Investigation of the Condition of Spent Fuel Pool Components

September 1981

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Pacific Northwest Laboratory
Richland, Washington 99352

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INVESTIGATION OF THE CONDITION OF
SPENT FUEL POOL COMPONENTS

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ABSTRACT

It is currently projected that spent nuclear fuel, which is discharged from the reactor and then stored in water pools, may remain in those pools for several decades. Other studies have addressed the expected integrity of the spent fuel during extended water storage; this study assesses the integrity of metallic spent fuel pool components. Results from metallurgical examinations of specimens taken from stainless steel and aluminum components exposed in spent fuel pools are presented. Licensee Event Reports (LERs) relating to problems with spent fuel components were assessed and are summarized to define the types of operational problems that have occurred.

The major conclusions of this study are:

- Aluminum and stainless steel spent fuel pool components have a good history of performance in both deionized and borated water pools.
- Although some operational problems involving pool components have occurred, these problems have had minimal impacts. However, these impacts should be assessed by spent fuel pool designers and operators to minimize similar problems in the future.

SUMMARY

There is an emerging recognition that spent fuel may remain in water pools for several decades until a decision regarding its final disposition is ratified and implemented.

In spent fuel pool (SFP) licensing hearings, hearing boards frequently request information regarding the condition of materials in the pools. In addition, numerous new SFPs and replacement components for existing pools are being planned. The investigations described in this report were conducted by Pacific Northwest Laboratory (PNL) to provide information on materials behavior during exposures in SFPs.

The sources of information summarized in this report include:

- SFP materials behavior problems based on operator reports to the U.S. Nuclear Regulatory Commission (NRC)
- observations from surveys and discussions with SFP operators
- results of metallurgical investigations on specimens that were exposed in SFPs.

The principal materials in SFP components are aluminum alloys--for fuel storage racks--and stainless steel (SS)--for storage racks, liners, and piping. There are two types of SFP chemistries:

- pressurized water reactor (PWR) pools having boric acid chemistry (0.02-M H_2BO_3 including 2000-ppm B); the boric acid is used to match the composition of the H_3BO_3 in the reactor primary system during refueling
- boiling water reactor (BWR) and away-from-reactor (AFR) pools using deionized water.

The principal problems mentioned in event reports involved pump seal failures and hose disconnections, minor leaks in SFP pipes and liners, minor fuel assembly damage during handling, air overpressurization damage during attempted cleaning of pool liner channels, piping and weld defects during fabrication, pipe breakage due to freezing during power outages, and corrosion-induced

events in SFP piping (see Appendix A for a more complete listing). Although these problems are minimal, they deserve the attention of storage pool designers, fabricators, and operators.

The scope of PNL examinations of SFP components is summarized in Table 1. Investigation of stainless steel SFP component specimens indicated that uniform corrosion was so small that it was not detectable by metallographic techniques ($<5\text{ }\mu\text{m}$ after up to 15 yr). Mild (25- to $75\text{-}\mu\text{m}$) intergranular corrosion was observed in the heat-affected zones near two welds on one SS component. In one investigation an intergranular crack had propagated through the wall of a 20-cm (8-in.) diameter pipe from a backup (stagnant) SFP heat transport system. The aluminum component examined had an oxide layer of less than $8\text{ }\mu\text{m}$ and some very minor pitting ($<10\text{ }\mu\text{m}$).

A systematic laboratory study is underway at PNL to investigate the effects of various SFP environments on materials behavior. This study will provide additional insights to materials behavior and to approaches to minimize impacts of materials problems on SFP operation. Discussions with pool operators also reflected that pool operations are generally mundane and free of serious problems.

