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Annual Report - FY 1979
Spent Fuel and Fuel Pool Component Integrity

A. B. Johnson, Jr.
W. J. Bailey
R. E. Schreiber
F. M. Kustas

May 1980

Prepared for the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830

Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
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SPENT FUEL AND FUEL POOL COMPONENT INTEGRITY

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SUMMARY

The Spent Fuel and Fuel Pool Component Integrity Program, sponsored by the Department of Energy's (DOE) Division of Spent Fuel Storage and Transportation, comprises four tasks:

1) International Activities
2) Spent Fuel Examinations
3) Fuel Pool Component Examinations
4) Corrosion Mechanism Assessments.

This report summarizes activities conducted under the Program at Pacific Northwest Laboratory (PNL) during FY 1979 and updates information regarding spent fuel and fuel pool component behavior in pool storage.

International meetings under the BEFAST[a] program and under INFCE[b] Working Group No. 6 during 1978 and 1979 continue to indicate that no cases of fuel cladding degradation have developed on pool-stored fuel from water reactors.

Rumors regarding failure of fuel cladding during water storage have been traced and found not to apply to commercial water reactor fuel; rather they apply to the problem of magnesium-clad gas reactor fuel or Zircaloy-clad metallic uranium fuel. Similarly, rumors that extended storage might cause problems in reprocessing have been investigated and also do not have merit for commercial water reactor fuel.

A case involving a substantial radiation release from stainless-clad water reactor fuel has been discussed with Mr. G. LeFort at Fontenay-aux-Roses. The incident involved fuel that had developed some failures in-reactor. Fuel was shipped from the Chooz reactor to La Hague. Sampling of the cask environment prior to unloading the fuel at La Hague indicated relatively high radiation levels within the shipping case. The pool operators were able to reduce the

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(a) BEhavior of Fuel Assemblies in STorage, under the Nuclear Energy Agency of OECD.
(b) INternational Fuel Cycle Evaluation

radiation levels and unload the fuel. Shipments of other stainless-clad fuel from the same reactor and from several other water reactors have occurred without incident.

We have examined sections of stainless steel components from three PWR spent fuel pools that have boric acid water chemistry. A section from a spent fuel rack stand, exposed for 1-1/2 yr in the Yankee Rowe (PWR) pool had 0.001- to 0.003-in.-deep (25- to 75-μm) intergranular corrosion in weld heat-affected zones but no evidence of stress corrosion cracking.

A section of a 304 stainless steel spent fuel storage rack exposed 6-2/3 yr in the Point Beach reactor (PWR) spent fuel pool showed no significant corrosion.

A section of 304 stainless steel 8-in.-dia (20-cm) pipe from the Three Mile Island No. 1 (PWR) spent fuel pool heat exchanger plumbing developed a through-wall crack. The pipe was sent to PNL for analysis; it was one of seven pipes with through-wall cracks. The crack was intergranular, initiating from the inside surface in a weld heat-affected zone. The zone where the crack occurred was severely sensitized during field welding. Several factors which may have contributed to the pipe failure include weld procedure, carbon content, and stagnant operation of the pipe. Testing is underway to identify other factors, such as the chemical environment, which may be significant.

The Kraftwerk Union (Erlangen, GFR) disassembled a stainless-steel fuel-handling machine that operated for 12 yr in a PWR (boric acid) spent fuel pool. There was no evidence of deterioration, and the fuel-handling machine was reassembled for further use.

A spent fuel pool at a Swedish PWR was decontaminated. The procedure is outlined in this report.

CONTENTS

FIGURES

TABLES

INTRODUCTION

The Carter Administration moratorium on reprocessing of nuclear fuel requires that the nuclear industry rely on water storage as the only near-term option for spent fuel management. Whereas pool storage is an established technology, the reprocessing moratorium imposes longer storage times for the spent fuel than were originally foreseen. Verification of spent fuel integrity arose as an issue in the Windscale hearings in the United Kingdom (Parker 1978) and has been an issue in some spent fuel pool modification hearings for United States reactors. The International Fuel Cycle Evaluation (INFCE) Working Group No. 6 also has recommended that verification of spent fuel integrity and spent fuel surveillance be conducted over the time span that spent fuel is stored in water pools.

The Pacific Northwest Laboratory (PNL) is conducting a program to investigate the integrity of spent fuel and fuel pool components. This report summarizes activities of that program for Fiscal Year (FY) 1979 (October 1, 1978 to September 30, 1979).

The report is developed in the following sequence:

- Overview of the United States Spent Fuel and Fuel Pool Components Integrity Program, including FY-1979 activities.
- International activities, including current perspectives on spent fuel integrity and a summary of spent fuel surveillance activities underway in other countries.
- Summary of developments to identify optimum spent-fuel candidates for the United States program, including characterization of the United States spent fuel inventory, including burnup distributions and number of fuel assemblies with Zircaloy and stainless steel cladding.
- Summary of examination on sections from spent fuel pool components exposed to boric acid spent fuel pool water chemistry.

- Appendix A contains several tables summarizing characteristics of the current spent fuel inventory. Appendix B describes a case history of chemical decontamination of a Swedish pressurized water reactor spent fuel pool.

PROGRAM OUTLINE AND ACTIVITIES

The Spent Fuel and Fuel Pool Component Integrity Program at PNL is sponsored by the Department of Energy's (DOE) Division of Spent Fuel Storage and Transportation through the Spent Fuel Project Office at the Savannah River Operations Office, Aiken, South Carolina. The spent fuel project is administered by duPont de Nemours. The Spent Fuel and Fuel Pool Component Integrity Program has the following tasks:

1) Task 1 - Foreign Spent Fuel Storage Monitoring and Participation

   The task activities for FY 1979 include:

   - DOE representative to the BEFAST[a] Committee
   - technical support to the United States INFCE Working Group No. 6 delegation
   - correspondence and discussions with foreign spent fuel storage technologists
   - participation in an international survey of spent fuel pool experience; represented on the steering committee that will evaluate the survey results.

2) Task 2 - Spent Fuel Acquisition and Examination

   The principal FY-1979 activities on this task comprised:

   - final negotiations for access to spent fuel from the Shippingport reactor (PWR),[b] which is the oldest Zircaloy-clad spent fuel remaining in water storage

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(a) BEhavior of Fuel Assemblies in STorage, under the Nuclear Energy Agency/Organization for Economic Cooperation and Development (NEA/OECD), Paris, France

(b) Pressurized Water Reactor

- assessments of spent fuel inventories as a function of burnup for Zircaloy-clad and stainless-clad fuel
- discussions with utilities and nuclear fuel vendors to identify characteristics and current locations of candidates for spent fuel surveillance activities.

3) Task 3 - Examination of Selected Spent Fuel Pool Components

The task activities for FY-1979 included:

- examinations of the following materials from PWR spent fuel pools:
  - a section of a 304 stainless steel spent fuel rack support from the Yankee Rowe Reactor
  - a section of 304 stainless steel spent fuel rack from the Point Beach reactor
  - a section of 8-in.-(20.3-cm)-dia spent fuel pool piping from the Three Mile Island No. 1 fuel storage pool.
- negotiations for metal sections exposed in deionized water pools. We have preliminary concurrence to obtain specimens from equipment in the La Crosse (BWR)[a] and RBOF[b] spent fuel pools.

4) Task 4 - Corrosion Mechanism Assessments

Activities on this task included:

- attendance at the Light Water Reactor Fuel Performance, American Nuclear Society Topical Meeting, Portland, Oregon, April 29 to May 3, 1979, to update information regarding nuclear fuel behavior and characteristics.
- monitoring literature pertinent to fuel storage environmental regimes and materials.

(a) Boiling Water Reactor
(b) Receiving Basin for Off-site Fuel, Savannah River Plant, Aiken, South Carolina.

- initiation of a laboratory test to investigate aspects of corrosion on spent fuel pool materials.
- conducting selected examinations on spent fuel equipment (outlined under Task 3).

STATUS OF INTERNATIONAL SPENT FUEL STORAGE ACTIVITIES

Two international activities are summarized in this section: the International Fuel Cycle Evaluation (INFCE) Working Group No. 6 and the Behavior of Fuel Assemblies in Storage (BEFAST) program, currently operating under the Organization for Economic Cooperation and Development (OECD).

INFCE WORKING GROUP NO. 6

The INFCE study comprises eight international working groups conducting a comprehensive evaluation of the nuclear fuel cycle. Working Group No. 6 is entitled: Spent Fuel Storage and Handling. The Working Group has generated the Spent Fuel Management Report, January 1980. The report summarizes spent fuel characteristics, inventories, handling, storage procedures, transportation, and institutional management frameworks.

BEFAST PROGRAM

The BEFAST program was developed as a result of a 1977 proposal by Austria to the International Energy Agency (IEA), which is a component of Paris-based OECD. Subsequently, the BEFAST program was transferred to the Nuclear Energy Agency (NEA), also a component of OECD. The status of the BEFAST program will be finalized at a November 1979 NEA meeting in Paris.

The BEFAST program proposes international cooperation in assessing the integrity of spent fuel in pool storage. The original proposal includes the following elements:

- Task A - International survey of spent fuel storage experience, sponsored jointly with the International Atomic Energy Agency (IAEA)
- Task B - Technical Program
  - B.1 Cladding Integrity after Extending Storage and Surveillance Methods
  - B.2 Special Failure Mechanisms

B.3 Fission Product - Cladding Interactions

B.4 Behavior of Defective Fuel Elements

- Task C - Risk Assessment

At an April 1978 meeting in Vienna, a proposal was adopted to conduct the survey of international spent fuel storage experience. That survey is underway, and a questionnaire has been developed. The questionnaire was translated into several languages and was distributed by IAEA, with response requested by December 1979. When the questionnaires are returned, a steering committee will evaluate them and publish a report.

Representatives from nine nations, the NEA and the IAEA attended an April 1979 meeting in Paris. The national representatives were from Austria (BEFAST chairman), Belgium, Finland, France, the Federal Republic of Germany, Italy, the Netherlands, Sweden, and the United States. Switzerland has subsequently indicated an intention to participate in BEFAST. Other countries have indicated interest but have not made specific commitments.

Some pool storage surveillance and research are underway in most of the attending countries. At the April meeting, there was strong agreement among most participants that BEFAST provides an effective framework for information exchange regarding spent fuel surveillance activities. France, which is now reprocessing spent fuel, does not foresee the need for extended fuel storage. Therefore, the French representative did not indicate specific plans for French participation beyond an observer status. The German government representative also did not indicate specific plans for BEFAST participation, but indicated that the Federal Republic of Germany would concur with German industry participation. The Kraftwerk Union representative indicated an intention to participate and share information from programs now underway.

Eventually, cooperative programs may develop under BEFAST if a compelling need arises. Currently, the two near-term BEFAST activities are to interpret and publish the spent fuel experience survey and to provide a forum for timely exchange and discussion of information from the various national spent fuel surveillance programs.

CURRENT PERSPECTIVES ON SPENT FUEL AND FUEL POOL COMPONENT INTEGRITY

Several publications address topics related to spent fuel integrity, including the following: Canada (Mayman 1978; Hunt, Wood and Bain 1979; Walker 1979), France (LeFort and Pouit 1978), Germany (Huppert 1978; Huppert and Zimmerman 1977; Peehs, Petri, Fuchs and Schlemmer 1978), Sweden (Vesterlund and Olsson 1978), the United Kingdom (Parker 1978; Warner 1977; Flowers 1977) and the United States (Johnson 1977; Johnson 1978; Johnson 1979; Zima 1979). There is a concensus that nothing in current experience suggests that degradation is occurring on spent fuel cladding from water reactors during storage in water pools.

Several rumors have surfaced from time to time suggesting that difficulties with reprocessing would arise if fuel were stored longer than 5 yr. We have traced the source of the rumors and have discovered that they relate to reprocessing problems with Zircaloy cladding from metallic uranium fuel or to an explosion of finely divided zirconium from machining operations at Oak Ridge National Laboratory. Zircaloy cladding from oxide fuel gave no problems in reprocessing at the Nuclear Fuel Services Plant, West Valley, New York, based on a discussion with Mr. J. P. Duckworth. Experience at the WAK reprocessing demonstration plant at Karlsruhe, Germany also suggests that reprocessing of pool-stored fuel will not present problems, according to Dr. K. L. Huppert. No other substantive problems have been identified to suggest that reprocessing of spent fuel will be problematic after extended storage. In fact, decay of radioactivity during storage offers advantages in reprocessing.

A rumor regarding spent fuel storage problems in Spain also has been investigated. In response to an inquiry requesting information on spent fuel storage problems in Spain, Dr. A. Uriarte of Junta de Energia Nuclear in Madrid provided the following information:

- There are no problems with the wet storage of Zircaloy-clad spent fuel in Zorita (PWR) and Garona (BWR).
- There were problems with the first charge in Garona (BWR). These problems occurred during the irradiation, not in the storage.

- There were problems with the storage of Magnox-clad[a] spent fuel (Vandellós).

A French incident of unexplained radiation release from spent fuel assemblies was indicated at the OECD/NEA Seminar on the Storage of Spent Fuel Elements, Madrid, June 1978. A subsequent discussion with Mr. LeFort at Fountenay-aux-Roses provided the following explanation for the release:

- Four stainless-clad fuel assemblies were shipped to LaHague from the Chooz reactor (PWR) in Belgium. The fuel was shipped dry.
- The fuel was from a lot that had experienced fuel failures in the reactor. However, known "leakers" are canned before shipment; therefore, the fuel was considered intact when loaded into the cask.
- The fuel had cooled between 2 and 3 yr in the reactor pool.
- Mr. LeFort did not have details regarding the in-reactor failure mechanism, although the failure may have involved some mechanical factors.
- Upon arrival at LaHague, cooling water was pumped into the cask. Upon sampling, the cooling water was found to contain radiation levels much higher than usual. However, the radiation was contained within the cask during shipment.
- The high activity created a nonroutine situation that was dealt with to avoid radiation exposure to plant personnel. Transfer of the fuel to the storage pool resulted in contamination of the pool water to a level below 10^{-1} Ci/m^3.
- The pool purification system was able to reduce radiation levels in the pool water to values that soon permitted normal operations at the pool.
- Mr. LeFort did not believe that the cause of the radiation release had been identified, but he considered that the release probably was the result of damage to the assemblies during shipping. This may have involved fuel with near-failures from the reactor exposure.

(a) Magnesium-base alloy (e.g., Mg-0.8% Al-0.0025% Be.)

Fuel with known defects is stored in French pools in cans with a sintered stainless filter.

Beyond this incident, handling, shipping and storage of spent fuel appear to proceed without substantial problems, based on the above publications, discussions at the BEFAST meetings and at the OECD/NEA seminar in Madrid. However, there is a strong international concensus that spent fuel surveillance activities should continue at a reasonable level.

Of possible interest to some pool operators is a case in Sweden involving chemical decontamination of a PWR spent fuel pool (Appendix B).

SPENT FUEL SURVEILLANCE ACTIVITIES - OTHER COUNTRIES

The United States' program to characterize spent nuclear fuel is being scoped to complement similar activities in other countries. Other national programs that involve inspection and/or examination of spent fuel after substantial storage are summarized below.

Canada

The Canadian program to characterize spent nuclear fuel is outlined in a recent publication (Hunt, Wood and Bain 1979). The program is scoped to examine 140 fuel elements (rods) of Zircaloy-clad oxide fuel from several Canadian reactors. The nondestructive and destructive examinations are planned on 5-yr intervals through 1995. The first fuel was discharged in 1962 with very low burnup. The second discharge was in 1963, with burnups of $\sim$5000 MWd/MTU. The first fuel examinations were conducted soon after the second discharge to define effects of the reactor exposure. The second examination on the same fuel occurred in 1978, after $\sim$16 yr in-pool (Figure 1). The Canadian intention is to re-examine the same fuel over a 30-yr period. Conclusions from the recent fuel examinations are (Hunt, Wood and Bain 1979):

- No deterioration, either by corrosion or mechanical damage, has occurred during 16 yr of storage in water.
- There has been no additional release of fission products from the UO_2 matrix during 11 yr of storage in water.

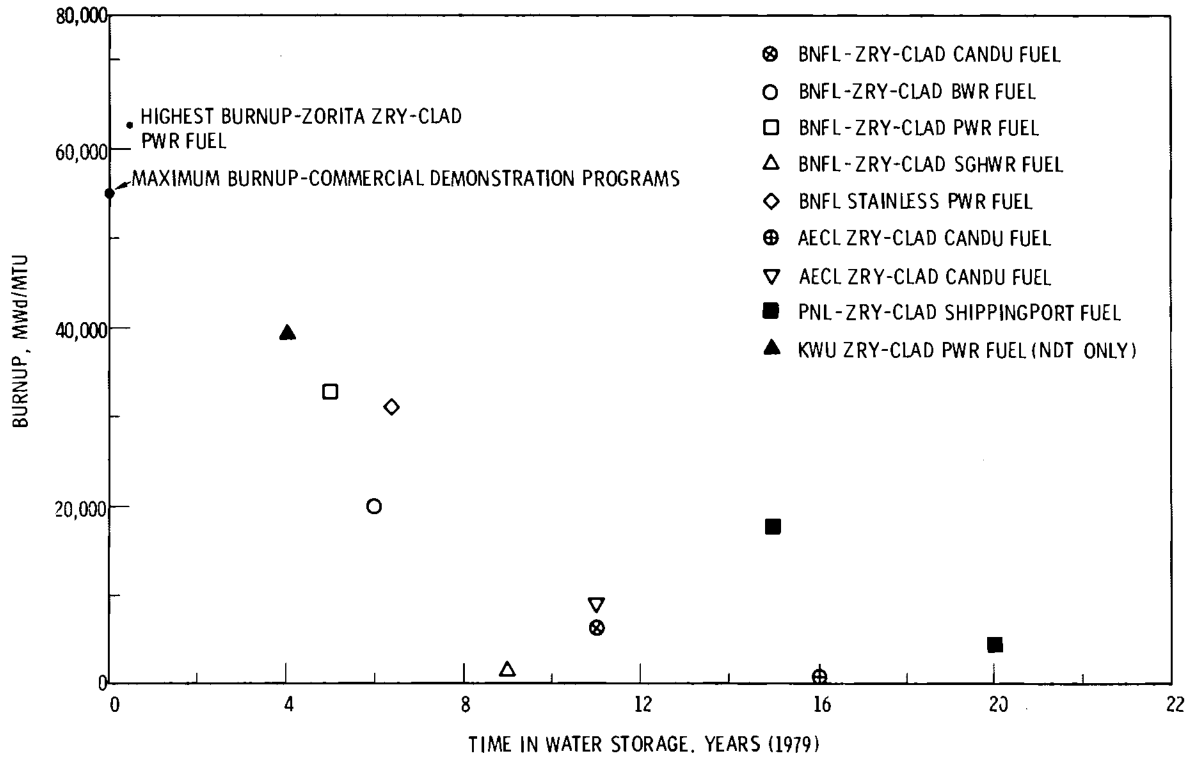

FIGURE 1. Summary of Spent Fuel Examinations to Define Effects of Water Storage (Open Points--Completed; Closed Points--Planned)

- No fission product-induced stress-corrosion cracking is anticipated during storage at temperatures below 373 K.

These observations lead to the general conclusion that all evidence to date indicates that fuel can be stored safely in water for at least 50 yr.

Kraftwerk Union - Federal Republic of Germany

Kraftwerk Union (KWU) is conducting periodic examinations of spent fuel from the Obrigheim reactor (PWR). The surveillance program includes:

- nondestructive examinations on 18 intact fuel rods (Figure 1) (destructive examinations will be instituted if unusual behavior is detected by NDT.)
- periodic visual and photographic examinations of 10 fuel rods with visible defects

- capsule tests to investigate radioactivity releases from defective and nondefective fuel under pool storage conditions
- investigation of radiolysis effects in closed capsules.

