaled their own degradation, either by hydrogen generation as indicated above or by perceptible increases in pool water radiation levels. Gas release from inside the fuel rod would be another possible signal of cladding perforation.

To date (1980) no spent fuel pool operator has seen any evidence by visual inspection or radiation monitoring that commercial water reactor fuel is degrading in water storage, over storage times spanning up to nearly 21 years for Zircaloy-clad fuel and up to 12 years for stainless steel-clad fuel.

Detailed Spent Fuel Rod Examinations

The argument can yet be made that slow cladding degradation could be in progress, which would not be detected by the spent fuel pool operators or even by the most careful visual inspection. Investigations in several countries have addressed in detail the status of spent fuel cladding after water storage (Table 1).

The examinations include both nondestructive and metallurgical investigations. Several nondestructive techniques provide methods to inspect the irradiated fuel cladding for defects, including:

Profilometry	For measurement of cladding dimensions to detect local protuberances

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(a) Corrosion of the magnesium-clad fuel can be controlled if the storage water pH is controlled at ≥ 11.5 and purity levels (Cl^- and SO_4^{-2}) and sludge concentrations are also controlled.

Eddy-Current Testing	For detecting cracks in the cladding that penetrate part way or fully through the wall
Ultrasonic Testing	For detecting water in the fuel rod and the presence of fuel-cladding bonding and cladding defects
Gamma Scanning	For detecting unusual distributions of fuel and fission products

Nondestructive techniques can be applied underwater in a spent fuel pool or in air in a hot cell.

Metallurgical procedures involve cutting sections of cladding and fuel from a fuel rod. The sections are mounted and prepared for optical microscope examinations at magnifications up to 500 to 1000 diameters. Scanning electron microscopy and microprobe analyses can further define cladding characteristics and compositions. Corrosion films about one micrometer (<0.00004 in.) thick can be detected and characterized by these sensitive methods.

Zircaloy-Clad Fuel

Table 1 indicates that periodic nondestructive examinations are underway in the FRG, including both intact fuel rods and fuel rods with reactor-induced defects. In three examinations since 1975, there is no evidence that either the intact or defective fuel rods is degrading in water storage.

Both nondestructive and metallurgical examinations have been conducted on Zircaloy-clad fuel in the United Kingdom and in Canada after water storage (Table 1). In neither case was there even minor evidence that cladding degradation was occurring in water storage. The Canadian investigators suggested that storage of Zircaloy-clad fuel for at least 50 years is a good prospect.[11]

Other evidence attests to the excellent durability of irradiated Zircaloy-clad fuel:

- Zircaloy-clad fuel assemblies charged into the Canadian NPD Reactor in 1964 are still performing well.
- Zircaloy-clad fuel bundles[(a)] left in the Shippingport Reactor for 17 years (1959-1974) attained a burnup of about 3540 GJ/kgU (41,000 MWd/MTU). Hot cell examinations in 1978 indicated that the range of measured oxide film thicknesses was only 2-24μm.[12] Also, there was no evidence of other forms of significant cladding degradation, even with that severe exposure history.
- Irradiated Canadian Zircaloy-clad fuel assemblies, stored in water for up to nine years, were returned to the NPD Reactor and operated well at relatively high power levels.[7]

(a) Seven fuel bundles are stacked axially to a fuel assembly in the Shippingport Reactor.

Thus, Zircaloy-clad fuel has been subject to several detailed examinations specifically designed to define whether degradation is occurring during water storage. So far, the answer has been--in all cases--that pool-induced deterioration is undetectable.

Stainless Steel-Clad Water Reactor Fuel

In 1977, British investigators performed nondestructive and metallurgical examinations on a stainless steel-clad LWR fuel rod (Table 1). There was no evidence of degradation.[5]

Other Considerations

Compared to the aqueous reactor conditions, which the spent fuel already has endured, the pool environments are mild, both in terms of fuel rod cladding temperatures (20-50°C in the pool; 290-350°C in the reactor) and radiation levels (neutron fluxes, $\sim 10^5$ versus $\sim 10^{14}$ neutrons/cm^2-sec; gamma levels, $\sim 10^6$ to 10^4 versus $\sim 10^9$ R/hr). A higher oxygen level in the pool water is the principal difference between the pool and reactor coolant system compositions. There is no evidence in the examinations cited in Table 1 that oxygen is adversely affecting the fuel cladding. However, that question will continue to be addressed in spent fuel surveillance programs.