TABLE 1. Summary of Spent Fuel Component Examination

Pool	Reactor Type	Years in Pool	Material	Number of Welds Visually Examined at PNL	Observation		
					Visual	Metallographic	Other
Yankee Rowe	PWR	1.4	30 SS	8	Heat tinting and spatter near welds suggesting high welding temperature; structurally sound; no pits or cracks.	4 welds examined; mild intergranular attack in 2 of 4 heat-affected zones. Maximum attack ~75 μm .	Elemental analysis; C content 0.07%.
Point Beach	PWR	6.7	304 SS	4	Untarnished finish; some heat tinting near welds; structurally sound with no pits or cracks.	4 welds examined; no evidence of intergranular attack or degradation.	Elemental analysis; C content <0.05%.
Zion	PWR	4.8	304 SS	17	Structurally sound; no pits or cracks; untarnished finish with some heat tinting near welds.	7 welds and 1 piece examined; a small amount of crevice pit corrosion (<15 μm); no other significant attack.	1 weld examined by Auger, SEM, and EDAX; weldment penetration along grain boundaries with no degradation; elemental analysis; C content <0.06%.
LACBWR	BWR	11.5 3.5	304 SS Boral	36	Untarnished finish; heat tinting near welds; structurally sound with no pits or cracks.	4 welds examined; no evidence of intergranular attack or degradation. Boral showed 6- μm oxide layer.	Elemental analysis; C content 0.05%.
RBOF	AFR	15 16	304 SS 6061 Al	14	Untarnished bright finish; some heat tinting near welds; structurally sound. Aluminum showed dull tarnished finish.	2 welds and 1 piece examined; no significant intergranular attack or corrosion. Aluminum components had an oxide layer of <8 μm and minor pitting (<10 μm).	

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INTRODUCTION

Spent fuel pools (SFPs) have provided an important link in nuclear fuel management since the first nuclear reactors began to discharge fuel 35 yr ago. Since then, the technology for spent fuel handling and storage has undergone some advances, such as:

- installation of stainless steel (SS) liners for the reinforced concrete structure
- use of ion exchange, filtration, skimmers, and vacuum cleaners to maintain water purity and to control levels of radioactive species in the water
- use of heat exchangers to control pool water temperatures
- use of neutron absorbers such as boron to permit closer packing of fuel assemblies
- use of advanced devices to improve fuel handling.

Compared to many technologies, there have been relatively few materials problems in SFP technology. Temperatures are low; water chemistries are relatively benign; and there is minimal need for equipment with moving parts. However, any technology is subject to some types of materials problems. A cross section of cases involving materials or equipment malfunctions in modern SFPs is summarized in Appendix A and is based on event reports to the U.S. Nuclear Regulatory Commission (NRC).

Currently, interim storage of spent fuel in water pools for several decades appears likely. Licensing boards for SFP modifications frequently request information regarding materials behavior. In response to these needs Pacific Northwest Laboratory (PNL),^(a) under a U.S. Department of Energy (DOE) contract, developed an assessment of SFP component behavior from the following sources:

(a) Operated for DOE by Battelle Memorial Institute.

- a prior survey of SFP experience⁽¹⁾
- NRC incident reports (Appendix A) summarizing a cross section of SFP component malfunctions
- metallurgical examination of specimens from components retired from SFPs.

This assessment provides a current perspective of materials behavior in SFPs. The Spent Fuel and Fuel Pool Component Integrity Program at PNL will continue to focus principally on fuel cladding integrity but will also include evaluation of fuel pool component materials.⁽²⁾

The remainder of this report consists of a discussion of SFP materials and environments and the results of examinations done on SFP components from Yankee Rowe, Point Beach, Zion, La Crosse, and the Receiving Basin for Offsite Fuel (RBOF). The examination of a section of a Three Mile Island-Unit 1 pipe with a through-wall defect is also summarized.

SPENT FUEL POOL MATERIALS AND ENVIRONMENTS

SPENT FUEL POOL MATERIALS

It is evident from Table 2 that stainless steel is the material most frequently used for SFP components such as liners for the concrete pool, storage racks, canisters, piping, and heat exchangers. The longest continuous exposure of SS fuel pool components to water occurred at the X-10 reactor pool (Oak Ridge National Laboratory), which first stored fuel in November 1943 and is still storing radioactive materials (1981). The longest exposure of SS spent fuel pool components to borated water appears to have occurred at the Indian Point 1 reactor, which began operation in 1962 and is continuing to store fuel in 1981.

Aluminum has been used as a rack and canister material principally in deionized water pools; however, aluminum racks have also been installed in a few pressurized water reactor (PWR) pools (see Table 3).

NORMAL SPENT FUEL POOL ENVIRONMENTS

The two principal SFP environments are summarized in Table 4: boiling water reactor (BWR) or away-from-reactor (AFR) pools, which use deionized water; and PWR pools, which use boric acid chemistries.

The PWR primary coolant includes boric acid as a chemical shim for fuel reactivity control. Most PWR pools use equal boric acid concentrations in the primary coolant and in SFP waters. The boric acid provides a measure of reactivity control in the pool; however, no credit is taken for this aspect in the design of PWR pool racks. The principal justification for use of boric acid in the SFP is to provide compatibility with the primary coolant during refueling.