The PWR fuel undergoing periodic nondestructive examinations has the following characteristics:

- burnup range - 15,400 to 38,900 MWd/MTU
- time in pool (as of November 1979): 597 to 1169 days.

WAK, Karlsruhe, Federal Republic of Germany

The WAK[(a)] spent fuel pool is associated with a fuel reprocessing demonstration plant at Karlsruhe. A surveillance assembly (PWR) is removed to the plant hot cell annually, photographed on four sides in air and returned to the pool. The same procedure was used on a BWR assembly having a defect that developed in-reactor. No detectable variation in the defect on the exposed UO_2 was detected. The assembly had been stored in water $\sim$5 yr.

British Nuclear Fuels, Ltd. (BNFL) - United Kingdom

In connection with the Windscale hearings, BNFL examined destructively the following fuel from water-cooled reactors (Flowers 1977; Warner 1977) (see Figure 1):

1. A Canadian Zircaloy-clad fuel bundle (6500 MWd/MTU) was examined metallurgically after 11 yr in pool storage. There was no evidence of pool-induced corrosion or other degradation.
2. Three BWR Zircaloy-clad rods (20,000 MWd/MTU, 6 yr in-pool) and three PWR rods (33,000 MWd/MTU, 5 yr in-pool) were examined nondestructively and metallurgically. Again, there was no evidence of pool-induced degradation.
3. A Zircaloy-clad Steam Generating Heavy Water Reactor fuel assembly containing two failed rods was placed in a closed can after a burnup of 1900 MWd/MTU. After 9 yr, the radioactive species contained in the water

(a) Wiederaufarbeitungsanlage

inside the can had risen to 1 mCi (∿5 ppm of ^{137}Cs). A detailed hot-cell examination of the fuel assembly indicated only small increases in fuel rod diameter at the defects, with no evidence that a $UO_2 \rightarrow U_3O_8$ conversion was occurring. There was no evidence of pool-induced degradation on the Zircaloy cladding or on the stainless-steel spacer. Some mild corrosion of ferritic steel mandrels had occurred.

4. A PWR fuel rod with stainless-steel cladding was exposed 3 yr in-reactor, 4 yr in pool.

No evidence of pool-induced fuel cladding degradation was observed in the examinations, including the fuel with reactor-induced defects.

Fuel surveillance activities have been conducted in other countries, e.g., underwater reconstitution of irradiated fuel in a Swedish BWR pool. However, the pool residence times have been short.

UNITED STATES FUEL EXAMINATION PROGRAM

The United States spent fuel surveillance program is directed toward the following:

- Analyzing Zircaloy-clad oxide fuel from the Shippingport reactor (PWR) stored since 1959. Rods from an assembly discharged in 1963 also will be examined. The Shippingport fuel assemblies are 10-1/4-in. (26-cm) long. The burnup on the assembly discharged in 1959 is estimated to be 4000 MWd/MTU. The fuel is well characterized and has had prior examinations. The 1959 assembly is the world's oldest Zircaloy-clad oxide fuel still in water storage.

- Investigating high-burnup Zircaloy-clad oxide fuel. Fuel management strategies include investigation of incentives for extended burnup. Several high-burnup demonstration programs are underway. (See Appendix A, Table A8) (Our assessment of spent fuel burnup history and characteristics appears later in this report.) The highest burnup of commercial Zircaloy-clad fuel is 62,000 MWd/MTU, rod average, which resided in the

Zorita reactor (PWR, Spain). The remaining high-burnup rods are in dry storage in Spain. Zircaloy-clad rods with 39,000 MWd/MTU burnup are examined periodically by KWU. Some commercial fuel rods in United States reactors have burnups exceeding 40,000 MWd/MTU.[a] Some fuel is projected to reach burnups up to 55,000 MWd/MTU by about 1981 (see Appendix A, Table A8), but final decisions are pending. We have explored prospects to obtain selected high-burnup demonstration fuel for long-term surveillance by discussions with utilities and nuclear fuel vendors. Observations on the demonstration fuel would anticipate by several years unusual storage characteristics that might develop on commercial high burnup fuel inventories. To date, examination of Zircaloy-clad spent fuel with burnups to 39,000 MWd/MTU has not shown evidence that the fuel cladding is degrading during water storage.

- Assessing the stainless-clad fuel inventory in pool storage (a program activity during FY 1979). In United States pools ∿1400 stainless-clad fuel assemblies are stored. Although this number represents less than 10% of the total fuel inventory, the number is sufficient to justify surveillance of stainless-clad fuel. During FY 1979 we have identified several potential surveillance candidates. Negotiations are underway for program access to a stainless-clad PWR assembly.

- Analyzing defective fuel. Spent fuel with cladding defects that developed in-reactor are stored in several fuel pools. Current experience suggests that the defective fuel is being stored without substantial problems (Johnson 1978). We have already pointed out surveillance of defective fuel by BNFL and KWU. In addition, the United States program will include surveillance of defective fuel.

  There is a substantial advocacy for canning fuel with defective rods although the evidence suggests that there is minimal in-pool radiation release from defects that develop in-reactor (Johnson 1978). It is

(a) Recently, Shippingport reactor fuel with a burnup of ∿41,000 MWd/MTU was examined (see WAPD-TM-1412).

important to recognize that the additional cost of the canisters is only one consideration in weighing the question of canning defective bundles. The canister surfaces will absorb some radioactivity from the pool water; therefore, the canisters must eventually be disposed of as radioactive material. Radioactive species contained by the canisters will partially decay, but the residual radioactive water inside the cans must also eventually be treated. These factors must be included in assessing the option of canning fuel assemblies that developed cladding defects in-reactor.

Comment on Westinghouse HEDL Program

Results from Hanford Engineering Development Laboratory (HEDL) investigations[a] relating to spent fuel behavior include examinations of PWR spent fuel, which is being utilized in geologic storage demonstration programs. PWR fuel from a Turkey Point No. 1 reactor is being examined in Battelle Columbus Laboratory (BCL) hot cells.[b] The fuel has a burnup of ∿28,000 MWd/MTU and was in reactor pool storage for ∿2-1/2 yr. Current plans exist to obtain and examine fuel from a boiling water reactor.

Turkey Point fuel assemblies have been placed in dry storage at the EMAD Nevada testing site, under the Dry Surface Storage Test (DSST) program. Capsule tests at elevated temperatures to evaluate possible failure mechanisms are underway at BCL, using irradiated Turkey Point rods. Additional Turkey Point assemblies will be examined and placed in dry storage tests under the Climax Program.

(a) Under the National Waste Terminal Storage Program
(b) Results of the first phase of the Turkey Point fuel examinations are contained in two reports:

- Davis, R. B. 1979. Data Report for the Non-Destructive Examination of Turkey Point Spent Fuel Assemblies B02, B03, B17, B41, and B43. TC-1284.

- Davis, R. B. and V. Pasupathi. 1979. Data Summary Report for the Destructive Examination of Rods G7, G9, J8, I9 and H6 from Turkey Point Fuel Assembly B17. TC-1540.

SPENT FUEL EXAMINATIONS

INTRODUCTION

A major purpose of the Spent Fuel and Fuel Pool Component Integrity Program is to investigate the corrosion and metallurgical conditions of pool-stored irradiated fuel after extended water storage. Ultimately, we plan to determine the durability of high burnup spent fuel (sound and defective) during such storage.

Olander (1976, pp. 114-115) states that burnup (also often called "exposure") can "...be expressed as the number of megawatt days of thermal energy released by fuel containing one metric ton (10^6g) of heavy-metal atoms (MWd/MTU)." There are also other measures of the integrated irradiation[a] to which the fuel material has been subjected; however, in this report we will use MWd/MTU (or MWd/MTHM when the heavy metal (M) in the fresh fuel contains uranium plus plutonium or thorium).

There has been a continuous increase in the average discharge fuel burnup. The average was about 8000 MWd/MTU in 1962 (Turner, Elgin and Hancock 1979). In January 1973, the average burnup of all United States discharged fuel was 11,200 MWd/MTU,[b] and the average burnup of all world-wide discharged Zircaloy-clad fuel was 8,200 MWd/MTU[b] (Nuclear Assurance Corporation 1973a, p. 29). For Zircaloy-clad fuel and on a worldwide basis as of January 1973, the highest discharge burnup for quantities of fuel of $\geq$5 MTU was $\sim$16,700 MWd/MTU[b] for BWRs and $\sim$25,200 MWd/MTU[b] for PWRs (Nuclear Assurance Corporation 1973b, pp. 24-25). The average discharge fuel burnup was 24,000 MWd/MTU in 1978 (Turner, Elgin and Hancock 1979).

(a) Traditionally, burnup units of MWd/T, fissions/cm^3, and percent burnup have been used. Hannum (1967) has discussed these three plus other burnup units and has pointed out some of the disadvantages of the three. For example, the unit MWd/T, is ambiguous: it is not clear whether the denominator is long, short, or metric ton, and it is also not clear whether the mass quoted is a ton of fuel material or heavy atoms.

(b) Burnup values have been revised from values in the original publication based on recent data from Fuel-Trac®, Nuclear Assurance Corporation.

As indicated by Roberts, Davis and Nash (1978), the current nominal design burnups for PWR fuel and BWR fuel are 33,000 MWd/MTU and 27,300 MWd/MTU, respectively (Turner, Elgin and Hancock 1979). The Electric Power Research Institute (EPRI) extended-burnup program that is presently underway in domestic power reactors involves the irradiation of fuel assemblies up to expected assembly average burnups as high as 55,000 MWd/MTU (Roberts et al. 1979). EPRI plans call for the discharge of Zircaloy-clad PWR fuel with a burnup of 55,000 MWd/MTU in 1981. The EPRI report (Roberts, et al. 1979) indicates that with such "...extended burnups, peak fuel pellet burnups will now exceed 60,000 MWd/t (up from 40,000 MWd/t)."

PROGRAM PHILOSOPHY

The task of establishing corrosion and metallurgical conditions of water-stored spent fuel has short- and long-term aspects. Short-term aspects include identification and characterization of optimum surveillance fuel candidates, using nondestructive and destructive techniques. Principal considerations are summarized under "Assessment of Surveillance Fuel Assembly Candidates." Long-term aspects comprise periodic surveillance, including frequent visual inspections, and nondestructive and destructive examinations at 5-yr intervals for as long as the need exists to characterize the spent fuel storage behavior.

Where possible, the surveillance fuel selection process involves fuel assembilies that have had prior examination to establish the condition upon reactor discharge.

Once an Away-from-Reactor (AFR) storage facility is in place, the surveillance fuel assemblies will be stored there. In addition to surveillance on the designated fuel assemblies, other assemblies in the AFR inventory will be inspected after random selection.

The program also includes monitoring spent fuel pool components such as storage rocks, pool liners, and piping to determine the corrosion behavior in the pool environments. Investigations of fuel cladding and component behavior under nonstandard conditions also are planned.

DISCUSSIONS WITH DOMESTIC FIRMS

Preliminary Discussions and Arrangements for Participation in the Program

During FY 1979 discussions were held with fuel vendors, utilities, and fuel storage facilities regarding access to nuclear fuel with desirable characteristics (including high burnup, long pool residence, and defective fuel) for long-term surveillance to define integrity of the fuel. We also discussed with fuel vendors their possible participation in the program.

Following initial contacts by telephone and letter, meetings were held in early July 1979 with Westinghouse Electric Corporation and in late July with General Electric Company. At both meetings, possible fuel assembly candidates were discussed and the prospects for vendor fuel examination crews to participate in poolside inspections of selected fuel assemblies were explored. To accommodate vendor off-peak fuel examination schedules, it appears that the best time for fuel examinations under this program would be in the summer.

Telephone and letter contacts were also made with Combustion Engineering, Inc. and Babcock and Wilcox Company in August and with Exxon Nuclear Company in May and September 1979.

A meeting was held with EPRI in July to discuss the possibility of including in this program selected fuel assemblies from the EPRI extended-burnup program that is currently under way in domestic power reactors. Such an arrangement is possible; however, arrangements will need to be formalized with participating utilities, fuel vendors and with EPRI.

In addition, we contacted by telephone, letter, and/or through discussions at technical meetings Consumers Power Company (Big Rock Point), Commonwealth Edison Company, Northeast Utilities (Connecticut Yankee), and Yankee Atomic Electric Company (Yankee Rowe). The purpose of the contacts was to locate information about specific fuel and/or to discuss the possibility of poolside examination of selected fuel assemblies at a reactor spent fuel pool.

Nuclear Fuels Services (NFS) was contacted by telephone about their fuel inventory (they have fuel from five reactors), the status of a specific fuel assembly (A8) from Yankee Rowe, and to explore the possibility of poolside examinations of fuel at the NFS facility and of shipping fuel assemblies from the facility. Specific information regarding fuel stored at the Savannah River Plant RBOF pool was requested during a July 1979 visit. The RBOF staff agreed to supply information regarding the following types of fuel: sound and defective Zircaloy-clad Saxton fuel, HWCTR fuel, and VBWR fuel.

Negotiations for Fuel

Negotiations have been finalized for the Department of Energy to obtain custody of Zircaloy-clad fuel from the Shippingport reactor. The fuel has been stored in water up to 20 yr. The fuel is not prototypic of commercial reactor fuel. However, it represents the world's oldest Zircaloy-clad oxide fuel stored continuously in water. Characteristics of the fuel appear below and in Table 1. The blanket fuel assemblies consist of fuel rods, each 10-1/4-in.-long by 0.411-in.-dia, with natural uranium dioxide pellets contained in Zircaloy-2 tubes. There are 120 fuel rods in each bundle.

SELECTION OF CANDIDATE FUELS FOR NEAR-TERM STUDIES

This task has centered on the identification of 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.

A literature survey on sound and defective fuel assemblies was initiated and is continuing. The results of the survey to date are shown in the tables and figures in the appendix. In general, we have included in tables and figures only those discharged fuel assemblies with assembly average burnups higher than approximately 20,000 MWd/MTU.

Of particular interest in the continuing defective fuel literature search were fuel assemblies with verified defects (e.g., by visual examination). The results of the survey to date are shown in Table A9 in Appendix A.

Assessment of Surveillance Fuel Assembly Candidates

The guidelines for fuel assembly selection are:

TABLE 1. Characteristics of Shippingport Fuel Available for Spent Fuel and Fuel Pool Component Integrity Program

Item[a]	Operation[b]	Depletion (MW Days per metric ton)	Years of Pool Storage at Shippingport and ECF[c]
15 Fuel Rods from Blanket Bundle 0551	5806 EFPH in PWR Core 1 Seed 1	about 4000 average	19 (1959 discharge)
Blanket Bundle 074 (previously designated backup for MELBA)	27780 EFPH in PWR core 1 Seeds 1, 2, 3, 4	18000 average	15

(a) The blanket bundles consist of 120 fuel rods on an 11 x 11 square array (bundle width is 5.195 in.). The rods are 10-1/4 in. long and have sealed end caps at both ends that are welded together to form tube sheets at the ends of the bundles. Seven bundles are stacked axially to form a fuel assembly in the reactor. The rods have Zircaloy-2 tubing with an outside dia of 0.411 in. nominal and contain annealed about 25 fuel pellets. The fuel pellets are right circular cylinders of sintered UO_2.
(b) Average rod surface temperature during PWR core 1 operation was about 400°F to 545°F.
(c) Water pit temperature at Shippingport and ECF prior to 1973 was 60 to 80°F. After 1973 the water pit temperature at ECF has been 45 to 60°F.

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

Not all of the factors will be optimized in a single fuel assembly; therefore, trade-offs will be necessary in selecting the best available fuel for examination and long-term surveillance.

There is a continuing effort to select the fuel assemblies for examination and long-term surveillance with particular emphasis on the following:

- 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.

Results of the continuing search for fuel assembly candidates are presented in Tables A1 to A9 and Figures A1 to A8 in Appendix A. The search for candidates led to the Zircaloy-clad fuel rods exposed to very high burnup in the Zorita reactor (Spain). However, the remaining fuel rods (peak rod burnup is 62,000 MWd/MTU) have been stored dry for all but ∿6 mo since discharge from the reactor. To date the highest burnup we have identified on United States commercial fuel assemblies with commercial Zircaloy-clad fuel rods in storage is 34,760 MWd/MTU (region average in Point Beach-2 in Table A3).[a] As shown in Table A6, the Nuclear Assurance Corporation (NAC) report (1979) indicated that the highest burnup for fuel assemblies with stainless-steel-clad fuel rods discharged from United States and foreign reactors was 37,500 MWd/MTU (batch average) and 32,000 MWd/MTU, respectively.

Figures A5 and A6 indicate that PWR fuel assemblies with stainless steel-clad fuel rods scheduled for discharge during the 1979-1982 time frame are projected to have average burnups in the 20,000-33,000 MWd/MTU range. Thus, it appears that the average burnup of stainless-clad fuel assemblies discharged over the next several years will be about the same as those of most similar fuel assemblies discharged during the last 8 yr (however, some domestic fuel of this type discharged in the 1972-1976 period had average burnups to ∿37,500 MWd/MTU).

The effort to identify suitable defective fuel is continuing. The emphasis in the search is to locate fuel assemblies with visible fuel rod defects (e.g., small hole in cladding, crack in cladding, missing section of fuel rod) that have previously been verified. Of particular interest are fuel assemblies with defects associated with peripheral fuel rods (i.e., defects are visible without disassembly of the fuel). There also appears to be some motivation to evaluate interim water pool storage of damaged fuel of the type now in the Three Mile Island-2 reactor.

(a) Shippingport fuel has been examined after reaching burnups up to ∿41,000 MWd/MTU. The fuel resided in the reactor for 17 yr; 2131 days of that time were at reactor operating conditions (see WAPD-TM-1412).

SELECTION OF CANDIDATE FUELS FOR LONG-TERM STUDIES

Long-term studies of fuel assemblies will include those from the near-term studies and, if possible, high burnup fuel assemblies selected from the EPRI and/or DOE extended-burnup programs currently underway in domestic power reactors. The EPRI extended-burnup program (Table A8) involves Zircaloy-clad fuel that is to be irradiated to burnups as high as 55,000 MWd/MTU. Under the EPRI program, Zircaloy-clad fuel is being irradiated in two BWRs and 4 PWRs. The DOE high-burnup fuel studies include the Consumer Power Project, Commonwealth Edison Project, Duke Power/Arkansas Power and Light Projects, and TVA-GE Project (Lang 1978; Nuclear Fuel 1979).

TECHNICAL BASES AND CRITERIA FOR EVALUATING FUEL BEHAVIOR

Efforts are underway to formulate the technical bases and criteria that are needed to evaluate fuel behavior under the Spent Fuel and Fuel Pool Component Integrity Program. As mentioned by the fuel vendors during the meetings, it is important to specify exactly what is required from the fuel examinations (especially in case of defective fuel).

EXAMINATION AND MONITORING OF FUEL CONDITION

By the end of FY 1980, we plan to complete hot cell examinations on at least one fuel assembly with Zircaloy-clad fuel rods and one PWR fuel assembly with stainless-steel-clad fuel rods.

EXPERIMENTAL PROGRAM POSSIBILITIES

The fuel and fuel component examinations have first priority in the program. As these activities are brought to a satisfactory level of effort, other investigations will be initiated, including behavior of spent fuel and fuel pool components under non-standard conditions.

SPENT FUEL COMPONENT INTEGRITY

The boiling water reactor (BWR) and Away-from-Reactor (AFR) fuel pools have deionized water chemistries. The pressurized water reactor (PWR) spent fuel pools have a boric acid pool chemistry to be compatible with the primary system chemistry during refueling.

Typical spent fuel pool components and materials appear in Table 2. Some materials have been exposed in spent fuel pools over two decades in deionized water (since 1947 at the NRX[a] pools). Very few corrosion problems have developed on the fuel pool components in deionized water chemistry. Mild steel tends to rust, but both stainless steel and aluminum have generally performed satisfactorily. Some specific experience with aluminum components is cited in a later section.