The current plan to store spent water reactor fuel in water has additional conservatisms. Numerous fuel rods could fail in a water pool without substantial impacts on the health and safety of the pool staff and with essentially zero effect on the public. In the very remote prospect that a major fuel failure mechanism is detected during surveillance, there are at least three options:

1. Encapsulate the fuel for further storage in water;
2. Remove the fuel to dry interim storage, either encapsulated or uncapsulated; or,
3. Package the fuel for placement in a geologic repository.

A fourth option, reprocessing, is underway or being developed in countries indicated in an earlier section. While reprocessing is not an immediate option in the United States, there is an unoperated plant, Barnwell, in the United States.

SPENT FUEL AND FUEL POOL COMPONENT INTEGRITY PROGRAM

The purpose of this program at the Pacific Northwest Laboratory (PNL) is to define the corrosion and metallurgical condition of pool-stored nuclear fuel and fuel equipment after extended water storage. The program is sponsored by the United States Department of Energy (DOE) under Contract DE-AC06-76 RLO 1830. PNL is operated for DOE by Battelle Memorial Institute. The objectives of this program are to develop and conduct a surveillance study on nuclear fuel stored in spent fuel pools to determine whether degradation of fuel cladding or fuel assembly fixtures is occurring, to examine selected spent fuel pool components for evidence of degradation, to monitor similar studies going on in other countries, and to cooperate in international information exchanges (e.g., BEFAST[a]). The program results will provide insights to fuel and pool equipment behavior of potential value in licensing and operating fuel storage pools.

During 1980, PNL initiated nondestructive and destructive examinations on the following irradiated fuel:

- One portion (15 rods) of a Shippingport fuel bundle with Zircaloy-clad fuel rods that has been stored in deionized water since 1959.
- One Shippingport fuel bundle with Zircaloy-clad fuel rods (120) that has been stored in deionized water since 1964.
- One Connecticut Yankee qualification fuel assembly with stainless steel-clad fuel rods (204) that has been stored in boric acid pool chemistry since 1975.

Preliminary results from the examinations performed to date are described below.

Examination of Zircaloy-Clad Fuel From Shippingport Reactor

A. Background

Fuel rods from two Shippingport PWR Core 1 blanket fuel bundles are currently being examined to assess the effects of extended water storage on Zircaloy-clad UO_2 fuel rods. The Core 1 fuel bundles consisted of 120 fuel rods approximately 260 mm (10.25 in.) long with an outside diameter of 10.4 mm (0.411 in.). Each rod contains 26 natural UO_2 pressed and sintered pellets. The fuel cladding is Zircaloy-2. The rods were welded to Zircaloy-2 tube sheets at each end to form a 132 mm (5.2 in.) square array. The overall dimensions of the fuel bundles were 132 by 132 by 260 mm (5.2 by 5.2 by 10.25 in.).

The Shippingport Atomic Power Station's Core 1 loading consisted of an enriched metallic uranium seed surrounded by a blanket region containing 791 of the fuel bundles described above. The first core started operation in December 1957 and operated until February 1964. During this period, the expendable seed bundles were replaced three times and selected blanket fuel bundles were removed for metallurgical examinations at each reloading. The results of these examinations are reported in References 13-23.

(a) The BEhavior of Fuel Assemblies in STorage (BEFAST) Project is under the Nuclear Energy Agency (NEA) of the Organization for Eonomic Cooperation and Development (OECD), which is based in Paris, France.

The present work concentrates on two of the original blanket fuel bundles. The first,Bundle No. 0551, was removed from the reactor during the first seed refueling, December 1959, after achieving an average burnup of approximately 346 GJ/kgU (4000 MWd/MTU). The assembly was then sent to the Expended Core Facility (ECF) where selected fuel rods were removed and destructively examined. The remaining fuel rods were stored in the ECF pool until July 1980. At this time, 15 of the rods in two linear clusters were shipped to the hot cells at Battelle Columbus Laboratories (BCL) Nuclear Materials Technology Facility for inclusion in the current examination program.

The second bundle currently being examined is No. 0074. This bundle was removed from the reactor at the end of the Core 1 operation, February 1964. The average burnup at discharge was estimated to be 1555 GJ/kgU (18,000 MWd/MTU). The bundle was examined visually and leak checked prior to storage in the ECF pool. Bundle 0074 was shipped to the BCL hot cells with the fuel rods from Bundle 0551.

B. Current Examinations

The primary purpose of the experimental program is to assess the effects of extended water storage on Zircaloy-clad fuel rods. The Shippingport fuel rods are attractive for this purpose because of their long storage time, i.e., 16 years and nearly 21 years; also, detailed information is available regarding the condition of the fuel rods immediately after discharge. Finally, additional rods are available for future surveillance. The present experimental program was designed to establish the current condition of the fuel rods and to compare the present results to those obtained after reactor discharge.