Some PWRs do not use the full boric acid concentration as indicated below.

- The Shippingport (Pennsylvania) reactor is listed as a PWR although it is not of standard PWR design. The primary system chemistry uses ammonia, and the SFP chemistry is deionized water.
- The Yankee Rowe (Massachusetts) reactor is an early-vintage (1961) PWR. The primary system utilizes boric acid as a chemical shim for

TABLE 2. Summary of Materials in Spent Fuel Pools(a)

Component	Subcomponents	Material(b)	Alloy
Wall	--	Reinforced Concrete	--
Pool Liner	--	Stainless Steel Epoxy, Fiberglass	304
Heat Exchanger(c)	--	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
Demineralized 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
Fuel Storage Racks(d)	Racks	Stainless Steel Aluminum	304L 6061-T6
	Embedded Supports Neutron Absorber	Stainless Steel Aluminum-B ₄ C Composite	304L Boral
Fuel Storage Canisters(d)	--	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 Off-Gas Hoods	--	Stainless Steel Aluminum	--
Cask Handling Crane	Cable and Grapple	Stainless Steel	--
Canister Crane	Cable and Grapple	Stainless Steel	--
Fuel Transfer Conveyor	--	Stainless Steel	304L
Insulators	--	Tygon, Neoprene	
Cask Impact Pad	Cladding Pad (not exposed to H ₂ O)	Stainless Steel Carbon Steel (3-1/2 in. thick)	
	Honeycomb (not exposed to H ₂ O)	Aluminum	

(a) Reference 1.

(b) Types identified in survey; other types may be used in some pools.

(c) 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.

(d) Some canister and rack walls contain boron-impregnated aluminum for reactivity control; the boron-containing materials are clad with stainless steel or aluminum.

TABLE 3. Examples of Aluminum Alloy Fuel Racks in Spent Fuel Pools

Spent Fuel Pool	Location	Pool Chemistry	Time in Pool	Remarks
Studsvik	Studsvik, Sweden	Deionized Water	1964 to present	No problems.
G. E.-Morris	Morris, IL, USA	Deionized Water	1972 to 1976	Corrosion at poor welds.(a)
WAK	Karlsruhe, GFR	Deionized Water	1969 to present(b)	No corrosion after anodizing.
Yankee Rowe	Rowe, MA, USA	Boric Acid(c)	17 yr; maximum	Small amount of pitting; good structural integrity.
Three Mile Island- Unit 1	Goldsboro, PA, USA	Boric Acid	1974 to present	No problems; insulated from SS liner.
Maine Yankee	Wiscasset, ME, USA	Boric Acid	1977 to present	No problems.
Oyster Creek	Toms River, NJ, USA	Deionized Water	7 yr(d)	No problems.
NFS	West Valley, NY, USA	Deionized Water	1965 to present	No problems.
NRX	Chalk River, Ontario, Canada	Deionized Water	1959 to present	No problems.
RBOF	Savannah River, SC, USA	Deionized Water	1963 to present	No problems.
FRSF(e)	Idaho Falls, ID, USA	(f)	1963 to present	Pitting corrosion.

(a) Most welds were uncorroded; only poorly made welds corroded.

(b) In 1973 one-half of the aluminum racks were replaced with stainless steel ones.

(c) 800-ppm B maximum.

(d) Examination of locations on aluminum racks where they contacted SS pool liner revealed no significant crevice corrosion.

(e) Operated by Idaho National Engineering Laboratory (INEL).

(f) Up to 730-ppm Cl^- and 590-ppm NO_3^- ; currently Cl^- = 360 ppm and NO_3^- = 430 ppm.

TABLE 4. Spent Fuel Pool Water Quality Specifications(a)

Parameter	PWR		BWR/AFR
	Normal	Transient Limits	
pH (77°C)	4.5 to 6.0(b)	--	5.3 to 7.5
Conductivity at 25°C, μmho/cm	1 to 30	--	3(c)
Chloride Ion, ppm (max)	0.15	1.5	0.5
Fluoride Ion, ppm (max)	0.1	1.5	--
Oxygen	Saturated	Saturated	--
Total Suspended Solids, ppm (max)	1.0	--	--
Temperature, °C	50	100	--
Boric Acid, ppm B(d)	0 to 2000 1950 to 2250	-- --	-- --
Activity Level, μCi/ml	0.05 varies with tech spec.		
Solid Filtration, microns	1 to 5		

(a) May vary somewhat from pool to pool.