The maximum United States experience with boric acid pool chemistry is $\sim$12 yr. Over that time, pool liners, racks and grapples have appeared to perform well. A corrosion problem which has developed in stainless-steel piping at some PWR pools is discussed later.

Dr. Martin Peehs of Kraftwerk Union in Germany recently indicated that a fuel-handling machine that had performed for 12 yr in a boric acid pool at the Obrigheim reactor (PWR) in Germany (GFR)[a] was disassembled and inspected without finding evidence of degradation.

Our FY-1979 program included some exploratory investigation of the integrity of stainless-steel rack materials exposed to boric acid chemistries. We obtained a section of a fuel rack support from the Yankee Rowe (PWR) reactor and a section of spent fuel rack from the Point Beach reactor. Both materials were 304 stainless steel. More recently we obtained a section of 304 stainless steel 8-in.-dia, schedule 40 pipe from the Three Mile Island spent fuel pool redundant cooling system. The pipe had developed a through-wall crack.

(a) Experimental nuclear reactor at Chalk River Nuclear Laboratories, Ontario, Canada.
(b) German Federal Republic

TABLE 2. Summary of Materials in Fuel Pools (Johnson 1977)

Component	Sub-Components	Material[a]	Alloy
Wall	--	Reinforced Concrete	--
Pool Liner:	--	Stainless Steel Epoxy, Fiberglass	304
Heat Exchanger[b]	--	Stainless Steel	304/316
Filter:	Vessel Filter Elements:	Stainless Steel Stainless Steel Diatom. Earth, Fiber	304L 304L
Recirculating Pumps:	Casing, Shaft Impeller	Stainless Steel Bronze	316
Demin, Water Return Pump	Casing, Impeller, Shaft	Stainless Steel	316
Deionization Unit	Tank	Stainless Steel	304L
Cask Head Support Racks	--	Stainless Steel	304L
Gates and Guides	--	Stainless Steel	304L
Canister Storage Racks:	Racks:	Stainless Steel Aluminum	304L 6061-T6
	Embedded Supports	Stainless Steel	304L
Fuel Storage Canisters:[c]	--	Stainless Steel Aluminum	304L 5083/5086/6063
Leaker Can Support Racks	--	Stainless Steel Carbon Steel, Epoxy-Coated	-- --
Control Rod Cluster Storage Racks	--	Stainless Steel	--
Portable Offgas Hoods:	--	Stainless Steel Aluminum	--
Cask Handling Crane	Cable & Grapple	Stainless Steel	--
Canister Crane	Cable & Grapple	Stainless Steel	--
Fuel Transfer Conveyer	--	Stainless Steel	304L
Insulators	--	Tygon, Neoprene	
Cask Impact Pad:	Cladding Pad (not exposed to H_2O)	Stainless Steel Carbon Steel (3-1/2-in. thick)	
	Honeycomb (not exposed to H_2O)	Aluminum	

(a) Types identified in survey; other types may be used in some pools.
(b) Carbon steel tubes were originally installed in heat exchangers at one pool; severe rusting caused a visibility problem in the pool water, resulting in retubing with stainless steel. Copper alloy tubes are used at one R&D facility pool.
(c) Some canister walls contain boron-impregnated aluminum for reactivity control, clad with stainless steel or aluminum.

This report provides a brief summary of the results of the investigations. The results will be reported in more detail in a topical report when the TMI investigation is completed.

INVESTIGATION OF YANKEE ROWE RACK SUPPORT

Macroscopic Observations and History

A section of a stainless steel fuel rack support from the Yankee Rowe spent fuel pool was received at Pacific Northwest Laboratory for metallurgical examination on February 22, 1979. The section had a total surface activity of 3.74 μCi Co-60 (January 4, 1979) with smearable contamination of 500 dpm/cm^2. The nominal conditions for the spent fuel pool water are given in Table 3. The section of rack was exposed to the water conditions in Table 3 for $\sim$17 mo, from December 1975 to May 1977.

Table 3 shows an elemental analysis of samples taken from the two plates (see Figure 2 for sample location). It is evident that materials used had the same general composition with only a slight variation in nickel content.

Figure 2 shows the overall size and geometry of the rack section that consisted of a piece of 1/4-in. 304 S/S angle with 3-in. flanges attached by threaded rods and weldments to a solid 3-3/4-in. square, 28-in.-long billet. The billet had a surface characterized by a brown, crusty scale suggesting it was a low-grade, low-Cr ferritic steel (e.g., Alloy 501). The stainless-steel angle, slightly discolored in spots but generally exhibiting a bright finish, was characterized by a flame-cut edge from removal operations and an abbreviated double T welded joint (approximating 1/2 of a box end joint).

From Figure 2 it appears the welding process used was a manual shielded metal arc technique with a coated electrode (308 S/S commonly used) for filler material. The middle macrograph in Figure 2 shows slag entrapment at the weld base metal fusion line and local grinding marks, while weld spatter is observed in the right macrograph. The latter effect is caused by an excessive welding current or variations in metal transfer from nonhorizontal welding positions. A heat tinting effect is also seen in the photographs, indicating that areas of the plates were exposed to fairly high temperatures during the joining process had not been ground after welding. The weld beads and fusion line areas appear structurally very sound with no visual pits or surface cracks.

TABLE 3. 304 Stainless-Steel Compositions (Two Welded Pieces)

A. Rack Composition

		Chemical Analysis													
Heat	Material	C	Mn	P	S	Si	Ni	Cr	Mo	V	Cu	Sn	Ti	Al	Co
YR3	304SS	0.07	1.63	0.028	0.016	0.48	9.69	18.18	0.41	0.033	0.28	0.014	0.008	0.007	0.130
YR4	304SS	0.07	1.63	0.027	0.016	0.47	9.26	18.17	0.41	0.032	0.27	0.015	0.008	0.008	0.118

B. Rack Exposure Conditions

• pH	6.8
• boron	800 ppm (max.)
• chlorides	<0.15 ppm
• conductivity	50 $ohms^{-1}$
• activity	3.5×10^{-3} μCi/ml
• temperature (dependent upon	75 - 95°F (24 - 35°C)
season)	75 - 95°F (24 - 35°C)

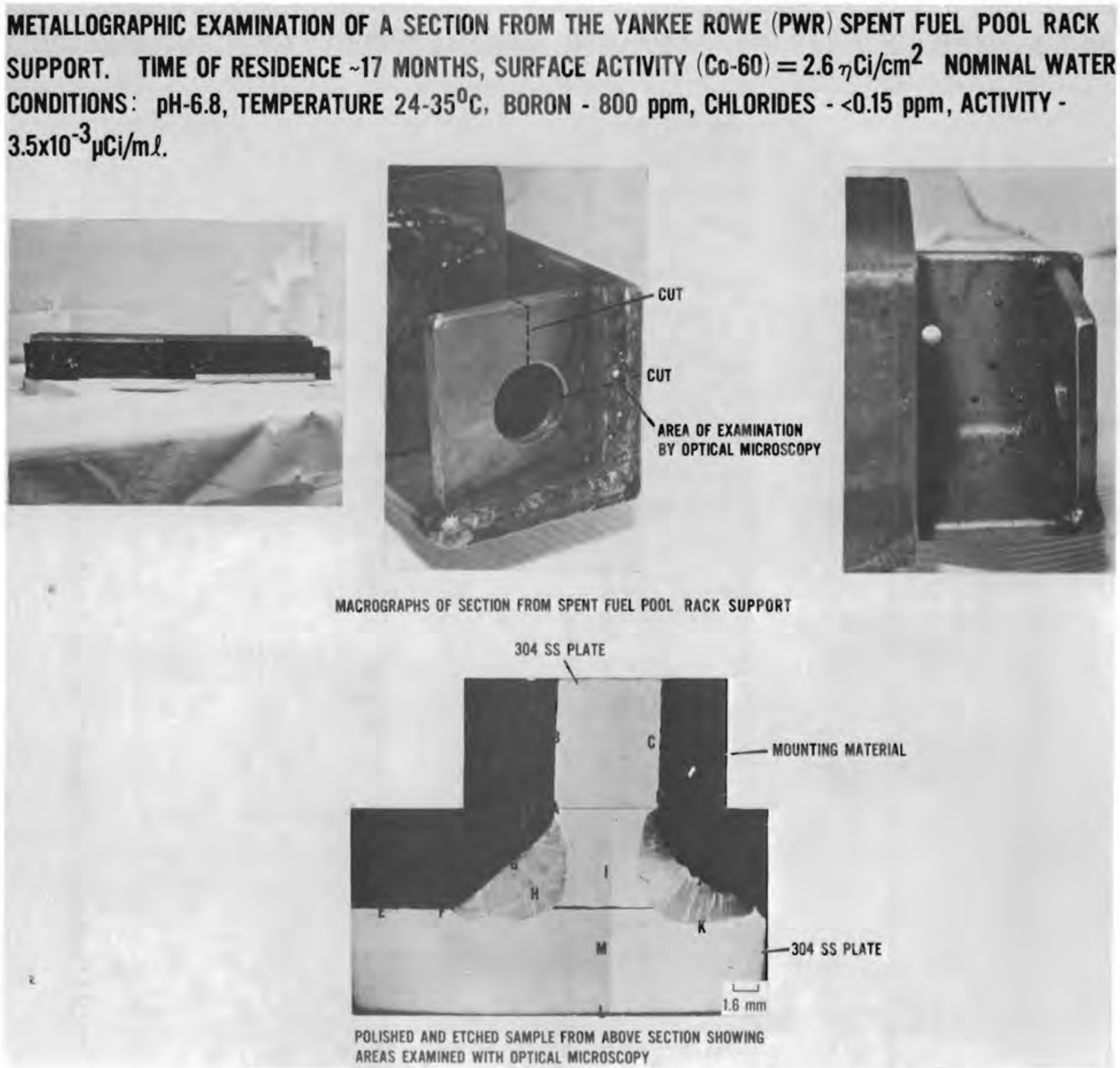

FIGURE 2. Metallographic Examination of a Section from the Yankee Rowe (PWR) Spent Fuel Pool Rack Support (Neg. No. 7906172-2)

Microstructural Observations

A sample of the 304 S/S angle, including two weldments, was cut from the section and prepared for optical microscopy. Microstructural examination was performed at points shown on the lower micrograph in Figure 2 and displayed in Figure 3. The major points of interest stemming from the examination were:

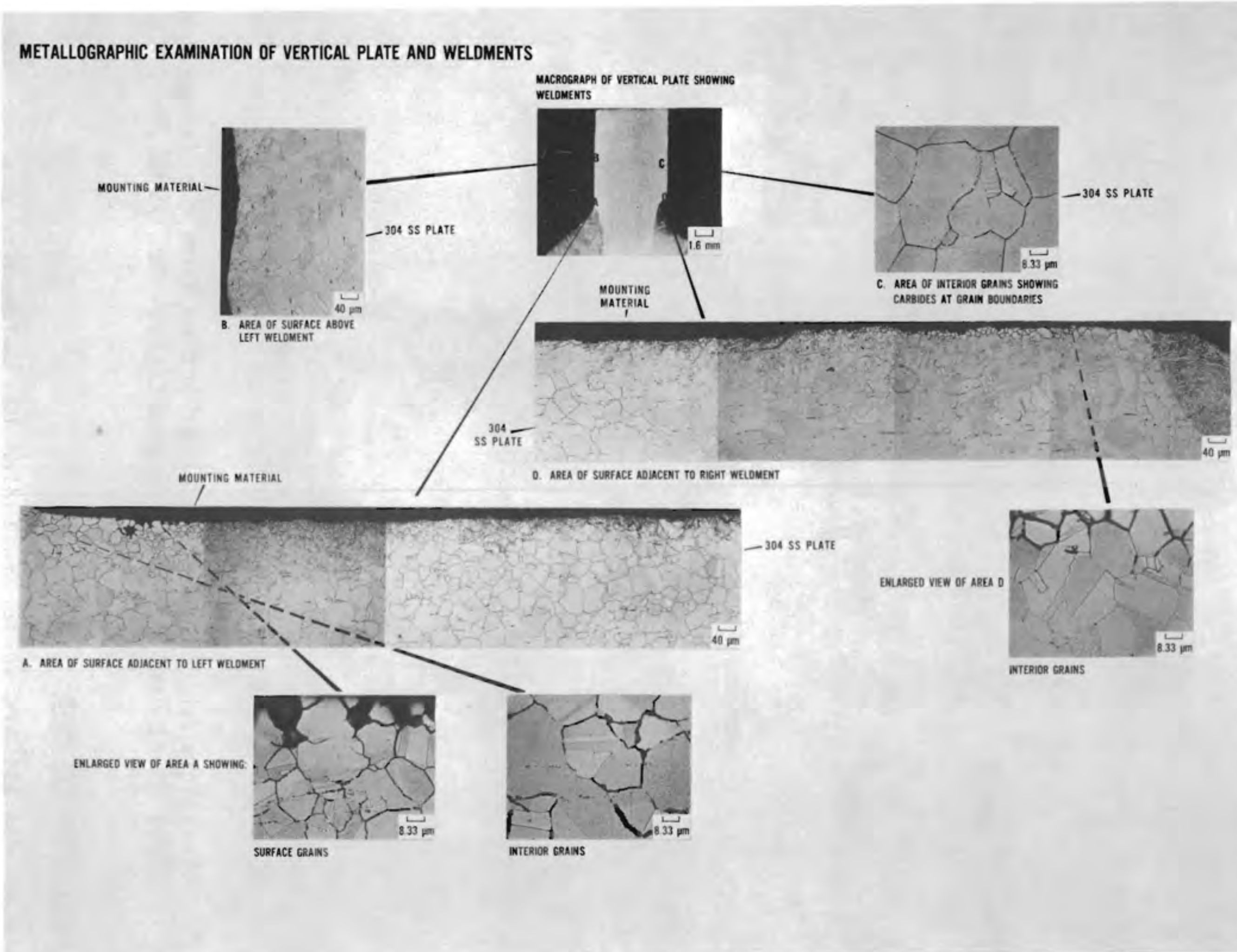

FIGURE 3. Metallographic Examination of Vertical Plate and Weldments

- a mild (25- to 80-μm deep) intergranular corrosion zone at points A and D on the vertical plate, with other surface locations (points B, E, F, L) lacking this effect. There was no evidence of intergranular stress corrosion cracking.
- plate microstructure characterized by equiaxed grains (∿0.033 mm diameter, ASTM #7) with elongated inclusion stringers depicting rolling direction and transgranular twins.
- nonuniform distribution of grain boundary precipitates.
- dendritically solidified weld metal (austenite with δ ferrite constituents), free of visual defects, with a small precipitate-free zone extending inward from the fusion line.
- slag entrapment within the joint space separating the vertical and horizontal plates.
- evidence of local deformation of surface grains in the form of slip bands running transgranularly through the matrix.

Detailed examination of the localized grain boundary penetrated zones at points A and D in Figure 3 revealed the following:

- a corrosion zone ∿1-mm wide from the fusion line extending from ∿25 to 80 μm in depth into the base metal.
- a decreasing gradient in both depth of grain boundary penetration and surface grain size with increasing distance away from the fusion line.
- an increasing gradient of grain boundary precipitates in surface regions with increasing distance from the fusion line.
- actual removal of surface grains in areas adjacent to the fusion line.
- attacked grain boundary widening to a great extent with grain boundary precipitates residing along the affected boundaries.

In summary, the most significant feature in the examination was the mild intergranular corrosion in the weld heat-affected zone. Table 2 suggests that the corrosion is likely related to weld procedure because areas with and without intergranular corrosion had 0.07 wt% carbon. However, the relatively high carbon, coupled with poor weld procedures, augments sensitization. There was no evidence of stress corrosion cracking.

INVESTIGATION OF POINT BEACH SPENT FUEL RACK SECTION

On May 30, 1979 we received a section of 304 stainless steel spent fuel rack from the Point Beach nuclear plant. Figure 4A shows a view of the rack removed from the spent fuel pool in a re-racking operation, including the area where the section was removed. Figure 4B shows the section of rack we received. Table 4 summarizes aspects of the rack composition and exposure. We obtained the analysis of two pieces of metal joined at the weld (see Table 4).

Metallographic examination of the weld areas (Figure 5A,B) indicated no evidence of stress corrosion cracking, intergranular attack or other degradation after ∿6-2/3 yr in boric acid chemistry. The relatively low carbon content may have been a factor in the absence of intergranular attack.

EXAMINATION OF CRACKED SECTION OF THREE MILE ISLAND NO. 1 SPENT FUEL POOL PIPE

A section of 304 stainless steel pipe from the Three Mile Island (TMI) No. 1 spent fuel pool heat exchanger received at PNL on July 9, 1979. The pipe had a through-wall crack. We have performed or arranged for the following analyses:

- visual inspection
- nondestructive testing
- metallography
- auger analysis
- scanning electron microscopy (SEM)
- X-ray diffraction of oxide
- elemental composition (performed by Jorgensen Steel, Seattle, WA)
- electro-potentiometric R (EPR) (performed by General Electric Co.)
- constant extention rate testing (CERT).

FIGURE 4A. 304 Stainless Steel Spent Fuel Pool Rack After 6-2/3 yr in Point Beach Reactor (PWR) Pool

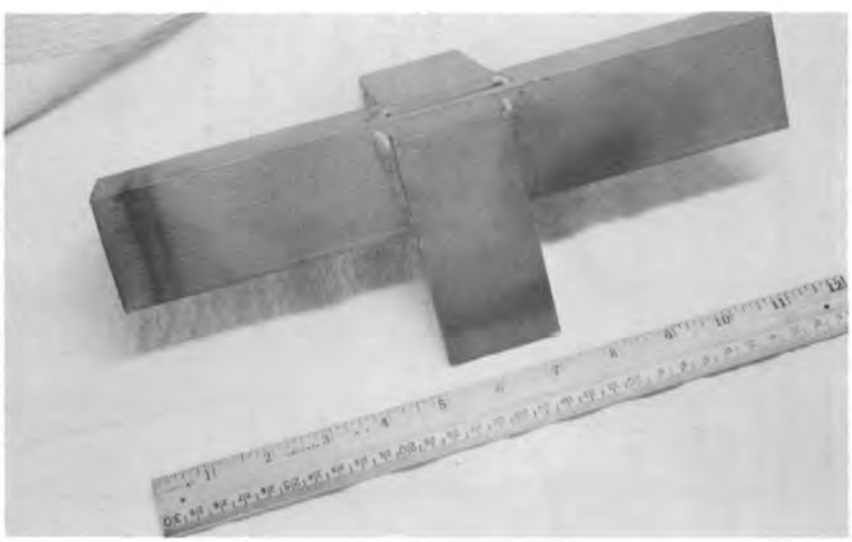

FIGURE 4B. View of Section Cut from Point Beach Rack

TABLE 4. Stainless-Steel Rack Composition and Exposure History

A. Rack Composition(a)

Heat	Material	Chemical Analysis C*	Mn	P	S**	Si	Ni	Cr	Mo	V	Cu	Sn	Ti	Al	Co
PB1	304SS	0.04	1.71	0.015	0.025	0.69	8.45	18.04	0.16	0.026	0.08	0.008	0.008	0.012	0.121
PB5	304SS	0.05	0.48	0.040	0.021	0.39	8.17	18.74	0.36	0.050	0.22	0.011	0.008	0.009	0.165

*0.045 and 0.065, respectively, by LECO method
**0.029 and 0.024, respectively, by LECO method

Racks supplied by Bechtel to Specifications ASTM-A240 and A276.

B. Rack Exposure History

First Rack Exposure to Borated Water - Sept. 7, 1972

Rack Removed from Spent Fuel Pool May 8, 1979.