Table 2 summarizes the experimental program for the Shippingport fuel rod examinations. Comparable results from the previous investigations are available for each type of examination, except gamma scanning and eddy current. These examinations were included in the current study to investigate cladding integrity and to establish a basis for future comparative examinations.

Eight rods, four from each bundle, are being examined in detail. The four rods from Bundle 0074 were taken from high flux positions in the bundle. Two rods were taken from each cluster of Bundle 0551. The end rods were removed from the seven-rod cluster while the two middle rods were removed from the eight-rod cluster.

Prior to rod removal, the exposed fuel rods in Bundle 0074 were given a visual examination. Figure 1 shows the general appearance of the fuel rods from each bundle when they arrived at the BCL hot cells. The cladding on fuel rods from both bundles were covered with an adherent black oxide and a light grey deposit, which wiped off readily. Numerous scratches and handling marks were also evident.

A wire lifting cable that was attached to Bundle 0074 was severely corroded, resulting in a reddish deposit on the associated fuel rods and tube sheets. This deposit also wiped off easily without leaving any visible evidence of an adverse effect on the underlying oxide.

The detailed examinations thus far completed on the eight individual rods have produced no evidence of an adverse effect from water storage on Zircaloy-clad fuel elements. The available information from each type of examination is summarized below.

Visual Examination: Detailed visual examinations of each of the eight fuel rods were made after they 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. In addition, the surfaces of the tube sheets in Bundle 0551 that were produced by cutting during the hot cell examination in 1960 showed no evidence of reaction with the water environment.

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 to try to identify the cause of the signal distortions. The results are not yet available.

Profilometry: The outside diameters along four of the fuel rods were measured by spiral profilometry. The average diameters were found to be 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, 10.44 ± 0.05 mm (0.411 ± 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. These values are slightly higher than measured after discharge, with the difference being most likely associated with the type of measurement used. The previous investigations measured the rod diameters optically at 0 and 90 degree orientations. The maximum ovality would not necessarily be obtained by this method.

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 range from 0.2 to 0.5 percent of the total fission gas generated in the rods for the low burnup rods and from 0.3 to 0.9 percent for the high burnup rods. These values compare favorably with the gas release measurements taken immediately after discharge from the reactor.

The integrity of the fuel rods was determined by: (1) evacuating the fuel rod and following 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. None of the measurements taken thus far have shown any evidence of a cladding defect in any of the eight fuel rods examined.

Burst Testing: Two fuel rods from each bundle were tested using the same procedures as had been used in the previous investigations. This 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. These values are within the range of the burst pressures reported from the previous investigations, which indicates that no serious degradation of the cladding has occurred during storage. However, detailed analyses of the burst data are not yet available and final determination of the effects of water storage on the mechanical properties should await this evaluation.

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. The metallographic examinations have not been completed but preliminary results from two transverse sections indicate no significant changes in the microstructures have occurred during water storage. The thickness of the oxide layers on the external surface of the cladding has remained constant and no significant difference in hydride distributions was observed in the cladding.

Hydrogen Analysis: No results available.

Burnup Analysis: No results available.

Details of the fuel examinations have been reported in another publication.[25]

C. Future Work

When completed, the current experimental program will have specifically addressed most of the potential degradation mechanisms that apply to extended water storage.

Upon completion of the hot cell examinations, the remaining fuel rods from the two bundles will be returned to a water storage pool for periodic surveillance and examinations. Results of the current examination will be correlated with those from the earlier examination of Bundle 0551 and the results from examinations of several other Shippingport fuel bundles.

Examination of Stainless Steel-Clad Fuel From Connecticut Yankee (Haddam Neck) Reactor

A. Background

Qualification Fuel Assembly S004, which is the subject of this portion of this paper, is from Connecticut Yankee (Haddam Neck) Reactor, a Westinghouse-designed pressurized water reactor (PWR) owned and operated originally by the Connecticut Yankee Atomic Power Company and now by the Northeast Utilities Service Company. The assembly was designed and fabricated by British Nuclear Fuels Limited for Gulf General Atomic (GGA). The assembly resided in the reactor core during Cycles 3, 4 and 5, a total irradiation time of 37 months (see Table 3). It was then discharged from the reactor and stored under water (contained ∿0.2 wt% boric acid) in the reactor spent fuel storage pool.