(b) Some fuel pool pH specifications range from 4.5 to 8.0; a few pools range from 7.0 to 10.6.

(c) 1.5 at some pools.

(d) Some PWR fuel pool specifications also indicate a lithium concentration range from 0.2 to 2.0 ppm, corresponding to values for PWR primary systems.

fine adjustments to fuel reactivity. The SFP started with a deionized water chemistry, and boric acid is added to the pool only when the borated primary system water is mixed with the pool water. Thus, boric acid concentrations in the Yankee Rowe SFP have been lower than in other PWR spent fuel pools, where boron concentrations are typically controlled at >2000-ppm B by intentional additions of boric acid. By January 1979 the boric acid concentration in the Yankee Rowe pool had risen to 800-ppm B compared to typical concentrations of 2000-ppm B in other PWR pools.

UNUSUAL SPENT FUEL POOL CHEMISTRIES

A few SFPs have operated with chemistries that depart from the norms indicated in the preceding section. The most notable example is the Fuel Receiving and Storage Facility (FRSF) pool at the Idaho National Engineering Laboratory (INEL), Idaho Falls, Idaho.

Due to problems with algae growth, the pool received heavy chlorine dosing, which resulted in the impurity levels indicated in Table 5. Improvements in controlling biological species at the pool included installation of a side stream ultraviolet light treatment; and in 1976 a scraped surface evaporator and a reverse osmosis system were added to the pool, resulting in the water chemistry improvements shown in Table 5. The high chloride levels in the pool resulted in obvious corrosion to aluminum pool components. However, SS components and SS- and Zircaloy-clad fuel resided in the pool for several years with no obvious problems.

The question sometimes arises: What would happen to the pool chemistry if a pool were left for an extended period without purification? An opportunity arose to make a relevant observation at the Test Area North (TAN) facility at INEL. The SFP was deactivated in approximately 1967, and until that time water purity had been maintained by ion exchange and filtration. The pool is housed inside a cinder block building and during windstorms airborne dust enters the building; therefore, the pool environment is not totally isolated from the desert terrain. The pool water was sampled in 1979 (see

TABLE 5. Water Composition in Fuel Receiving and Storage Facility Spent Fuel Pool at INEL(a)

<u>Component</u>	<u>Content Before CRS(b) Operation, μg/ml</u>	<u>Content After CRS Operation, μg/ml</u>
Cl	730	360
Na	715	415
NO ₃	590	430
Ca	18	4.2
SO ₄	53	30
Si	3.4	3.5
Mg	2.6	1.4
HCO ₃	59	52
TDS(c)	2250	1320

(a) McClure, L. W. 1977. "Chloride Removal from Storage Basin Waters." American Nuclear Society Trans. 27:487.

(b) CRS = chloride removal system.

(c) TDS = total dissolved solids.

Table 6). Perhaps most significant is the chloride content of the pool water, which had risen to only 1.8 ppm after 12 yr with no purification.

SFP CONFIGURATIONS AND COMPONENTS

Figure 1 shows a cross section of a SFP, indicating the schematic configuration and the major components. Figure 2 shows two views of a spent fuel storage rack as it was removed from a PWR pool after 6.7 yr of storage.

TABLE 6. Analysis of TAN Fuel Storage Canal Water^(a)
After 12 yr with No Purification

<u>Chemical Species</u>	<u>Concentration, ppm wt</u>	<u>Radioactive Species</u>	<u>Concentration, dps/ml</u>
Al ⁻	0.18	¹³⁷ Cs	32.6
B	0.12	¹³⁴ Cs	6.45
Ca	1.8	⁶⁰ Co	5 x 10 ⁻²
Cu	0.021	⁹⁰ Sr	4 x 10 ⁻¹
Mg	0.003		
Na ^(b)	7.4		
Ni	0.0066		
Si ^(c)	4.1		
K ^(b)	5.4		
Hg	<2 x 10 ⁻⁵		
Cl ⁻	1.8		
F ⁻	0.4		

-
- (a) Canal water volume = 2,649,000 l (700,000 gal);
estimated dissolved solids = 50 to 100 kg.
(b) K and Na may be from NaK (from fast reactor fuel
stored in the canal).
(c) Probably from airborne particulates.