Pool Chemistry:

- pH Range — 4.7 - 4.8
- Boron Range — 1900-2100 ppm (wt)
- Cl^- — ∿0.05 ppm
- Temperature Range — 68 to 113°F; 20 to 45°C

Radiation Readings on rack: 20 mR/hr at one ft after rinsing

Fuel Assemblies stored at rack position:(c) A13 for 653 days

C16 for 222 days

875 days (total)

(a) Analysis performed by Jorgensen Steel Co., Seattle, WA.
(b) Normal range: 70 to 80°F (21 to 27°C)
(c) The indicated fuel assemblies were stored adjacent to the rack section removed for analysis.

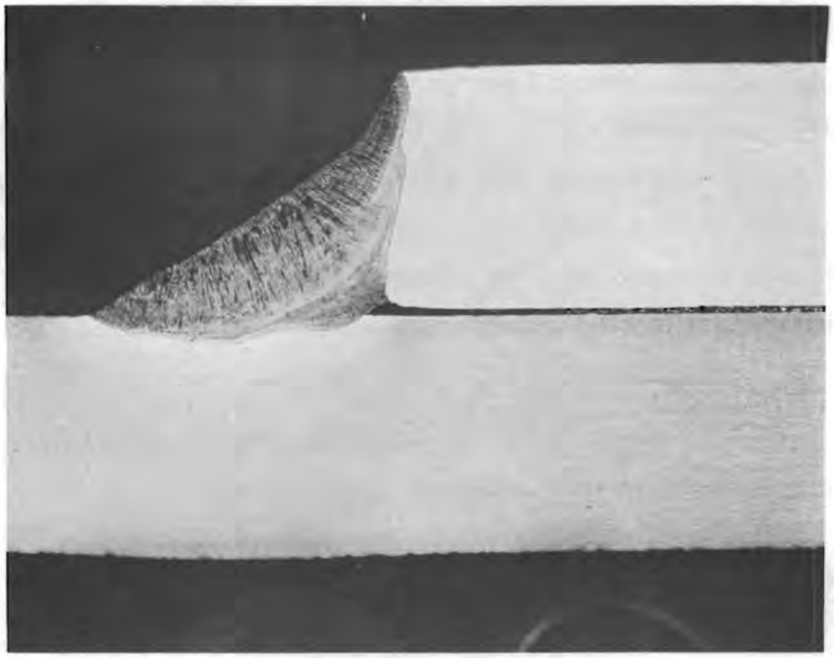

FIGURE 5A. View of Weld Area (6.3X) - 304 Stainless-Steel Rack Weld (Neg. No. 4P1111B)

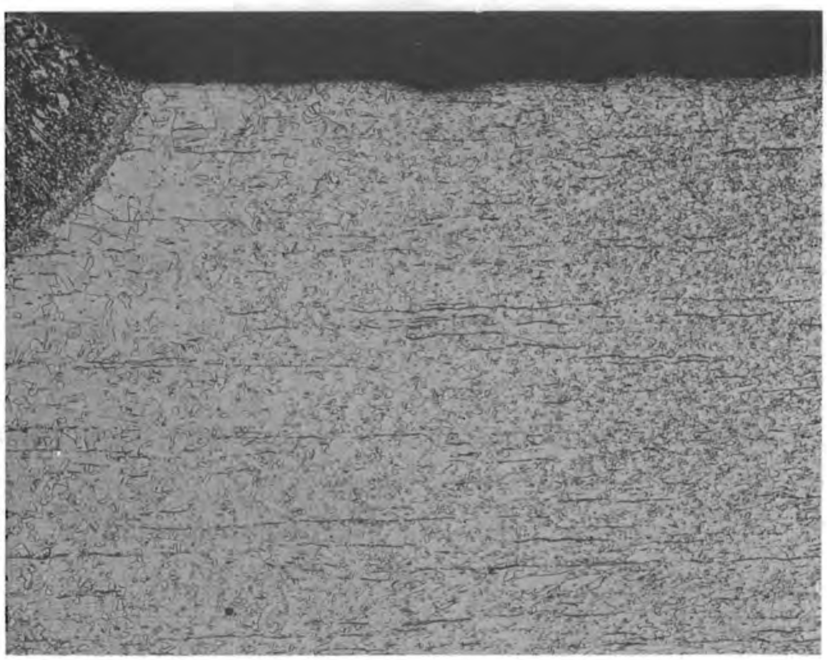

FIGURE 5B. Expanded View Near Weld in Figure 5A (Neg. No. 4P1110B)

Metallography indicated that the crack had propagated by an intergranular mode in a weld heat-affected zone. Investigations are continuing in our laboratory, at Battelle Columbus Laboratories and Ohio State University (the latter two studies are sponsored by the Electric Power Research Institute). The studies are investigating factors that caused the cracks to develop. Cracks have occurred in six other pipes at TMI No. 1, all involving field welds. Weld procedures that lead to severe sensitization appear to be a factor. Whether impurities such as chlorides were involved is still under investigation.

Alternate materials more resistant to weld sensitization should be investigated, at least for PWR spent fuel pool applications.

USE OF ALUMINUM ALLOYS IN SPENT FUEL POOLS

Aluminum alloys have been used as rack materials, grapples, etc. in several spent fuel pools. Table 5 summarizes pools where we are aware of specific experience. Information regarding the specific alloys are not readily available. The most common rack alloys are shown in Table 2.

Very few problems have developed with aluminum alloys in spent fuel pool rack applications. The GE-Morris pool staff saw some corrosion at improperly heat-treated at the WAK pool; some hydrogen evolution occurred when the new racks were placed in the pool water. They were removed and anodized, with no subsequent observation of corrosion. There was some indication that corrosion had occurred on aluminum racks at the Three Mile Island No. 1 (PWR) pool. However, the aluminum corrosion occurred in a beaker test, apparently when boric acid was concentrated over a period of time by water evaporation.

An aluminum alloy fuel rack in the ICPP pool at the Idaho National Engineering Laboratory (INEL) corroded substantially. High chloride levels maintained in the pool water from chlorine treatments to suppress algae appear to account for the aluminum corrosion.

We inspected an aluminum rack removed from the Yankee Rowe (PWR) pool after 17 yr. The rack had a dark-gray, tenacious oxide, except at a few locations where small pits had developed. The rack was structurally sound when removed.

TABLE 5. Aluminum Alloys Fuel Racks in Spent Fuel Pools

Spent Fuel Pool	Location	Pool Chemistry	Time in Pool	Remarks
Studsvik	Sweden		1964 to present	No problems
G. E. - Morris	Illinois, USA	Deionized Water	1972 to 1976	Corrosion at poor welds
WAK	Karlsruhe, GFR		1969 to present[b]	No corrosion after anodizing
Yankee Rowe	Rowe, MA, USA	Boric acid[c]	17 yr; max.	Small amount of pitting - good structural integrity
Three Mile Island (Unit 1)	Harrisburg, PA, USA		1974 to present	No problems - insulated from stainless-steel liner
Oyster Creek	New Jersey, USA		7 yr[d]	No problems
NFS	West Valley, NY, USA	Deionized water	- to present	No problems
RBOF	Savannah River, SC, USA		- to present	No problems
ICPP	Idaho (INEL), USA	(a)	1963 to present	Pitting corrosion

(a) Up to 730 ppm Cl^-, 590 ppm NO_3 ; currently Cl^- is 360 ppm, NO_3 is 430 ppm.
(b) In 1973 one half of the aluminum racks were replaced with stainless steel.
(c) 800 ppm B, maximum
(d) Examination of locations on aluminum racks where they contacted stainless steel pool liner revealed no significant crevice corrosion.

Despite some concerns regarding aluminum alloy integrity, the experience suggests that with proper water chemistry and metallurgical control that aluminum can function satisfactorily in spent fuel pool applications.

SUMMARY - SPENT FUEL POOL EQUIPMENT INVESTIGATIONS

The performance of 304 stainless-steel components in spent fuel pools with deionized water chemistry appears to have been free of significant problems.

The recent emergence of pipe cracking in the TMI No. 1 spent fuel pool piping, together with a few similar cases in other boric acid pools signals the need for identification of the factors involved. Investigations are underway at the laboratories indicated above. Results will be reported at appropriate stages.

While the pipe cracking phenomenon needs to be understood and eliminated, the PWR fuel pool components have generally performed satisfactorily over times up to $\sim$12 yr.

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APPENDIX A

LITERATURE SEARCH TO IDENTIFY SOUND AND DEFECTIVE FUEL ASSEMBLY CANDIDATES

LITERATURE SEARCH TO IDENTIFY SOUND AND DEFECTIVE FUEL ASSEMBLY CANDIDATES

Fuel assemblies with Zircaloy-clad fuel rods that have been discharged from domestic and foreign BWRs are shown in Tables A1 and A2, respectively.

Fuel assemblies with Zircaloy-clad and stainless steel-clad fuel rods discharged from domestic PWRs are shown in Table A3. Fuel assemblies with Zircaloy-clad fuel rods discharged from foreign PWRs are shown in Table A4.

In June 1979, Nuclear Assurance Corporation (NAC) (1979) conducted a computer search for PNL to determine the number of light-water reactor fuel assemblies with stainless steel-clad fuel rods in storage in 1979. The search indicated that nearly 2200 such assemblies are in storage, of which about two-thirds are stored in United States spent-fuel pools. Data from the NAC report (including information on stainless steel-clad fuel and also some on Zircaloy-clad fuel) are shown in Tables A5 to A7. PWR fuel assemblies with Zircaloy-clad fuel rods that have been or are to be discharged from domestic reactors are shown in Table A5. PWR fuel assemblies with stainless steel-clad fuel rods that have been or are to be discharged from domestic and foreign reactors are shown in Tables A6 and A7, respectively.

Additional information on high burnup fuel assemblies with Zircaloy-clad fuel rods that are to be discharged from domestic reactors is shown in Table A8. These fuel assemblies are associated with the EPRI extended-burnup program.

Data from Tables A1 to A8 on fuel that has been or is to be discharged are shown in graph form in Figures A1 to A15. On the graphs, the ordinate is a specified burnup unit and the abscissa is the discharge date.

Average assembly burnups for BWR fuel assemblies with Zircaloy-clad fuel rods from domestic and foreign reactors are shown in Figures A1 and A2, respectively. Average burnups for PWR fuel assemblies with Zircaloy-clad fuel rods from domestic and foreign reactors are shown in Figures A3 and A4, respectively. Average burnups for PWR fuel assemblies with stainless steel-clad fuel rods from domestic and foreign reactors are shown in Figures A5 and A6, respectively.

Actual (A) or estimated (E) peak pellet burnups in BWR and PWR fuel are shown in Figures A7 through A15. The estimated values were calculated by assuming a peak/average ratio of 1.3.

Peak pellet burnups for BWR fuel assemblies with Zircaloy-clad fuel rods from domestic and foreign reactors are shown in Figures A7 and A8, respectively. Peak pellet burnups for PWR fuel assemblies with Zircaloy-clad fuel rods from domestic and foreign reactors are shown in Figures A9 and A10, respectively. Peak pellet burnups for PWR fuel assemblies with stainless-steel-clad fuel rods from domestic and foreign reactors are shown in Figures A11 and A12, respectively.

Figure A13 summarizes the burnup data for domestic and foreign BWR fuel assemblies with Zircaloy-clad fuel rods; Figure A13 includes the data from Figures A1, A2, A7, and A8. Figure A14 summarizes the burnup data for domestic and foreign PWR fuel assemblies with Zircaloy-clad fuel rods; Figure A14 includes the data from Figures A3, A4, A9, and A10. Figure A15 summarizes the burnup data for domestic and foreign PWR fuel assemblies with stainless-steel-clad fuel rods; Figure A15 includes the data from Figures A5, A6, A11, and A12.

A recent report (Turner, Elgin and Hancock 1979) included several figures showing the actual fuel burnup achieved in various discharge batches as determined from a compilation by NAC (using the Fuel-Trac® program). Three of those figures are included in this appendix. Figure A16 shows the burnup in fuel discharged from operating nuclear plants. Figure A17 shows the burnup in fuel from operating boiling water reactor plants. Figure A18 shows the burnup in fuel from operating pressurized water reactor plants.

TABLE A1. United States BWR Fuel

Reactor	Fuel Type	Burnup,[a] (MWd/MTU) Fuel Assembly Average[b]	Peak Pellet	Cladding	Fuel Assembly Characteristics: Active Fuel/Fuel Rod Length (in.)/(m)	Discharged Prior to	Total Rods (Assemblies)/Rods Still in Core/Rods Discharged[c]	Comments	Reference
Dresden - 1	III F	~23,000[h][i]	31,000[h][i]	Zry	108.25/	9/30/74[j]	3744(104)/468/3276[i]		G.G.PSAR,[d] NEDO-21660[e]
	III B	~18,500[i]	27,000[i]	Zry	109/	9/30/74[j]	6912(192)/1080/5832[i]		NEDO-21660
	V	~18,000[i]	28,000[i]	Zry	108.25/	9/30/74[j]	3816(106)/2556/1260[i]		NEDO-21660
Dresden - 2	DN	16,300	23,800	Zry	144/	12/31/76[j]	24,941(509)/13,230/11,711[k]		NEDO-21660
Big Rock Point	B	23,430	35,400	Zry	70/	2/11/71	3630(30)[k]/242[l]/3388[l]		G.G.PSAR, NEDO-21660
Oyster Creek	JC	18,700[n]	30,900[n]	Zry	144/	3/29/76[j]	27,440(560)[k]/9016[m]/18,424[m]		NEDO-21660
Nine Mile Point	GEA	19,700	28,400	Zry	144/	12/31/76[j]	1960(40)/1666/294[k]		NEDO-21660
Dresden - 1	SA-1[n]	(o)	40,000[r]	Zry		9/69[p]	98(1)/0/98	(q)	G.G.PSAR, NEDO-21660, GEAP-10217,[f] NEDO-10173[j]
Big Rock Point	D-3[n]		30,000	Zry	/~72	5/70	121(1)/0/121	(s)	NEDO-21660, NEDO-10173
	D-1, D-2[n]	~18,000		Zry	/~72	2/68	242(2)/0/242	(t)	NEDO-10173, NEDO-21660

(a) Underlined burnup values were stated in reference(s).
(b) As stated in NEDO-21660, this is the assembly average burnup for those assemblies remaining in the reactor core or assembly average discharge burnup when no assemblies remain in the core.
(c) Total no. of fuel rods (associated no. of fuel assemblies)/no. of fuel rods still in the reactor core/no. of fuel rods discharged.
(d) Preliminary Safety Analysis Report, Grand Gulf Nuclear Station Units 1 and 2, Docket No. 50-416/-417, November 17, 1972. (Vol. 3, Section 4.0, pp. 4.2-13 through 4.2-16 and Tables 4.2.2 through 4.2.5. Note: Tables 4.2.2 and 4.2.3 summarize experience with production Zircaloy-clad UO2 fuel as of October 1, 1971.)
(e) R. B. Elkins, Experience with BWR Fuel Through December 1976. NEDO-21660, July 1977.
(f) F. H. Megerth (comp.), Zircaloy-Clad UO2 Fuel Rod Evaluation Program, Quarter Report No. 11, May-July 1970. GEAP-10217, August, 1970. (p. 44)
(g) H. E. Williamson and D. C. Ditmore, Current State of Knowledge High-Performance BWR Zircaloy-Clad UO2 Fuels. NEDO-10173, May 1970. (pp. 4-6 and 4-8).
(h) On p. 4-15 of the Grand Gulf PSAR,[d] it states that the maximum Type III F assembly average exposures have reached 26,000 MWd/T with peak fuel segments having attained 37,000 MWd/T.
(i) Information as of 9/30/74.
(j) Some fuel assemblies still in the core as of this date.
(k) Information as of 12/31/76.
(l) Information as of 2/11/71.
(m) Information as of 3/29/76.
(n) Fuel assembly identification numbers.
(o) GEAP-10217[f] indicates that the fuel rod average burnup is $\leq$29,700 MWd/MTU.
(p) Actual discharge date.
(q) SA-1 was disposed of as required by AEC contract.
(r) In the Grand Gulf PSAR,[d] a value of 40,000 MWd/Te is shown in Table 4.2.4 and of approximately 37,900 MWd/Te on p. 4.2-16.
(s) Fuel Assemblies D-1, -2, and -3 were transferred to the NFS West Valley Facility when the BRP pool was to be lined. These assemblies can be readily disassembled.
(t) D-2 was detected as a "leaker" at time of discharge; however, no positive visual evidence of failure was noted during detailed visual examination at the site.

TABLE A2. Foreign BWR Fuel

Reactor	Fuel Type	Burnup,[a] (MWd/MTU) Fuel Assembly Average[b]	Burnup,[a] (MWd/MTU) Peak Pellet	Fuel Assembly Characteristics Clad-ding	Fuel Assembly Characteristics Active Fuel/Fuel Rod Length (in.)	Fuel Assembly Characteristics Active Fuel/Fuel Rod Length (m)	Discharged Prior to	Total Rods (Assemblies)/Rods Still in Core/Rods Discharged[c]	Reference
Gargliano	SA	19,600	29,500	Zry	107/		12/31/76[d]	4,244(66)/3200/1044[e]	NEDO-21660[f]
Gargliana	A	14,300	26,600	Zry	105.7/		12/31/76	16,848(208)/0/16,848	NEDO-21660
Gargliano	A	15,180	26,160	Zry	105.7/		10/1/71[d]	16,848(208/6804/10,044	NEDO-10505[g]
KRB	A	14,634	22,409	Zry	130.0/		10/1/71[d]	13,356(371)/5784/7572	NEDO-10505
Tsuruga	JAA	21,500	30,600	Zry	144/		12/31/76[d]	2,352(48)/882/1470[e]	NEDO-21660

(a) Underlined burnup values were stated in reference(s).
(b) As stated in NEDO-21660[d]
(c) Total no. of fuel rods (associated no. of fuel assemblies)/no. of fuel rods still in the reactor core/no. of fuel rods discharged.
(d) Some fuel assemblies still in the core as of this date.
(e) Information as of 12/31/76.
(f) R. B. Elkins, Experience with BWR Fuel Through December 1976. NEDO-21660, July 1977.
(g) H. E. Williamson and D. C. Ditmore, Experience with BWR Fuel Through September 1971. NEDO-10505, May 1972.