A qualification fuel assembly contains well-characterized fuel rods, a rather unique and desirable characteristic among fuel rods from commercial light water reactors (LWR). Twenty of the 204 fuel rods in S004 were precharacterized, i.e., the diameters of the preirradiated fuel rods were measured at three azimuthal locations at 305 mm (12 in.) intervals along the rod. The fuel stack weight and overall rod length were also measured.

The solid, right circular fuel pellets in the S004 fuel rods have dished ends. The as-fabricated pellet composition was UO_2 enriched with 4% ^{235}U. The pellets are contained in welded, Type 304 stainless steel tubes with a 10.76 mm (0.4235 in.) OD. The initial overall fuel rod length was 3.2172 m (126.66 in.). Table 4 lists pertinent fabrication data.

After being stored in the boric acid environment for 60 months, the fuel assembly was inspected at the reactor spent fuel pool and then shipped to the BCL hot cells for the postirradiation examination.

B. Current Examinations

The objective of the planned examinations is to adequately establish the present condition of the fuel qualitatively and, where possible, quantitatively so that the effects of irradiation and initial pool storage can be determined. It also provides a reference condition so that if significant change occurs during subsequent extended interim pool storage, it can be detected. Characterization here refers to the nondestructive and destructive tests which recorded and/or quantified selected chemical, physical or mechanical properties. The nondestructive examination addressed the fuel assembly as well as the individual fuel rods. Fuel assembly tests included sipping and visual examination. Fuel rod tests included: visual examination, profilometry, gamma scanning, eddy-current testing and weighing. All examinations have been or are being conducted at the BCL hot cells.

The destructive examination involved two individual fuel rods. These examinations include:

(a) For fuel rods: fission gas collection and analysis, void volume determination, and calculation of internal pressure prior to puncturing;

(b) For cladding: metallography (optical and scanning electron microscope) and mechanical property determinations; and,

(c) For fuel: burnup analysis, autoradiography, ceramography, density, leaching rate, and shielded electron microprobe analysis.

Visual and metallographic examinations of the fuel rods and fuel rod samples were conducted to characterize the cladding and any features that might be associated with fuel rod degradation. Characteristics such as crud deposition, cladding oxidation, fission product attack, pitting, stress-corrosion cracking, fuel pellet cracking, fuel grain size and cladding microstructure are being documented. Metallurgical features of particular interest, because they may be more sensitive to the effects of the storage environment, are the cladding longitudinal seam weld and the welds where the cladding joins the end caps. Fuel cladding mechanical property testing, which will include a typical tensile test, a D-ring tensile test and a ring compression test, will be conducted at three strain rates to characterize the cladding strength and resistance to cracking. Fuel leaching tests measure the rates of removal of radionuclides from fuel by water. These characteristics are important for defining spent fuel performance and the capability of irradiated fuel rods to maintain their integrity and retain radionuclides during extended storage in water.

For the selected fuel rods, the results to date include those from: visual examinations, gamma scanning, profilometry, eddy-current testing, weighing and fission gas analyses. The metallographic examination is in progress. No cladding cracks have been revealed by visual examination, eddy-current testing and metallography. There is no discernable crud layer or oxide film on the cladding at magnifications to about 500 diameters on metallographic sections (see Figure 2). Longitudinal scratches on the cladding (see Figure 3) were caused by the grid spacer contacts during removal of the rods from the fuel assembly. No evidence of unusual axial gaps between fuel pellets was observed in the gamma scans. Profilometry measurements indicate areas of large fuel rod ovality; however, that ovality does not appear to have influenced the cladding integrity.

Fission gas collection was the first destructive examination performed on the two S004 fuel rods. The S004 rod with the lower burnup, 2139 GJ/kgU (24,754 MWd/MTU), had an internal gas pressure of approximately two atmospheres (at 23°C), which is similar to the pressures in fuel rods with high burnups, 3171 GJ/kgU (36,700 MWd/MTU), from other comparable Connecticut Yankee fuel assemblies. The pressure in the S004 fuel rod with the higher burnup, 2970 GJ/kgU (34,370 MWd/MTU) was nearly seven times that in the lower burnup S004 rod.

Short-Term and Longer-Term Aspects of the Spent Fuel Surveillance Program

The short-term focus is to identify and acquire optimum spent fuel candidates for nondestructive and destructive examinations. Both intact fuel and fuel with reactor-induced defects are included in the negotiations. Emphasis is directed toward candidate fuel assemblies with the following desired characteristics:

- High burnups and/or extended pool residence;
- Prior examinations to define phenomena caused by the reactor exposure; and,
- Fuel assemblies that are available for periodic examinations for as long as fuel integrity surveillance is needed.

Prospects for obtaining selected high-burnup demonstration fuel for extended surveillance have been explored by discussions with utilities, nuclear fuel vendors, the Electric Power Research Institute, and DOE. Observations on such fuel would anticipate by several years unusual storage characteristics if they were to develop on commercial high burnup fuel inventories. To date, surveillance on fuel with burnups from 164GJ/kgU (1900 MWd/MTU) to 3370 GJ/kgU (39,000 MWd/MTU) has not indicated evidence that fuel cladding degradation is occurring.

After the candidate fuel assemblies are examined by visual and other nondestructive and destructive inspection techniques the fuel assemblies will be stored at sites typical of reactor spent fuel pools and Away-from-Reactor (AFR) storage facilities, where they are to be available for additional surveillance. Once an AFR storage facility is in place, the surveillance fuel assemblies will be stored there. The program includes plans for providing a surveillance capability at that facility. The longer-term aspects of the program comprise periodic surveillance, including visual inspections plus nondestructive and destructive examinations at five-year intervals for as long as the need exists to characterize the spent fuel behavior. In addition to surveillance on the designated fuel assemblies, other fuel assemblies in the AFR facility inventory will also be inspected after random selection.

SUMMARY AND CONCLUSIONS

For the United States and several other countries, water storage of spent water reactor fuel is the only currently licensed option--it could remain the principal fuel management option for several decades. Water reactor Zircaloy-clad and stainless steel-clad fuel have behaved well in water storage pools. As of 1980, the maximum storage experience is up to nearly 21 years with Zircaloy-clad fuel and up to 12 years with stainless steel-clad fuel. There has been no problem to date with storage of water reactor fuel in spent fuel pools and no evidence, either theoretical or actual, that the fuel cladding is degrading. Most of the evidence is based on visual observations and the absence of radiation releases. However, important information is available from nondestructive and destructive examinations of spent fuel. The results suggest that no perceptible degradation of fuel cladding has occurred. This favorable experience supports the expansion of spent fuel storage facility capacities and the extension of storage times in water for commercial water reactor fuel, provided that the surveillance programs that are beginning or are underway continue to confirm that fuel assembly degradation is within the range of acceptability.

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REFERENCES

1. P.A. Anderson and H.S. Meyer, Dry Storage of Spent Nuclear Fuel. NUREG/CR-1223, April 1980.

2. DOE/NE-0007 (NRC Confidence Hearing Testimony, April 15, 1980)

3. A.B. Johnson, Jr., Behavior of Spent Nuclear Fuel in Water Pool Storage. BNWL-2256, Pacific Northwest Laboratory, September 1977.

4. B.F. Warner, "The Storage in Water of Irradiated Oxide Fuel Elements," Testimony presented at Windscale Inquiry on Spent Fuel Reprocessing, Windscale, U.K. (October 1977).

5. R.H. Flowers, "Results of an Examination of Irradiated Oxide Fuel Following Storage in Water," Testimony presented at Windscale Inquiry on Spent Fuel Reprocessing, Windscale, U.K. (October 1977).

6. G. Vesterlund and T. Olsson, "Degraderingmeckanismer Vid Bassanglagering och Hantering av Utbrant Kraftreaktorbransle," RB 78-29, ASEA-ATOM, Vasteras, Sweden (to be published); English translation, BNWL-TR-320, Pacific Northwest Laboratory (May 19, 1978).

7. S.A. Mayman, "Canadian Experience with Wet and Dry Storage Concepts," presented at Executive Conf. Spent Fuel Policy and Its Implications, Buford, Georgia, April 2-5, 1978.

8. K.L. Huppert, "Spent Fuel Storage-Philosophies and Experience," Proc. NEA Sem. Storage of Spent Fuel Element, Madrid, Spain, June 20-23, 1978.

9. M. Peehs, W. Petri, H.P. Fuchs, and F. Schlemmer, "Behavior of Spent LWR Fuel Assemblies," Proc. NEA Sem. Storage of Spent Fuel Elements, Madrid, Spain, June 20-23, 1978.

10. W.J. Bailey, C.J. Morris, F.R. Reich and K.L. Swinth, Assessment of Current Onsite Inspection Techniques for Light-Water Reactor Fuel Systems. NUREG/CR-1380 (PNL-3325), Volume 1, Pacific Northwest Laboratory, March 1980.

11. C.E.L. Hunt, J.C. Wood and A.S. Bain, Long-Term Storage of Fuel in Water. AECL-6577, Atomic Energy of Canada Limited, Chalk River, Ontario, 1979.

12. E. Hillner, Corrosion and Hydriding Performance Evaluation of Three Zircaloy-2 Clad Fuel Assemblies After Continuous Exposure in PWR Cores 1 and 2 at Shippingport, PA. WAPD-TM-1412, Westinghouse Electric Corporation, January 1980.