TABLE A3. United States PWR Fuel

Reactor	Region No.	Burnup,[a] (MWd/MTU): Fuel Assembly Average[b]	Burnup,[a] (MWd/MTU): Peak Pellet	Fuel Assembly Characteristics: Cladding	Pre-press.[c]	Active Fuel/ Fuel Rod Length (in.)	(m)	Discharged Prior To	Rods(Assemblies) Discharged[d]	Reference	Comment
Ginna	1	21,120	~27,500	Zry-4	No		3.68/	4/1/74	7,339(41)	WCAP-8183,Rev.1[e]	
	5	25,250	~32,800	Zry-4	Yes		3.68/	12/31/77	8,592(48)	WCAP-8183,Rev.7[f]	
Robinson-2	2	26,700	~34,700	Zry	Yes	144/	3.66/	10/1/74	10,608(52)	WCAP-8183,Rev.2[g]	
	3	22,940	~29,800	Zry	Yes	144/	3.66/	10/1/74	10,608(52)	WCAP-8183,Rev.2	
	4	24,500	~31,800	Zry	Yes	144/	3.66/	12/31/76	10,812(53)	WCAP-8183,Rev.6[h]	
Point Beach-1	1	20,610	~26,800	Zry	No	144/	3.66/	12/31/76	8,771(49)	WCAP-8183,Rev.6	
	2	30,860	~40,100	Zry	Yes	144/	3.66/	12/31/76	7876(44)	WCAP-8183,Rev.6	
	2	31,000	~40,300	Zry	Yes	144/	3.66/	10/1/74	7160(40)	WCAP-8183,Rev.2	
	3	24,600	~32,000	Zry	Yes	144/	3.66/	10/1/74	7160(40)	WCAP-8183,Rev.2	
	4	25,850	~33,600	Zry	Yes	144/	3.66/	12/31/75	7876(44)	WCAP-8183,Rev.4[i]	
	5	23,500	~30,600	Zry	Yes	144/	3.66/	12/31/76	7339(41)	WCAP-8183,Rev.6	
Point Beach-2	2	28,610	~37,200	Zry	Yes	144/	3.66/	6/30/76	7,160(40)	WCAP-8183,Rev.5[j]	
	3	34,760	~45,200	Zry	Yes	144/	3.66/	12/31/77	7,160(40)	WCAP-8183,Rev.7	
Surry-1	2	23,080	~30,000	Zry	Yes	144/	3.66/	6/30/76	10,608(52)	WCAP-8183,Rev.5	
	3	22,270	~29,000	Zry	Yes	144/	3.66/	12/31/76	10,608(52)	WCAP-8183,Rev.6	

TABLE A3. (contd)

Reactor	Region No.	Burnup,[a] (MWd/MTU): Fuel Assembly Average[b]	Burnup,[a] (MWd/MTU): Peak Pellet	Fuel Assembly Characteristics: Cladding	Fuel Assembly Characteristics: Pre-Press.[c]	Fuel Assembly Characteristics: Active Fuel/Fuel Rod Length (in.)	Fuel Assembly Characteristics: Active Fuel/Fuel Rod Length (m)	Discharged Prior To	Rods(Assemblies) Discharged[d]	Reference	Comment
Surry-2	2	22,590	~29,400	Zry	Yes	144/	3.66/	6/30/76	10,608(52)	WCAP-8183,Rev.5	
	2	26,000	~33,800	Zry	Yes	144/	3.66/	4/22/76		(k)	
Turkey Point-3	3	24,190	~31,400	Zry	Yes	144/	3.66/	12/31/76	10,608(52)	WCAP-8183,Rev.6	
	3	29,140	~37,900	Zry	Yes	144/	3.66/	12/31/77	10,608(52)	WCAP-8183,Rev.7	
Kewaunee	2	29,780	~38,700	Zry	Yes		3.7/	12/31/77	7,160(40)	WCAP-8183,Rev.7	
Yankee Rowe		31,000[q]	46,000[q]	348SS		94/		8/65[r]	(1)	(1,m,n)	(s)
		29,493[t]		SS		94/		2/72	(1)		(t)
		28,986[t]		SS		94/		2/72	(1)		(t)
		25,962[t]		SS		94/		2/72	(1)		(t)
		23,504[t]		SS		94/		2/72	(1)		(t)
Haddam Neck[aa]		25,000	~35,000[u]	SS	Yes	121.8/		4/72		(o)	
San Onofre		~16,000	~34,000[u]	SS	Yes	120.0/		4/72		(o)	
Indian Point-1		24,000[v]	39,000[v]	304SS	No	98.5/		3/66		WASH-1082[m]	
Saxton		(w)	40,000 to 51,000[x]	Zry	No	/39		5/1/72[r]	~250(7)	WCAP-3385-57[p]	
Shippingport		na[y]	36,600	Zry-2	na			3/66		(1,m)	(z)

TABLE A3. (contd)

(a) Underlined burnup values were stated in reference(s). Nonunderlined burnup values were calculated by assuming a peak/average ratio of 1.3.

(b) Region average discharge burnup, unless otherwise noted.

(c) Were fuel rods prepressurized?

(d) No. of fuel rods (associated no. of fuel assemblies) discharged.

(e) V. J. Plocido, R. E. Schreiber, and J. Skaritka, Operational Experience with Westinghouse Cores (up to April 1974). WCAP-8183, Rev. 1, July 1974.

(f) T. L. O'Hara and J. A. Iorii, Operational Experience with Westinghouse Cores (up to December 13, 1977). WCAP-8183, Rev. 7, March 1978.

(g) V. J. Plocido and R. E. Schreiber, Operational Experience with Westinghouse Cores (up to October 1974). WCAP-8183, Rev. 2, November 1974.

(h) R. E. Schreiber and J. A. Iorii, Operational Experience with Westinghouse Cores (up to December 31, 1976). WCAP-8183, Rev. 6, June 1977.

(i) R. E. Schreiber, V. J. Plocido, and J. A. Iorii, Operational Experience with Westinghouse Cores (up to December 31, 1975). WCAP-8183, Rev. 4, March 1976

(j) R. E. Schreiber and J. A. Iorii, Operational Experience with Westinghouse Cores (up to June 30, 1976). WCAP-8183, Rev. 5, September 1976

(k) James T. Rhoades, "Fuel Performance of Surry 1 and 2", ANS Topical Meeting on Water Reactor Fuel Performance, St. Charles, Illinois, May 1977, (pp. 7, 9, and 10)

(l) Glen Reed and Edmund Tarnuzzer, "Examining Yankee Plant Performance in 1965", Nucleonics 24(3):42-47, March 1966.

(m) Jackson and Moreland (Division of United Engineers and Constructors, Inc.) and S. M. Stoller Associates, Current Status and Future Technical and Economic Potential of Light Water Reactors. WASH-1082, March 1968. (pp. 5-38 and 5-39/5-40)

(n) Leon Joseph, "Performance of Fuel Elements in Nuclear Power Plants", Nucleonics 24(3):51-54, March 1966.

(o) W. J. Dollard and F. W. Kramer, "Westinghouse Nuclear Fuel Operating Experience", American Power Conference, Chicago, Illinois, April 18-20, 1972

(p) W. R. Smalley, Evaluation of Saxton Core III Fuel Materials Performance, WCAP-3385-57, July 1974.

(q) Average burnup of Fuel Assembly A8 was 31,000 MWd/MTU[l,m] ; the peak single-rod burnup of Fuel Assembly A8 was $\sim$46,000 MWd/MTU[l,m,n]

TABLE A3. (contd)

(r) Actual date reactor shut down.

(s) Fuel Assembly A8 not available; it was shipped to reprocessor (probably NFS) in September 1965.

(t) Four fuel assemblies (B309, B300, B305, B311) have been stored in the Yankee Rowe spent fuel pool since February 1972. The respective burnups are 29,493 ; 28,986; 25,962; and 23,504 MWd/MTU.

(u) Fuel assembly peak burnup.

(v) Burnup values in MWd/MT; fuel was ThO_2-UO_2; average assembly burnup for the peak fuel assembly was 24,000 MWd/MT.[m]

(w) Estimated fuel rod peak/average burnup ratios were 1.2 to 1.3.[k] Core average burnup was 8082 MWd/MTM; however, the UO_2-PuO_2 fuel rods (~250) were only in 7 of the 21 fuel assemblies.

(x) Burnup in MWd/MTM; fuel was UO_2-PuO_2.

(y) na = data not available.

(z) Blanket rods.

(aa) Also known as Connecticut Yankee.

TABLE A4. Foreign PWR Fuel

Reactor	Region No.	Burnup,[a] (MWd/MTU): Fuel Assembly Average[b]	Burnup,[a] (MWd/MTU): Peak Pellet	Fuel Assembly Characteristics: Cladding	Pre-press.[c]	Active Fuel/Fuel Rod Length (in.)	Active Fuel/Fuel Rod Length (m)	Discharged Prior To	Rods (Assemblies) Discharged[d]	Reference	Comments
Cabrera (Zorita)	1	25,950	~33,700	Zry	No	96/	2.43/	4/1/74	3,580(20)	WCAP-8183,Rev.1[e]	
	2	24,350	~31,700	Zry	No	96/	2.43/	4/1/74	4,117(23)	WCAP-8183,Rev.1	
	4	28,150	~36,600	Zry	Yes	96/	2.43/	12/31/75	3,222(18)	WCAP-8183,Rev.4[f]	
	5	28,660	~37,300	Zry	Yes	96/	2.43/	12/31/77	6,265(35)	WCAP-8183,Rev.7[g]	
Beznau-1	1	21,700	~28,200	Zry	No	120/	3.05/	4/1/74	7,339(41)	WCAP-8183,Rev.1	
	2	20,280	~26,400	Zry	No	120/	3.05/	4/1/74	7,160(40)	WCAP-8183,Rev.1	
	4	28,440	~37,000	Zry	Yes	120/	3.05/	12/31/77	7,876(44)	WCAP-8183,Rev.7	
	5	27,790	~36,100	Zry	Yes	120/	3.05/	12/31/77	6,444(36)	WCAP-8183,Rev.7	
Beznau-2	2	28,050	~36,500	Zry	Yes	120/	3.05/	12/31/76	10,024(56)	WCAP-8183,Rev.6[h]	
	2	29,670	~38,600	Zry	Yes	120/	3.05/	12/31/77	7,160(40)	WCAP-8183,Rev.7	
	3	28,230	~36,700	Zry	Yes	120/	3.05/	12/31/75	7,160(40)	WCAP-8183,Rev.4	
	4	25,590	~33,300	Zry	Yes	120/	3.05/	12/31/76 & 12/31/77	5,728(32) 7,160(40)	WCAP-8183,Rev.6 WCAP-8183,Rev.7	
Takahama-1	2	26,820	~34,900	Zry	Yes	144/150	3.66/3.80	12/29/75	10,608(52)	WCAP-8183,Rev.7	
	3	21,760	~28,300	Zry	Yes	144/150	3.66/3.80	12/29/75	10,608(52)	WCAP-8183,Rev.7	
Cabrera (Zorita)		58,000[j]	65,000[k]	Zry	Yes	96/	2.43	end of 1976[l]		(i)	(m)

Footnotes: See next page.

TABLE A4. (contd)

(a) Underlined burnup values were stated in reference(s). Nonunderlined burnup values were calculated by assuming a peak/average ratio of 1.3.

(b) Region average discharge burnup, unless otherwise noted.

(c) Were fuel rods prepressurized?

(d) No. of fuel rods (associated no. of fuel assemblies) discharged.

(e) V. J. Plocido, R. E. Schreiber, and J. Skaritka, Operational Experience with Westinghouse Cores (up to April 1974). WCAP-8183, Revision 1, July 1974.

(f) R. E. Schreiber, V. J. Plocido, and J. A. Iorii, Operational Experience with Westinghouse Cores (up to December 31, 1975). WCAP-8183, Revision 4, March 1976.

(g) T. L. O'Hara and J. A. Iorii, Operational Experience with Westinghouse Cores (up to December 13, 1977). WCAP-8183, Revision 7, March 1978.

(h) R. E. Schreiber and J. A. Iorii, Operational Experience with Westinghouse Cores (up to December 31, 1976). WCAP-8183, Revision 6, June 1977.

(i) E. Roberts, et al., "Fuel Modeling and Performance of High Burnup Fuel Rods", ANS Topical Meeting on Water Reactor Fuel Performance, St. Charles, Illinois, May 1977. (pp. 133-135)

(j) Peak rod average burnup was as high as 58,000 MWd/MTU.[i]

(k) These fuel rods have achieved the highest burnup to date (i.e., May 1977) in any commercial reactor.[i]

(l) Irradiation period was 1968-1976.[i]

(m) Five or six fuel rods are in dry storage (two rods have average burnups of ∿58,000 MWd/MTU).

TABLE A5. Data from NAC Report[(a)] on PWR Fuel Assemblies with Zircaloy-Clad Fuel Rods That Have Been or Are To Be Discharged from U.S. Reactors

Average Burnup,[(b)] (MWd/MTM)	Discharge Date	Reactor	No. of Fuel Assemblies	Fuel Fabricator(s)
23,900-28,400	10/75-8/81	Yankee Rowe	188	Exxon
26,200-30,400	7/73-5/75	Connecticut Yankee	4	NUMEC and Gulf-General Atomic

TABLE A6. Data from NAC Report[(a)] on PWR Fuel Assemblies with Stainless-Steel-Clad Fuel Rods That Have Been or Are To Be Discharged from U.S. Reactors

Average Burnup,[(b)] (MWd/MTM)	Discharge Date	Reactor	No. of Fuel Assemblies	Fuel Fabricator(s)
22,600-37,000	2/72-10/75	Yankee Rowe	76	Westinghouse and GNFC
25,000-25,200	12/72[(c)]-11/74	Indian Point-1	160	Westinghouse
3,000-36,600	10/70-9/81	San Onofre-1	470[(d)]	Westinghouse
19,500-37,500	4/70-6/81	Connecticut Yankee	573[(d)]	Westinghouse, Gulf-General Atomic, and B&W

TABLE A7. Data from NAC Report[(a)] on PWR Fuel Assemblies with Stainless Steel-Clad Fuel Rods That Have Been or Are To Be Discharged from Foreign Reactors

Average Burnup,[(b)] (MWd/MTM)	Discharge Date	Reactor	No. of Fuel Assemblies	Fuel Fabricator(s)
18,900-32,000	3/73-8/82	Trino Vercellese	354	Westinghouse and COREN
7,700-33,000	6/72-4/82	Sena Chooz	354[(c)]	Westinghouse, W-CERCA, W-MMN, Framatome, and BCR

(a) Source: Fuel-Trac,® Nuclear Assurance Corporation, "Current Status of Stainless Steel Clad Light Water Reactor Fuel Assemblies," June 1979. (Data revised on basis of recent data from NAC.)
(b) Estimated burnup at discharge is used for fuel that is to be discharged at a future date.
(c) Operation of Indian Point-1 suspended in November 1974.
(d) Includes some mixed oxide fuel.

TABLE A8. EPRI Extended-Burnup Program (Table is from EPRI NP-1024-SR; the Burnup Number Represents the Expected Fuel Assembly Average Burnup at the End of the Cycle)

Project	Vendor	Utility/Reactor	Fuel array/ Reactor type	1977	1978	1979	1980	1981	1982	1983
RP510	General Electric Co	Philadelphia Electric Co Peach Bottom-2	8 x 8 BWR	Cycle 1	Cycle 2	Cycle 3 24,000 MWd/t		Cycle 4 32,000 MWd/t		Cycle 5 40,000 MWd/t
RP895	Exxon Nuclear Co., Inc	Jersey Central Power & Light Co Oyster Creek	8 x 8 BWR		Cycle 1	Cycle 2	Cycle 3	26,000 MWd/t	Cycle 4 36,000 MWd/t	Cycle 5 >40,000 MWd/t
RP586	Combustion Engineering Inc	Baltimore Gas and Electric Co Calvert Cliffs-1	14 x 14 PWR	Cycle 1	Cycle 2	Cycle 3 35,000 MWd/t	Cycle 4 45,000 MWd/t	Cycle 5 55,000 MWd/t		
RP611	Westinghouse Electric Corp	Commonwealth Edison Co Zion-1 Zion-2	15 x 15 PWR	Cycle 2	Cycle 3 39,000 MWd/t		Cycle 4 48,000 MWd/t	Cycle 5 55,000 MWd/t		
RP611	Westinghouse Electric Corp	Portland General Electric Co Trojan	17 x 17 PWR		Cycle 1	Cycle 2 20,000 MWd/t	Cycle 3	Cycle 4 40,000 MWd/t	Cycle 5 48,000 MWd/t	

— — — Planned but not committed

● Fuel inspection point

TABLE A9. Some Examples of Fuel with Defects

Reactor	Description of Defect	Reference
Domestic:		
Oconee-1	Fuel Assembly 1A19 (once-burned fuel) had a 1/4-in. diameter hole in a peripheral rod	Docket 50269-421,-358
Browns Ferry-2	Fuel Assembly N-10 had failure site ~0.1 in. in diameter surrounded by a 0.25 in. diameter hydride area	Docket 50281-335
San Onofre-1	Two damaged fuel rods noted	(a)
Dresden-3	Rod A-3 in Fuel Assembly DD418 had a 6-in. long crack; also, a section of cladding (~1 in. long) was missing.	Docket 50249-1074
Calvert Cliffs-1	Fuel Assembly 1B060 has a bowed peripheral rod with an apparent failure	Licensee Event Report (LER)78-11
Fort Calhoun	Fuel Assembly C004 has swollen fuel rod (Pin 6 is swollen above the upper retention grid.	LER76-37
Millstone-1	Fuel assembly bowed considerably over its entire length	Docket 50245-421 (A050-245/74-5)
Connecticut Yankee	Four Batch 8 fuel assemblies (burnups of 34,300-35,800 MWd/MTU) have cracks in the cladding of peripheral fuel rods.	LER79-01/1T
Oyster Creek	Seven fuel assemblies (burnups ranged between 17,000 and 24,000 MWd/MTU) each had a single visibly failed fuel rod. One failed rod had a clearly defined through-wall hole. Other failed rods had axial cracks ranging from several inches to as as much as 24 inches in length; one rod had an axial crack in the plenum region.	(b)
Yankee Rowe	Visible evidence of fuel rod failure detected in two fuel assemblies (burnup was 30,000 MWd/MTU). Through-wall fretting wear observed in vicinity of top two spacer grids.	(b)
Point Beach-1	Two fuel rods in Fuel Assembly D-03 were extensively damaged; 11 in. of one rod was missing	Docket 50266-362 and 50266-393; LER 50266/75-18

TABLE A9. (contd)

Reactor	Description of Defect	Reference
Domestic:		
LaCrosse (LACBWR)	Four fuel assemblies (I-41,-52,-57, and -59) with stainless steel-clad fuel rods have visible defects; fuel assembly burnups are 20,478-21,532 MWd/MTU.	Docket 50409-276
LaCrosse (LACBWR)	Three of 26 damaged fuel assemblies had some sections of a total of seven stainless steel-clad fuel rods missing (a total length of about 55 in. of fuel rod is missing); average burnup for the 26 fuel assemblies is >16,000 MWd/MTU.	NUREG-0090-8
Foreign:		
Muhleberg (BWR)	Outer fuel rod had spiral crack that was 60 cm long	Docket 50271-518 and 50293-451, (c), and (d)
Zorita (PWR)	One peripheral fuel assembly had two broken fuel rods and one severely damaged rod.	(a)
Mihama-1(PWR)	Fuel damage caused by fretting corrosion. At least two fuel rods were broken.	(e), (f)

(a) W. J. Dollard and F. W. Kramer, "Westinghouse Nuclear Fuel Operating Experience", American Power Conference, April 1972.

(b) G. A. Sofer and K. N. Woods, "Non-destructive Examination of Exxon Nuclear Fuel in LWR Reactors", Am. Nucl. Soc. Topical Meeting on Light Water Reactor Fuel Performance, Portland, Oregon, April 29 - May 3, 1979 (pp39-48)

(c) "Selected Safety-Related Occurrences Reported in November and December 1973," Nuclear Safety, p. 210, March-April 1974.

(d) "Operating Experience with PWR and BWR Stations in Switzerland," Nuclear Engineering International, pp. 561-565, June-July 1975.

(e) Nucleonics Week, 18(4):9, January 27, 1977.

(f) Nucleonics Week, 18(5):8, February 3, 1977.