13. L.D. Larson, "Dimensional Inspection of Irradiated PWR Core 1 Blanket Fuel," WAPD-TM-262, Westinghouse Electric Corporation, December 1960.

14. B. Rubin, "Fission Gas Release in PWR Core 1 Blanket Fuel Rods Upon Conclusion of Seed 1 Life," WAPD-TM-263, Westinghouse Electric Corporation, February 1961.

15. B. Rubin, "Examination of PWR Core 1 Blanket Fuel Rods for Microstructure, Hydrogen Pickup and Burst Strength," WAPD-TM-264, Westinghouse Electric Corporation, February 1961.

16. C.D. Sphar, "Evaluation of Isotopic Content of Irradiated Natural Uranium Dioxide Rods from PWR Core 1," WAPD-TM-265, February 1962.

17. L.R. Lynam, "Metallurgical Examination of Fuel Rod from PWR-1 Blanket at End of Second Seed Life," WAPD-TM-308, Westinghouse Electric Corporation, December 1961.

18. L.R. Lynam, E.F. Lozco, G.J. Salvaggio, and J.G. Goodwin, "Fabrication Defects Observed in PWR Core 1 Blanket Fuel Rods," WAPD-TM-321, Westinghouse Electric Corporation, September 1962.

19. L.R. Lynam, "Examination of PWR Core 1 Blanket Fuel Rod for Microstructural Changes, Hydrogen Pickup, Burst Strength and Fission Gas Release at the End of the Second Seed Life," WAPD-TM-326, Westinghouse Electric Corporation, April 1963.

20. L.R. Lynam, "Metallurgical Examination of PWR Core 1 Blanket Fuel Rods at the End of the Third Seed Life," WAPD-TM-433, Westinghouse Electric Corporation, August 1964.

21. R.L. Henke, ed., "Summary Design Report for Melba Test in PWR Core 2 Blanket," WAPD-TM-483, Westinghouse Electric Corporation, March 1966.

22. B. Rubin and L.R. Lynam, "Examination of Fuel and Cladding in PWR-1 Blanket Rods," Nuclear Applications, Volume 2, pp. 449-504, 1966.

23. E. Hillner, "Corrosion and Hydriding Performance of Zircaloy Tubing After Extended Exposure in the Shippingport Pressurized Water Reactor," Zirconium in Nuclear Applications. ASTM STP 551, American Society for Testing and Materials, 1974, pp. 449-462.

24. K.L. Huppert and G. Zimmerman, "Experience with Regard to Long-Term Storage of LWR Fuel Elements," Reactor Congress, Mannheim, March 29-April 1, 1977, AED-CONF-77-013-122, pp. 447-450 (in German).

25. A.B. Johnson, Jr., W.J. Bailey, E.R. Bradley, and D.C. Langstaff, "Significance of Shippingport and Connecticut Yankee Fuel Examinations to Extended Water Storage of LWR Fuel" (PNL-SA-8833), a paper for presentation at the IAEA Advisory Group Specialists' Meeting on Spent Fuel Storage Alternatives, Las Vegas, Nevada, November 17-21, 1980.

TABLE I

Surveillance Activities Underway on the Behavior of Spent Fuel in Water Storage

COUNTRY	FUEL ROD CLADDING MATERIAL	STUDIES UNDERWAY
Canada	Zircaloy	Nondestructive and destructive examinations on 143 fuel rods are planned on five-year intervals through 1995.[11] First examination completed-1978.
United Kingdom	Zircaloy; Stainless Steel[(a)]	British Nuclear Fuels (BNFL), Ltd., is conducting periodic examinations.[4,5] First examination completed-1977.
Federal Republic of Germany	Zircaloy	Kraftwerk Union (KWU) is conducting periodic nondestructive examinations of 28 fuel rods, ten of which are defective.[9] First examination-1975.
		Wiederaufarbeitungsanlage (WAK) is annually photographing one PWR fuel assembly.[24]
United States	Zircaloy; Stainless Steel	The long term aspects of the program at Pacific Northwest Laboratory (PNL) comprise periodic surveillance, including frequent visual inspections and nondestructive and destructive examinations at five-year intervals for as long as the need exists to characterize the spent fuel storage behavior. Currently (1980), three fuel assemblies are being examined.

NOTE: The BEFAST[(b)] Committee, functioning under OECD/NEA[(c)], meets periodically to review and communicate spent fuel behavior information from member status.

(a) One PWR stainless steel-clad fuel rod destructively examined.
(b) BEhavior of Fuel Assemblies in STorage (BEFAST)
(c) Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency (NEA), based in Paris, France.

TABLE 2

Summary of Experimental Program for
Shippingport Fuel Examinations

TYPE OF EXAMINATION	FUEL ASSEMBLY NO. 0551	FUEL ASSEMBLY 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 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 for Hydrogen	2 Rods (4 Samples)	2 Rods (4 Samples)

TABLE 3

Connecticut Yankee (Haddam Neck) Operating
Information During Cycles 3, 4 and 5

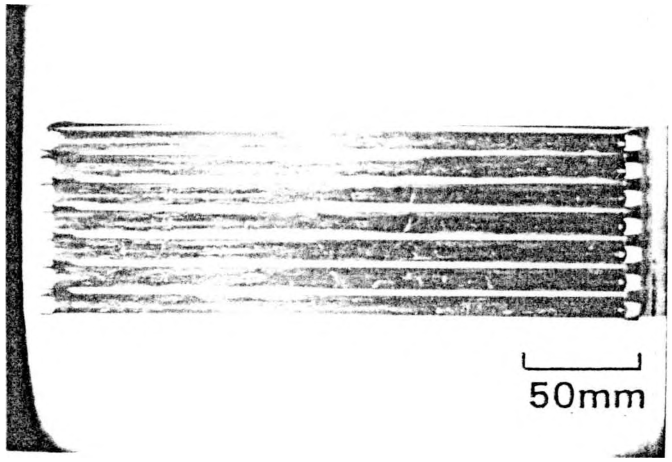
CYCLE NUMBER	STARTUP DATE	SHUTDOWN DATE	IRRADIATION TIME(EFFECTIVE FULL-POWER DAYS)
3	May 21, 1971	June 15, 1972	365
4	July 16, 1972	July 18, 1973	321
5	December 14, 1972	May 18, 1975	460

Note: Two other fuel assemblies (H07 and G11) with stainless steel-clad fuel rods irradiated in Connecticut Yankee (Haddam Neck) reactor are undergoing detailed examinations at the hot cell facility at Battelle Columbus Laboratories under a program sponsored jointly by the Northeast Utilities Service Company and the Electric Power Research Institute. The two assemblies were discharged at the end of Cycles 8 and 7, respectively. Results of these examinations eventually will be available for comparison with examination results from Qualification Fuel Assembly S004.

TABLE 4

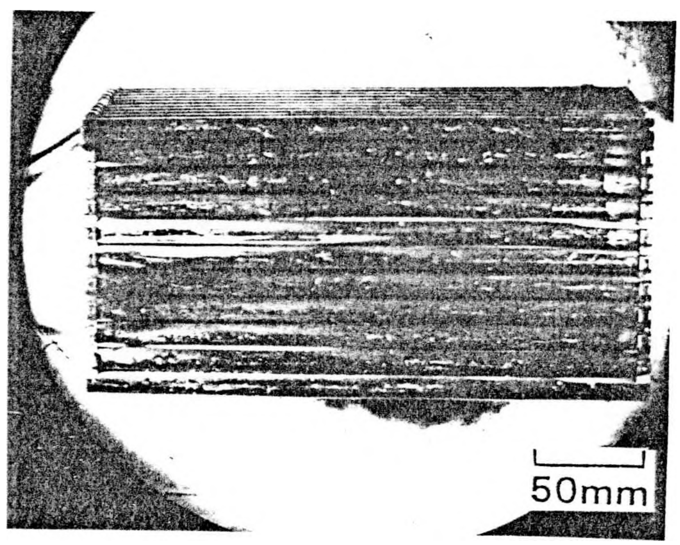
Fabrication Data for Connecticut Yankee (Haddam Neck)
Qualification Fuel Assembly S004

Fuel Vendor/Fuel Designer and Fabricator	Gulf General Atomic/British Nuclear Fuels Limited
Type (Rod Array)	15 x 15 (No Prepressurized Rods)
Fuel Rods	
• 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	Type 304L Stainless Steel
• Fuel Length	3.0798-3.0925 m (121.25-121.75 in.)
Fuel Pellet	
• Geometry	Solid Right Circular Cylinder (Dished Ends)
• Material	UO_2
• Enrichment	4.00 weight percent ^{235}U
• Density	10.215 g/cc
• Weight/Rod	2.264 kg (4.987 lb)



Neg. No. C8312

Fuel Bundle 0551 - Appearance of 7-rod cluster from that bundle after nearly 21 years of pool storage (1959-1980)



Neg. No. C8519

Fuel Bundle 0074 - Appearance after 16 years of pool storage (1964-1980)

FIGURE 1. Photographs Showing the General Appearance of Zircaloy-Clad Fuel Rods in Two Shippingport Fuel Bundles.