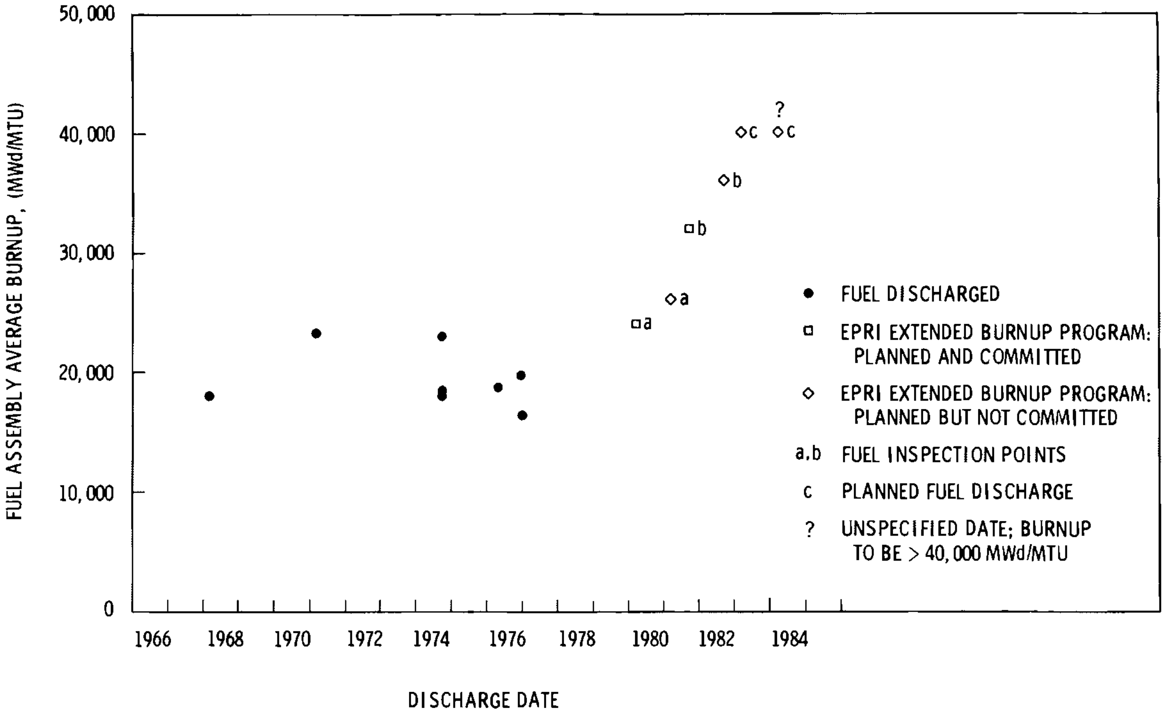

FIGURE A1. Fuel Assembly Average Burnup for U.S. BWR Fuel Assemblies with Zircaloy-Clad Fuel Rods

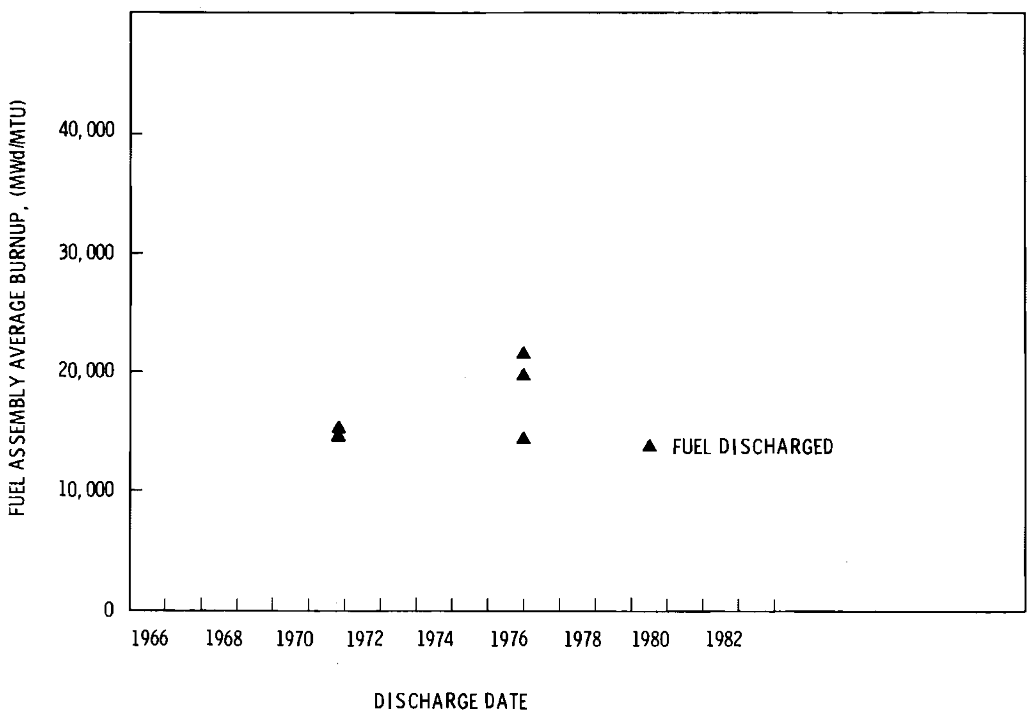

FIGURE A2. Fuel Assembly Averages Burnup for Foreign BWR Fuel Assemblies with Zircaloy-Clad Fuel Rods

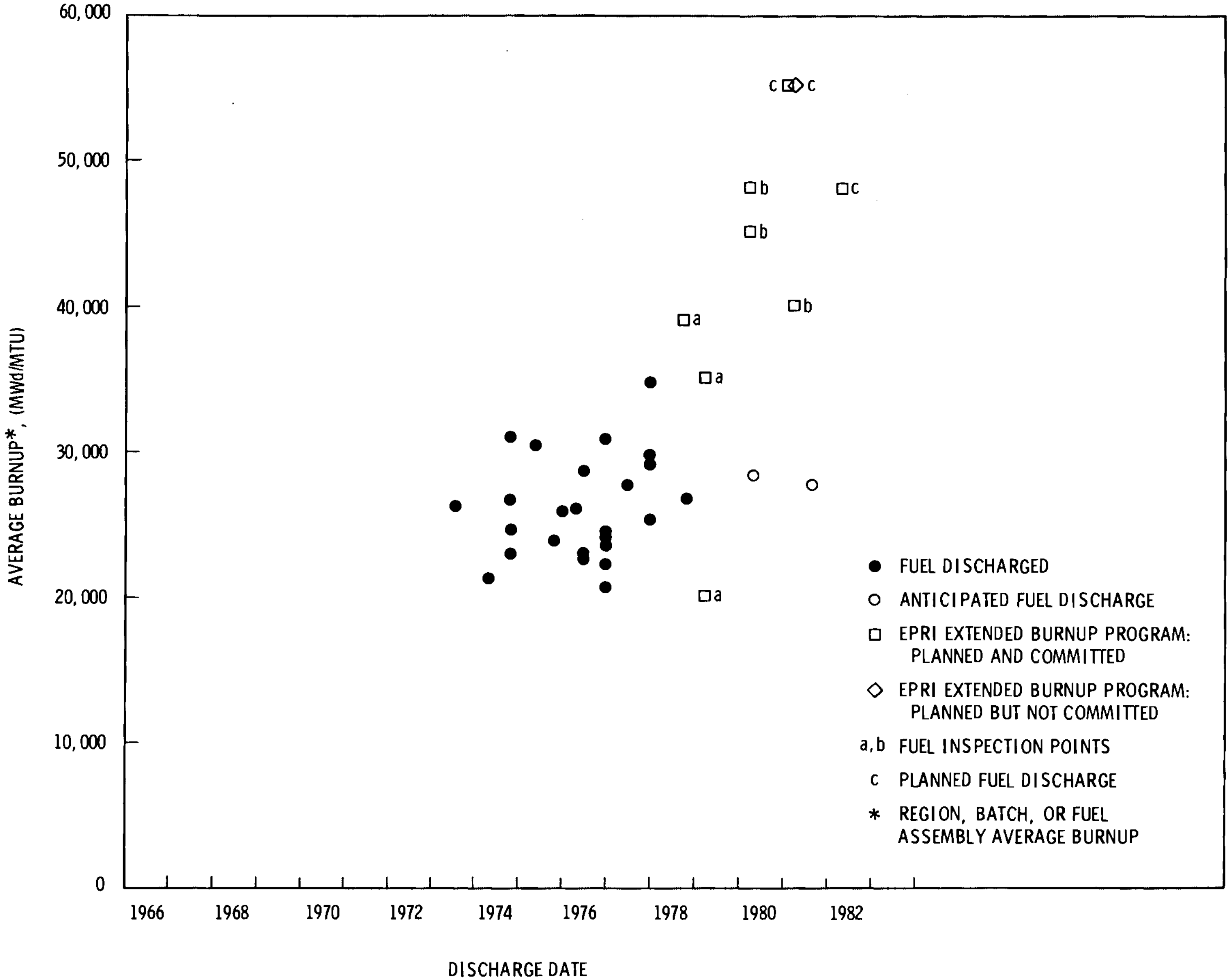

FIGURE A3. Average Burnup for U.S. PWR Fuel Assemblies With Zircaloy-Clad Fuel Rods

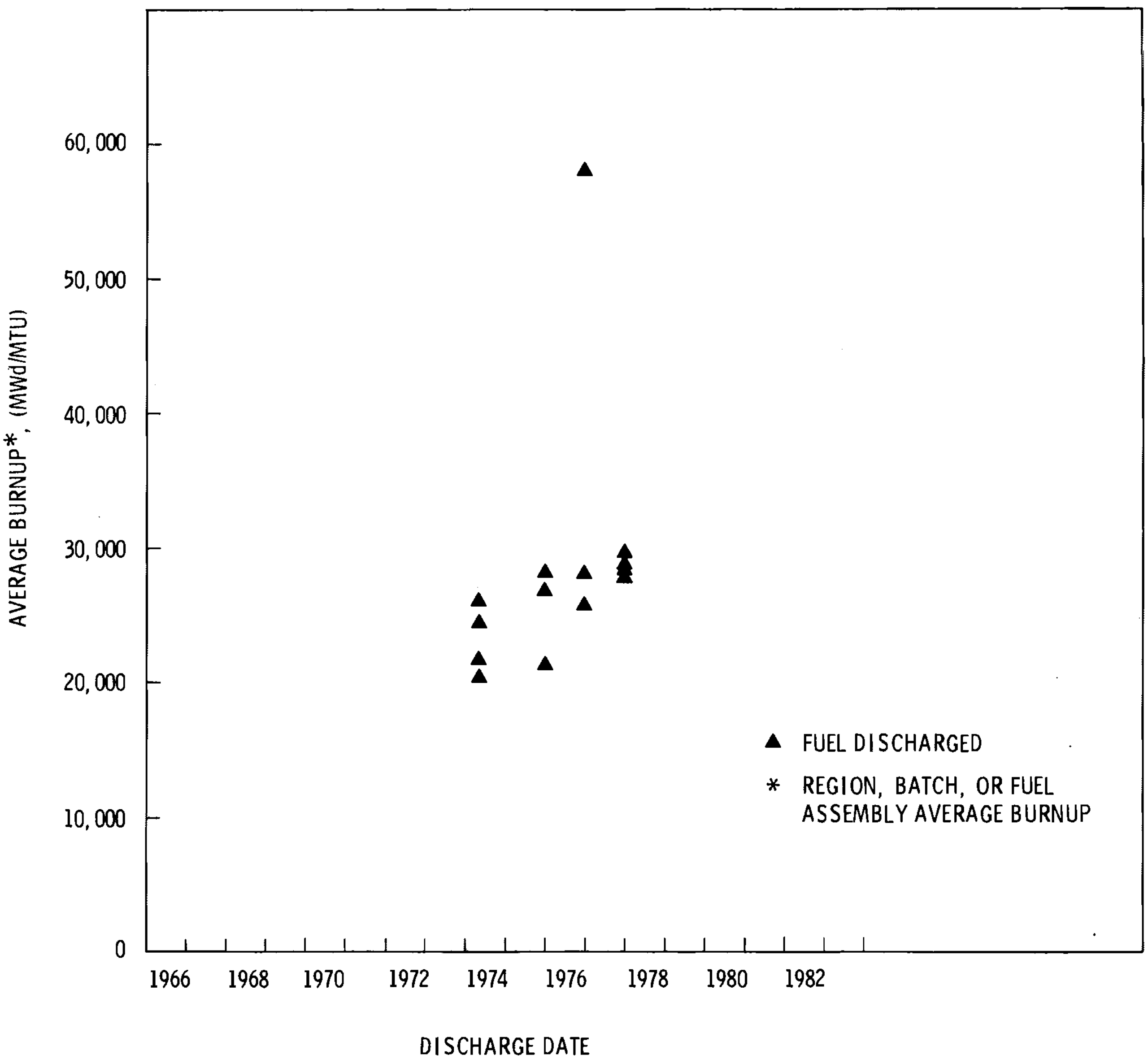

FIGURE A4. Average Burnup of Foreign PWR Fuel Assemblies with Zircaloy-Clad Fuel Rods

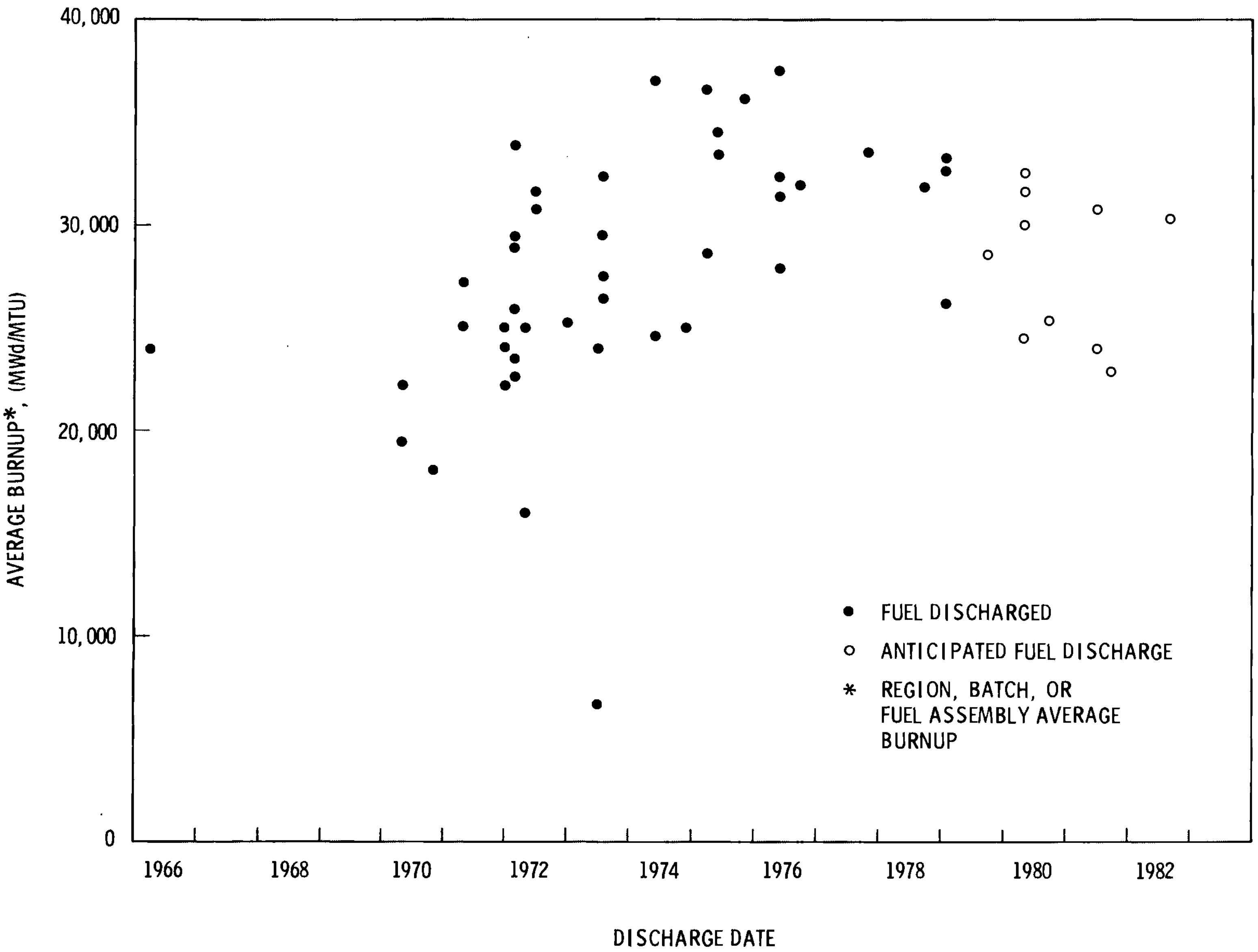

FIGURE A5. Average Burnup for U.S. PWR Fuel Assemblies with Stainless-Steel-Clad Fuel Rods

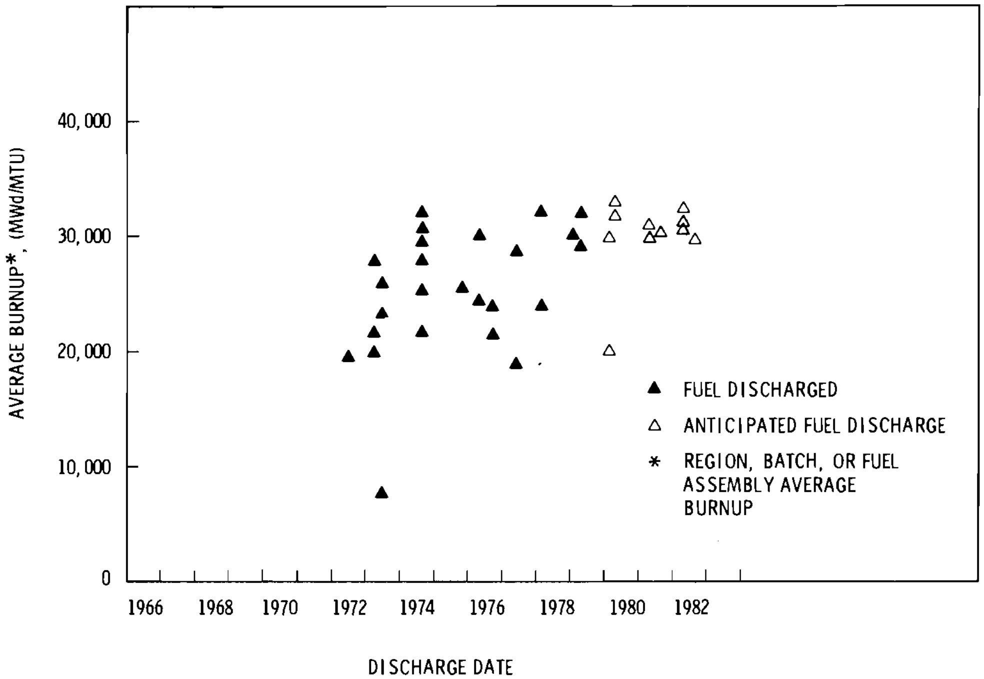

FIGURE A6. Average Burnup for Foreign PWR Fuel Assemblies with Stainless-Steel-Clad Fuel Rods

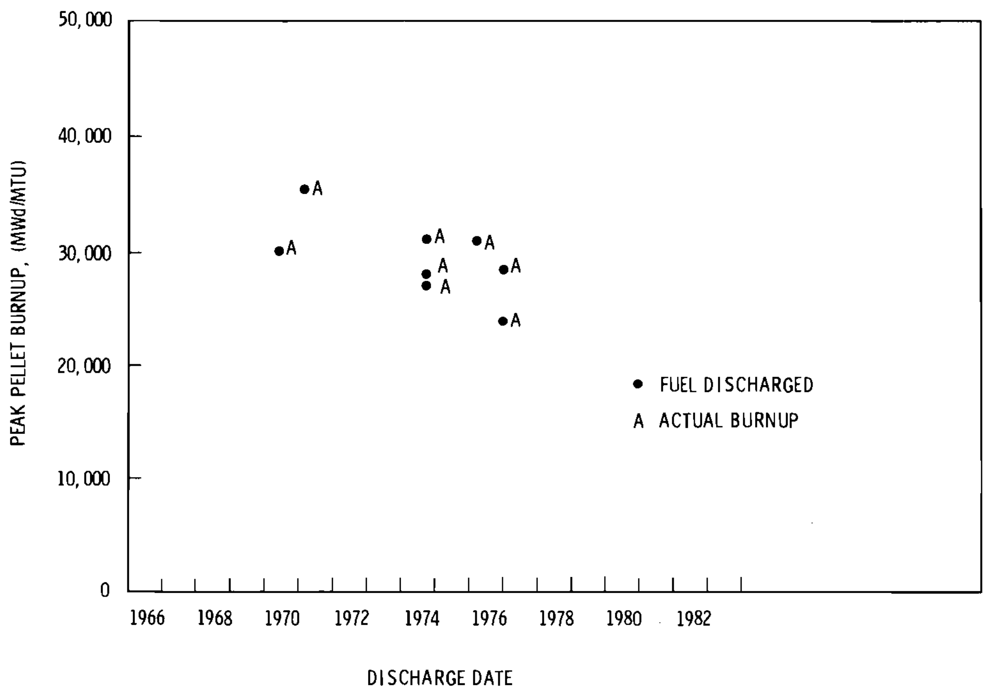

FIGURE A7. Peak Pellet Burnup for U.S. BWR Fuel Assemblies with Zircaloy-Clad Fuel Rods