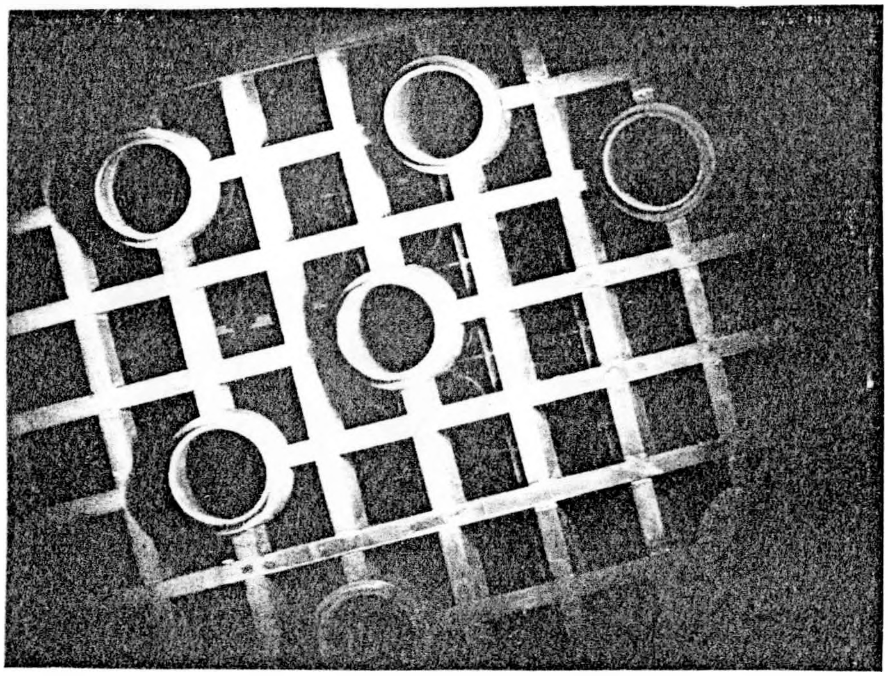


FIGURE 2. Photograph of Top Nozzle of Connecticut Yankee Fuel Assembly S004, Which Shows the Top End of the Control Rod Guide Tubes. Assembly End Fittings and Spacer Grids Had Good Integrity.

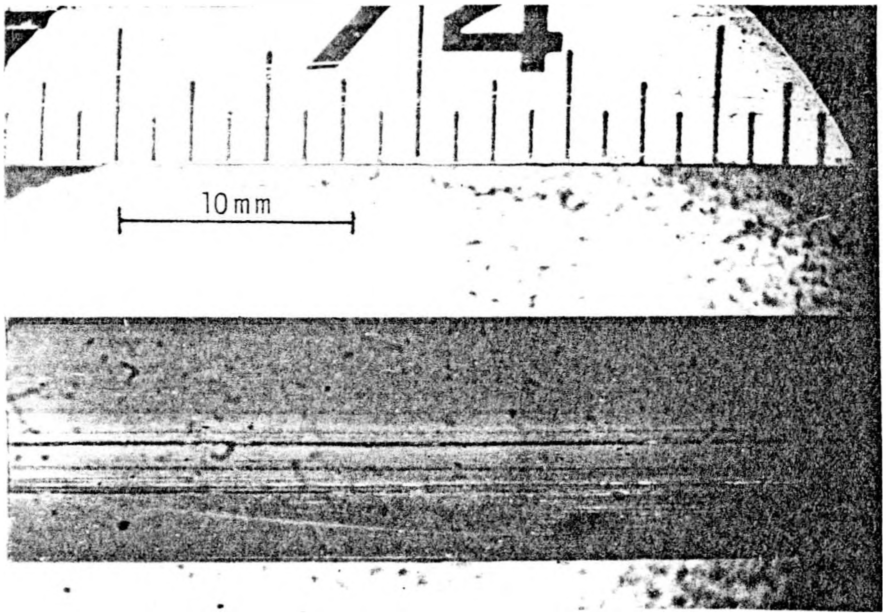


FIGURE 3. Photograph of Stainless Steel-Clad Pressurized Water Reactor (PWR) Fuel Rod from Connecticut Yankee Fuel Assembly S004. The Cladding Surface is Relatively Clean and is Free of Cracks. Longitudinal Scratches Were Caused by Fuel Rod-to-Spacer Grid Contact During Rod Removal.