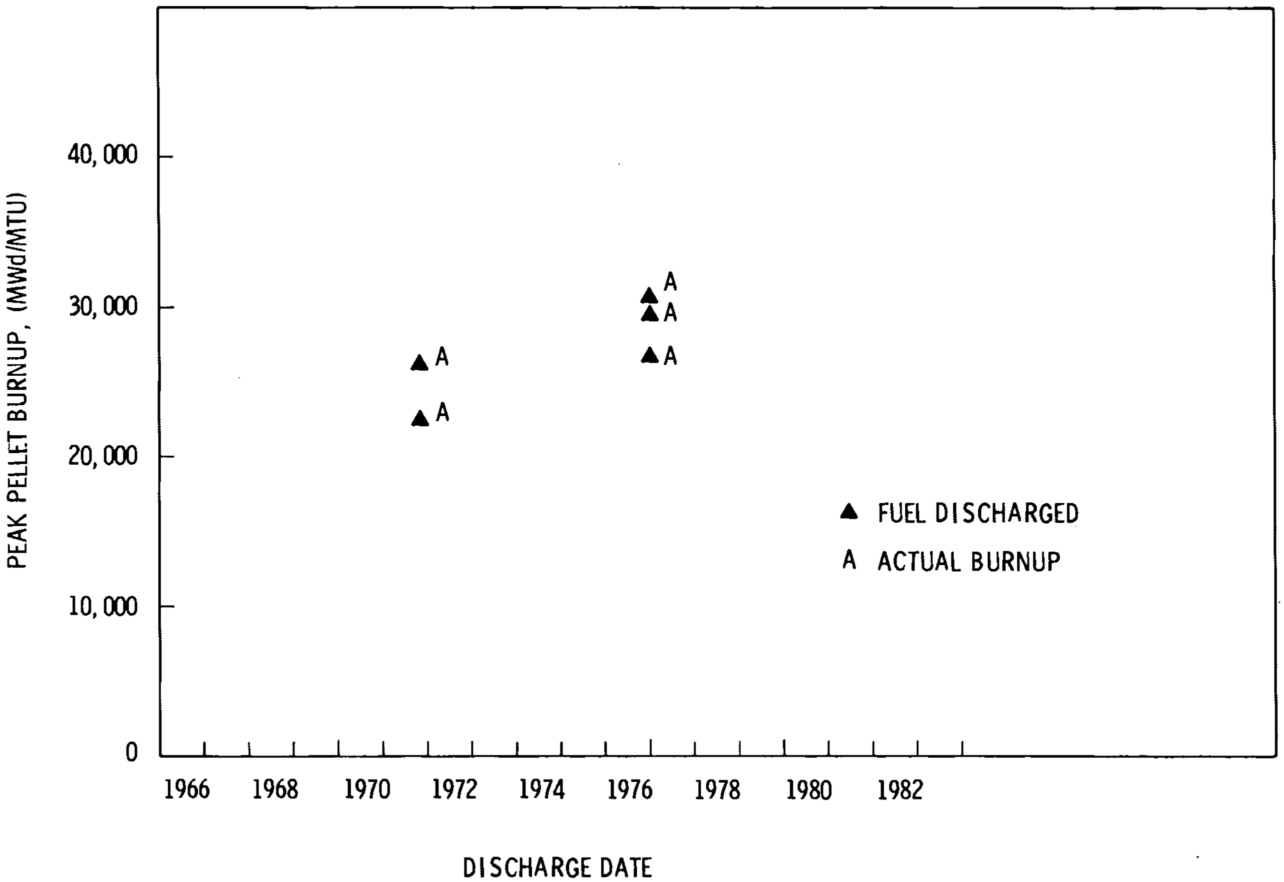

FIGURE A8. Peak Pellet Burnup for Foreign BWR Fuel Assemblies with Zircaloy-Clad Fuel Rods

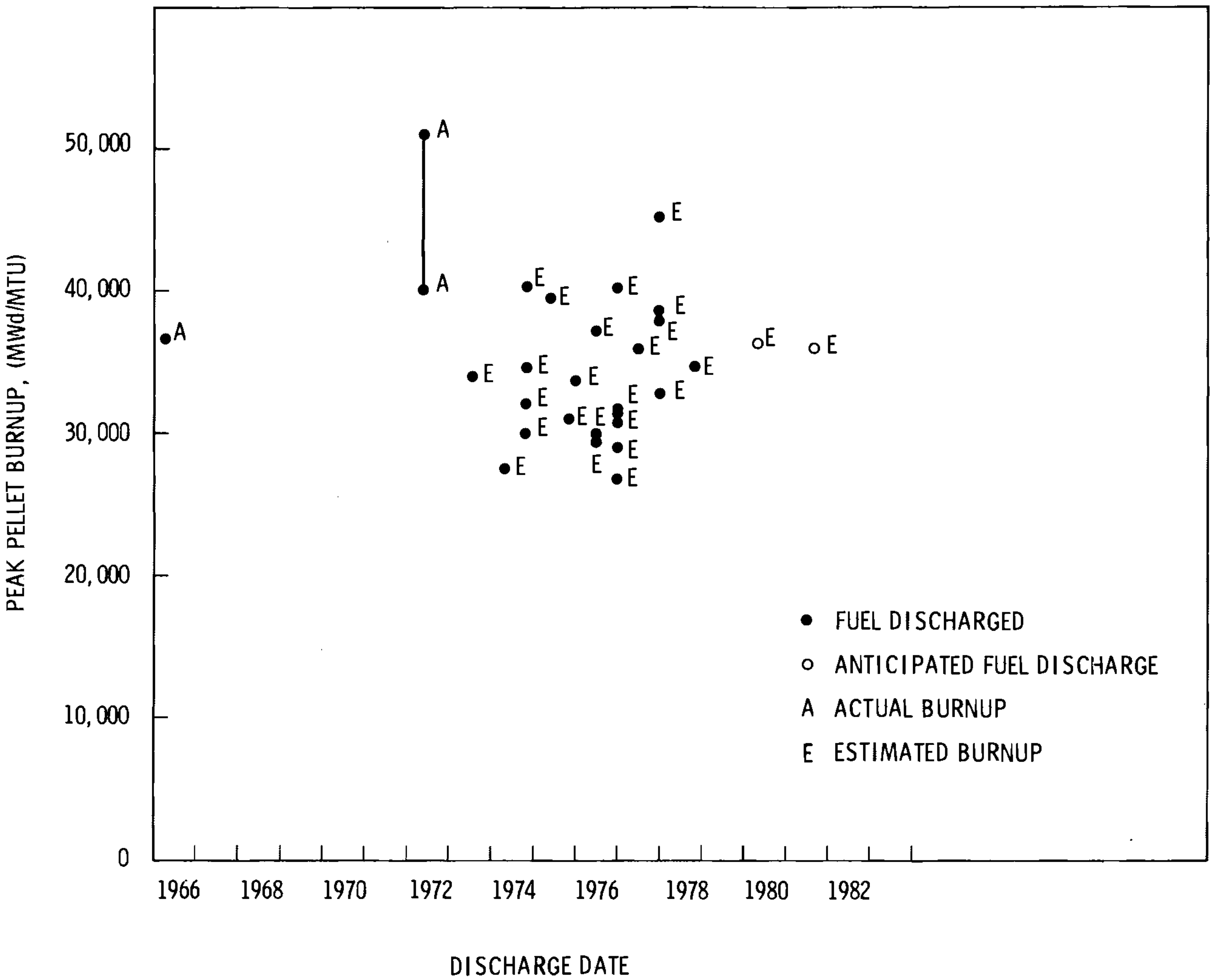

FIGURE A9. Peak Pellet Burnup for U.S. PWR Fuel Assemblies with Zircaloy-Clad Fuel Rods

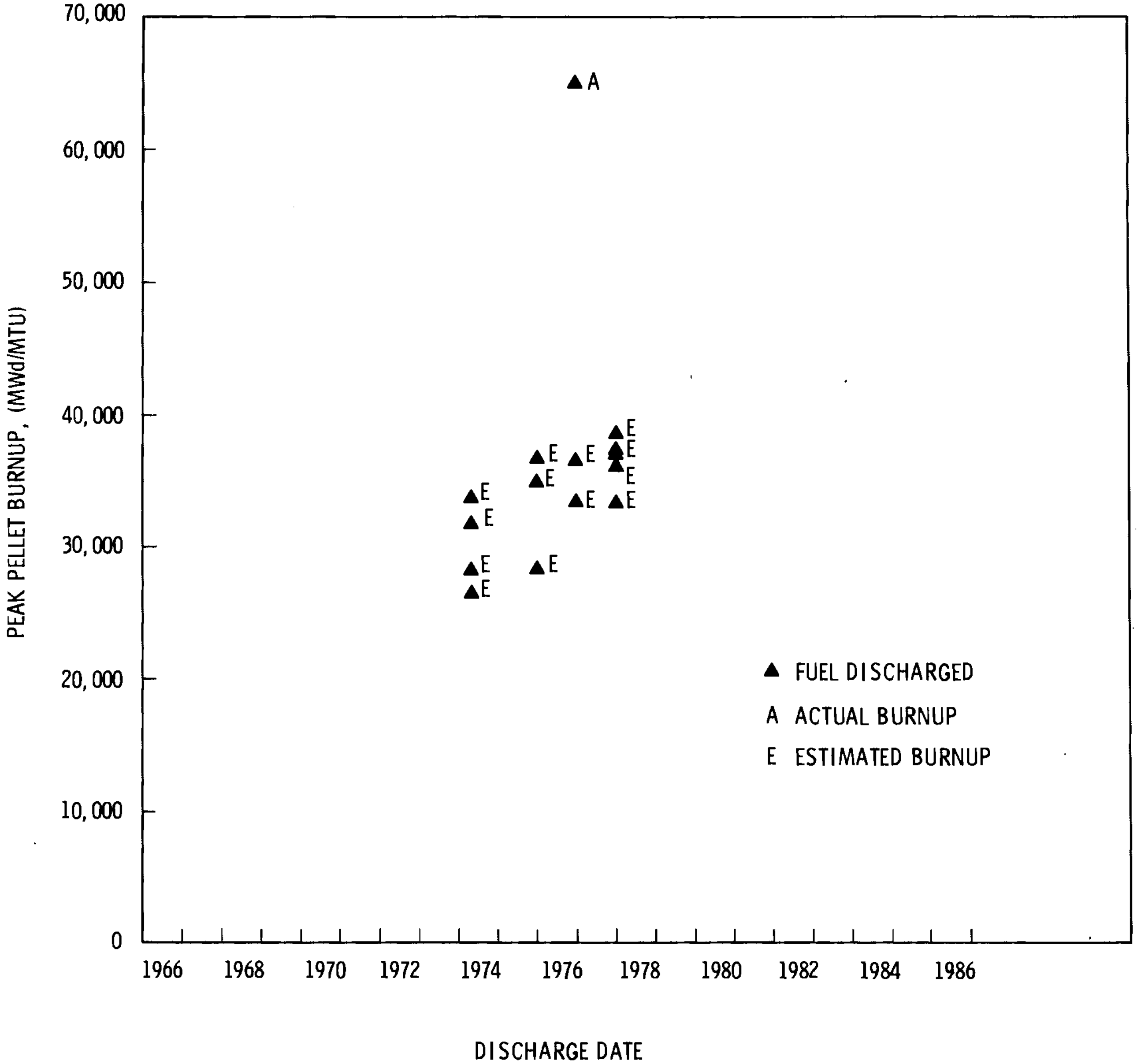

FIGURE A10. Peak Pellet Burnup for Foreign PWR Fuel Assemblies with Zircaloy-Clad Fuel Rods

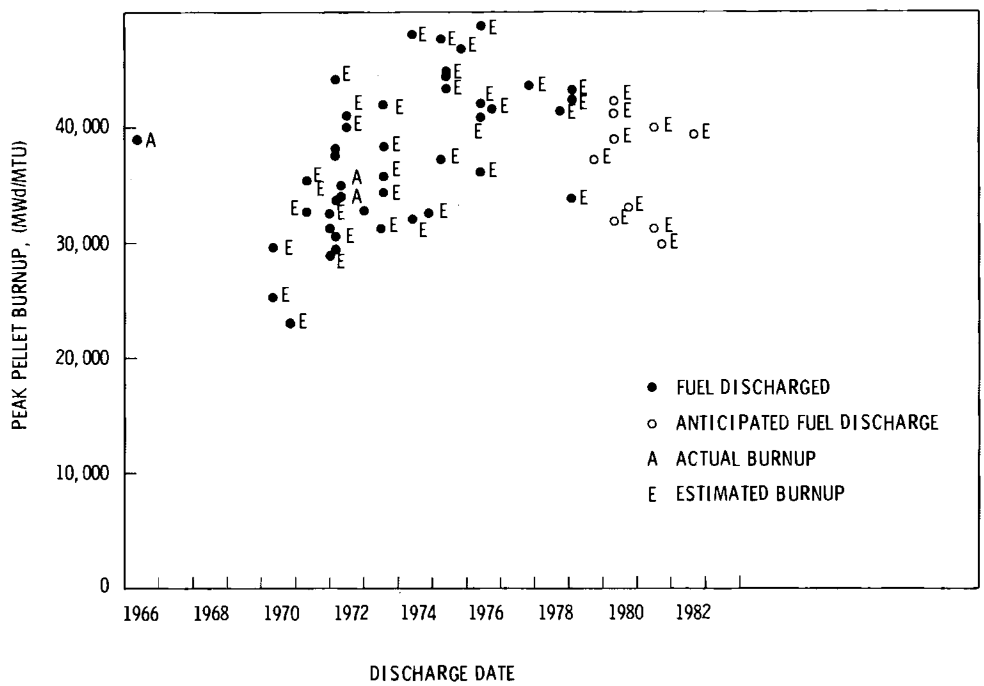

FIGURE A11. Peak Pellet Burnup for U.S. PWR Fuel Assemblies with Stainless-Steel-Clad Fuel Rods

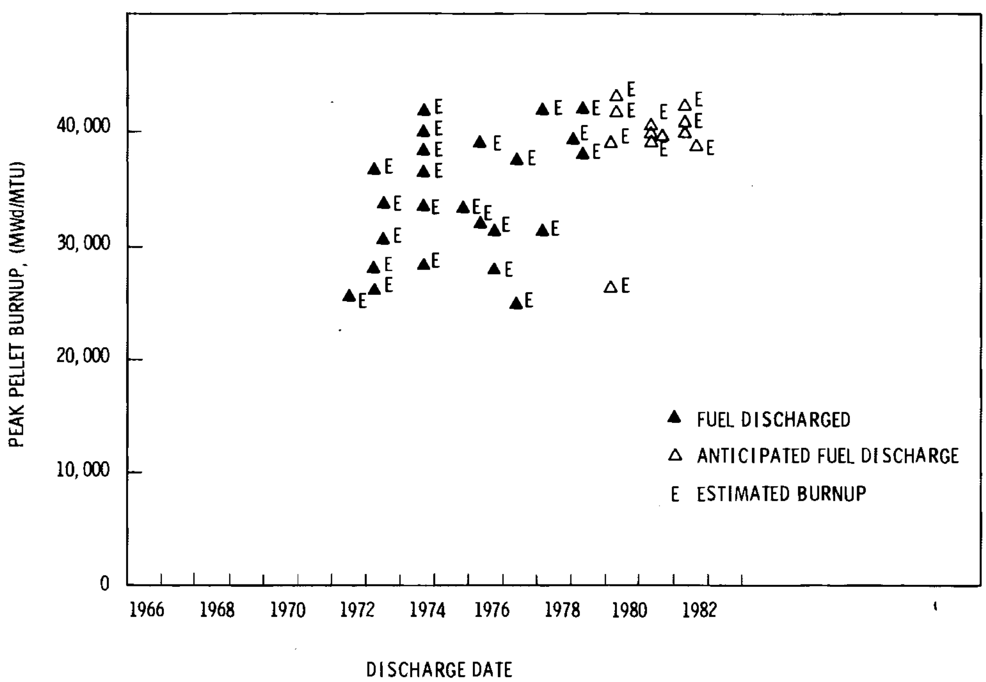

FIGURE A12. Peak Pellet Burnup for Foreign PWR Fuel Assemblies with Stainless-Steel-Clad Fuel Rods

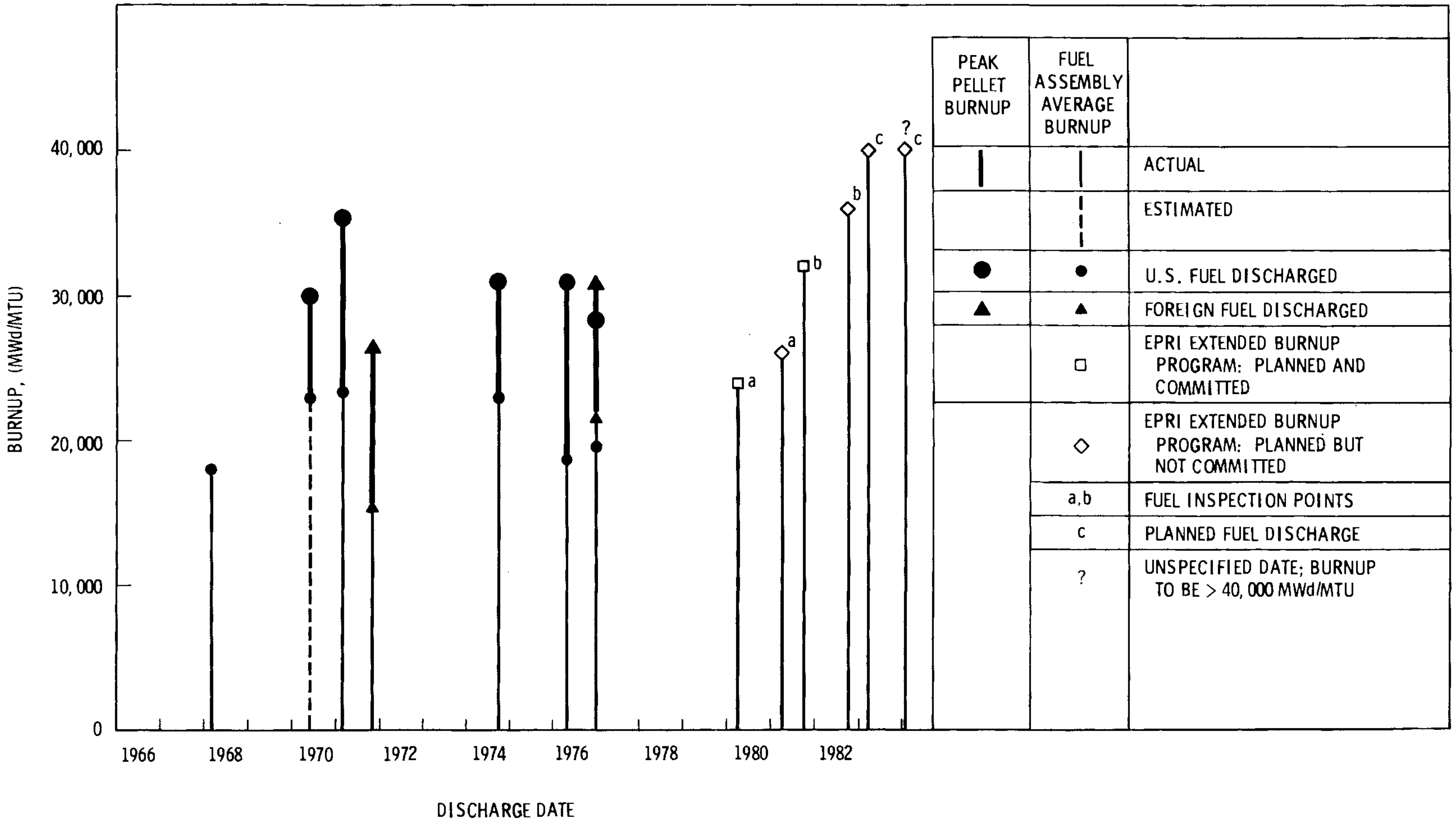

FIGURE A13. Burnup of Discharged BWR Fuel Assemblies with Zircaloy-Clad Fuel Rods

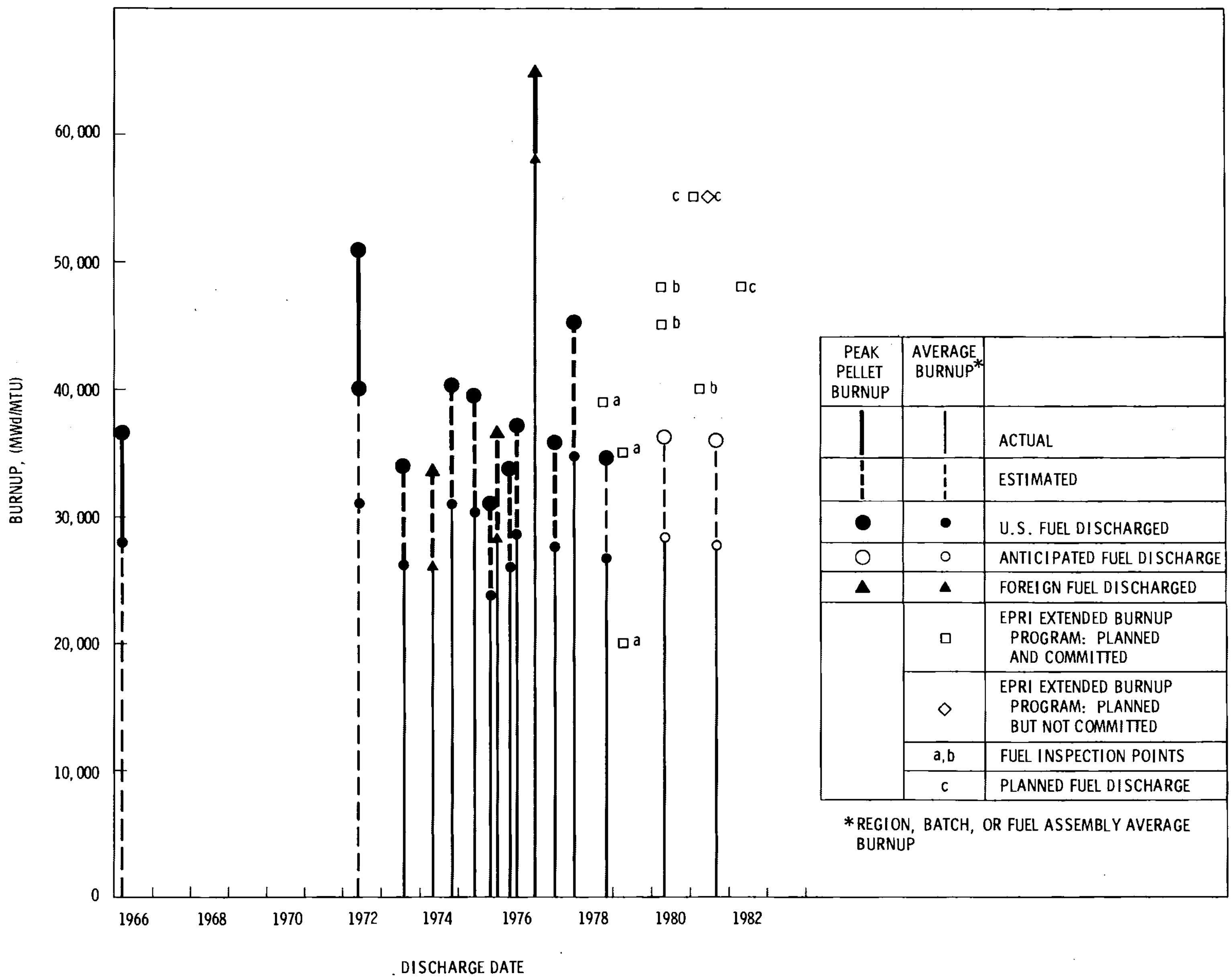

FIGURE A14. Burnup of Discharged PWR Fuel Assemblies with Zircaloy-Clad Fuel Rods

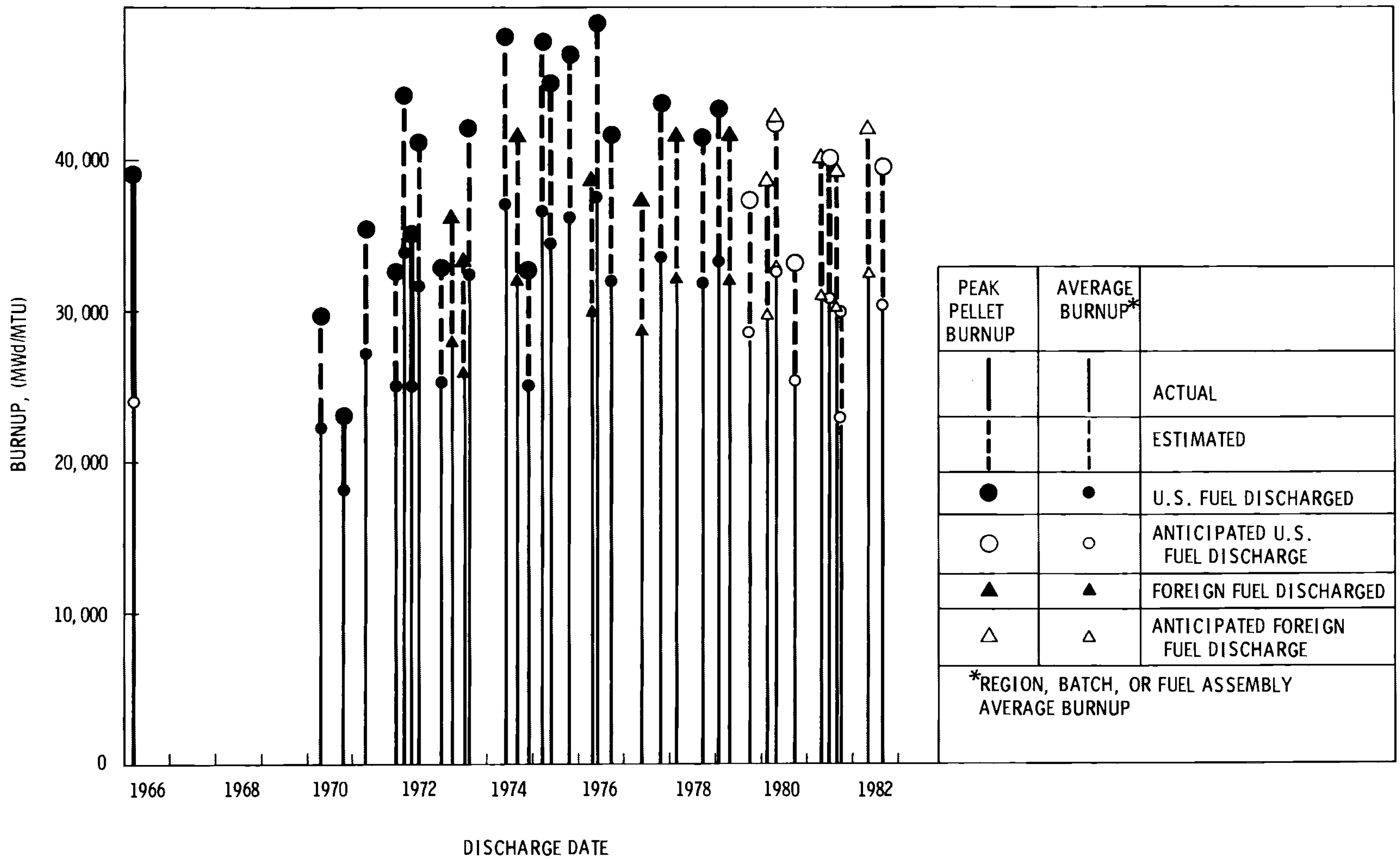

FIGURE A15. Burnup of Discharged PWR Fuel Assemblies with Stainless-Steel-Clad Fuel Rods

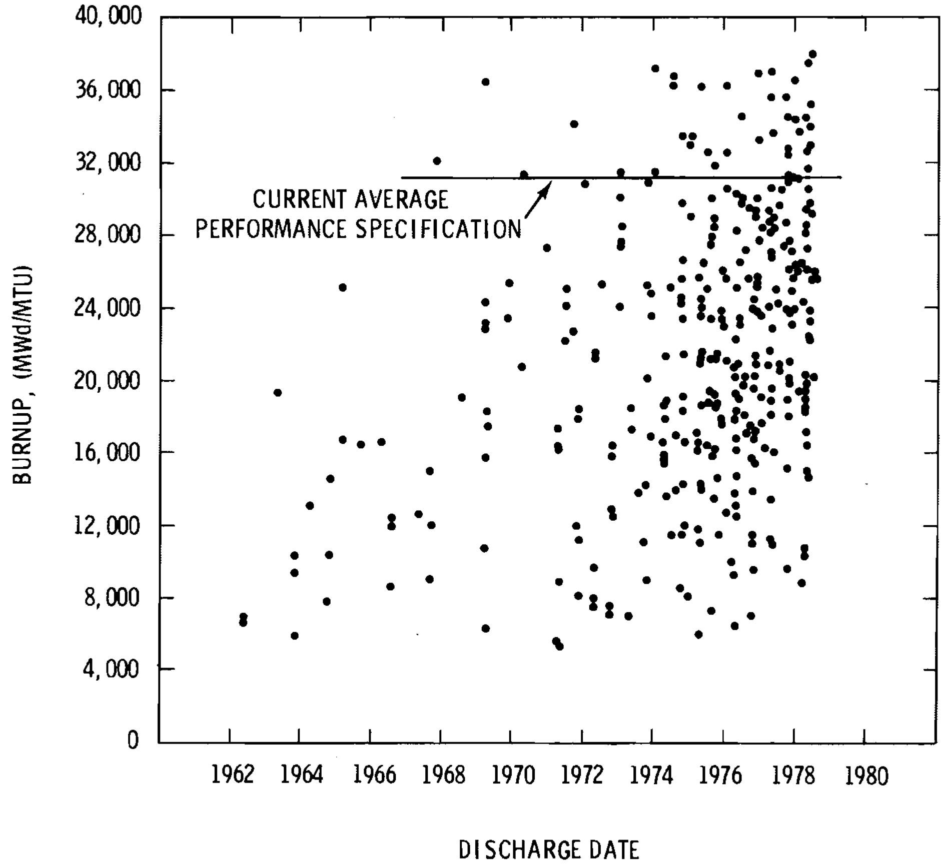

FIGURE A16. Burnup in Fuel Discharged from Operating Nuclear Power Plants (Data points are unscreened, unweighted exposures of discharged fuel, regardless of amount of fuel involved) (Source: Southern Science Applications, Inc.)

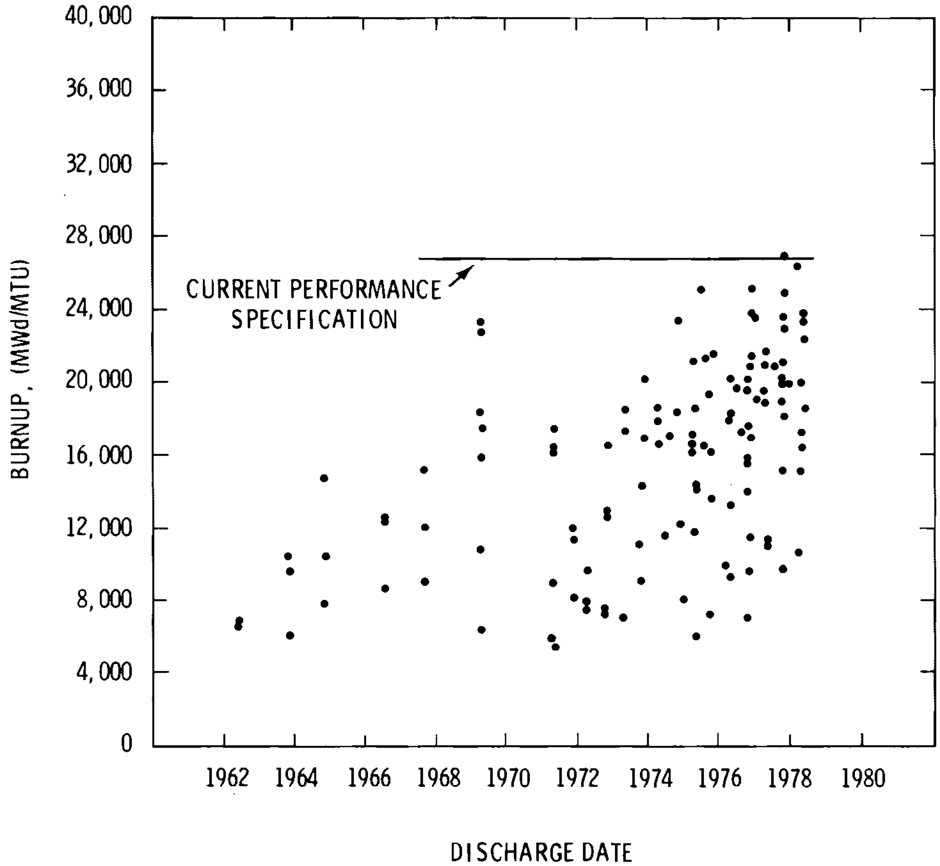

FIGURE A17. Burnup in Fuel Discharged from Operating Boiling Water Reactor Plants. (Data points are unscreened, unweighted exposures of discharged fuel, regardless of amount of fuel involved.) (Source: Southern Science Applications, Inc.)

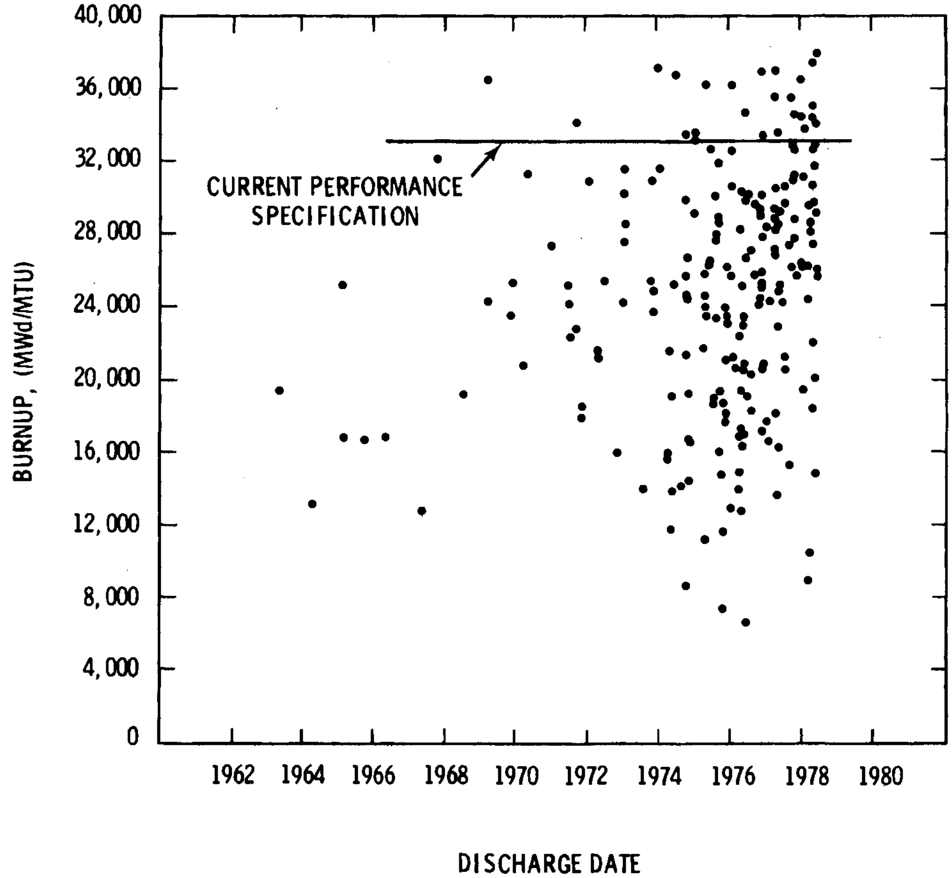

FIGURE A18. Burnup in Fuel Discharged from Operating Pressurized Water Reactor Plants (Data points are unscreened, unweighted exposures of discharged fuel, regardless of amount of fuel involved) (Source: Southern Science Applications, Inc.)

APPENDIX B

CASE HISTORY: CHEMICAL DECONTAMINATION OF A SWEDISH PWR SPENT FUEL POOL

CASE HISTORY: CHEMICAL DECONTAMINATION OF A SWEDISH PWR SPENT FUEL POOL

THE REACTOR POOL DECONTAMINATION AT RINGHALS 2 PWR, APRIL 1978

During a reactor outage in April 1978, repair work was planned in the lower region of the spent fuel pool. However, radiation levels were too high, and a decontamination using chemicals and high-pressure water flushing was carried out. Since the active crud particles consisted of nearly pure magnetite, a one-step treatment with TURCO 4521 (Citrox) was used. The crud particles were deposited as a very thin layer on all of the vertical surfaces. The lower areas where the repairs were planned had a number of hot spots with dose-rates from 2 to 70 rem/hr. This area was filled to a depth of about 60 cm (5 m^3 of solution) with 6% TURCO 4521. Rubber tubes were connected in series with a vessel having electric heaters and a pump.

During the treatment it was impossible to maintain the intended temperature (70^oC). The temperature decreased to 40^oC during the treatment. The circulation lasted for about 6 hr. The activity in the water had then reached a constant level. Flushing with high-pressure water was then started. During flushing, large amounts of crud were trapped in the draining valves making these troublesome to operate. The decontamination was, as a whole, a success; the decontamination factors were from 10-40 and all the hot spots were removed, even though the chemical did not completely dissolve the crud particles. The pool in this kind of plant (PWR) has numerous crevices, corners, and other areas where the crud could accumulate. The waste solution was evaporated, mixed with concrete and stored in steel drums.

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The authors wish to thank Jan Arvesen of Studsvik Energiteknik AB, Nyköping, Sweden, for the information in Appendix B. Thanks are also due the staffs at the Yankee Rowe, Point Beach and Three Mile Island Unit 1 nuclear plants for providing specimens and information. Dr. A. Uriarte of Junta de Energia Nuclear kindly provided information regarding spent fuel storage experience in Spain, and Mr. LeFort provided information regarding a spent fuel shipping incident in France. The cooperation of the Nuclear Assurance Corporation in providing spent fuel inventory information is appreciated. Numerous utilities, fuel vendors and the Electric Power Research Institute have responded to requests for discussions and information.

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