FIGURE 1. Schematic of Spent Fuel Storage Pool

Neg. 8008771-1

Neg. 8008771-2

FIGURE 2. Sequential Views of First-Generation Point Beach (PWR) Spent Fuel Storage Rack Being Removed from the Borated Water Pool

SPENT FUEL POOL COMPONENT EXPERIENCE

There have been several publications⁽¹⁻⁶⁾ that have considered the materials behavior of SFP components. Event reports that SFP operators supply to the NRC are another source of information regarding pool materials behavior.

SURVEYS OF SPENT FUEL STORAGE EXPERIENCE

A 1977 report⁽¹⁾ summarized spent fuel storage experience in the United States and Canada. The report centered on spent fuel behavior but included a section on behavior of SFP equipment. A Fiscal Year 1979 Annual Report⁽²⁾ also included aspects of SFP equipment behavior; most of the information from that source is included in this report.

The Nuclear Energy Agency (a component of the Organization for Economic Cooperation and Development, Paris) and the International Atomic Energy Agency (IAEA), Vienna, are cooperating in a survey of world spent fuel storage experience. The survey includes requests for indications of materials behavior in SFPs, and survey results are expected to be issued in late 1981.

From the 1977 survey and more recent discussions with SFP operators it is evident that materials problems have had a minor impact on SFP operation. Where significant materials problems have occurred, pool operators have been able to deal with them on a timely and effective basis.

INSIGHTS TO SPENT FUEL POOL COMPONENT BEHAVIOR FROM EVENT REPORTS

Malfunctions of SFP components are reported to the NRC; 44 event reports involving SFP equipment are summarized in Appendix A to provide a cross section of the types of malfunctions that have occurred. The reports do not represent all events reported but may provide fuel pool designers and operators with insights that would preclude similar events. A review of the consequences indicates that there were cases where some radioactive water leaked from damaged SFP components; however, the leakage had minimal impacts because of relatively low concentrations of radioactive species (generally 10^{-2} to 10^{-5} $\mu\text{Ci/ml}$). None of the event reports indicated that the component failures resulted in human injury.

BEHAVIOR OF ALUMINUM SPENT FUEL POOL COMPONENTS

Aluminum alloys are used for construction of spent fuel storage racks and canisters; Table 3 summarized some of the more significant uses of aluminum alloys in SFPs. Aluminum components have been used in deionized water for up to 25 yr at the NRX reactor pool at Chalk River, Ontario, Canada (not shown in Table 3). Although aluminum racks are seldom used in boric acid (PWR) pools, they have been used in the boric acid pools at Maine Yankee (Wiscasset, Maine) and Three Mile Island-Unit 1 (Goldsboro, Pennsylvania). An aluminum rack also resided in the Yankee Rowe pool for up to 17 yr at boric acid concentrations up to 800-ppm B as boric acid.

Due to the need for an increased storage capacity in most power reactor SFPs, neutron-absorbing material such as boral is being extensively used to fabricate high-density storage racks. Boral consists of an aluminum/boron carbide cermet encased in aluminum; this material has been used in the Brookhaven Medical Research Reactor fuel storage area since January 1959⁽³⁾ as well as in Yankee Rowe, Maine Yankee, La Crosse, and other nuclear power plants. Water leakage into boral that is sealed in stainless steel has led to swelling due to hydrogen production of the incasement. Radiolytic decomposition of organic binder compounds in B_4C compacts appears to have been another source of gaseous hydrogen in spent fuel storage racks. Currently, racks are designed with vent holes at the tops of the rack panels to keep hydrogen pressures from developing.

EXAMINATION OF A YANKEE ROWE (PWR) STAINLESS STEEL SPENT FUEL POOL RACK SUPPORT SECTION

A section of a SFP rack support from Yankee Rowe was received at PNL for metallurgical examination on February 22, 1979. It had a total surface activity of $3.74\text{-}\mu\text{Ci } ^{60}\text{Co}$ (January 4, 1979) with smearable contamination of 500 dpm/cm^2 . Figure 3 shows the layout of the SFP area and indicates where the rack section was removed. Table 7 indicates the chemical analysis performed on samples from vertical and horizontal plates of the rack support and the nominal conditions for the pool water. The rack section was exposed in the pool for ~17 months (from December 1975 to May 1977).

The overall size and geometry of the rack support section is shown in Figure 4. It consisted of a piece of 6.35-mm (1/4-in.) 304 SS angle with 7.6-cm (3-in.) flanges attached by threaded rods and weldments to a solid 9.5-cm (3-3/4-in.) square 76.12-cm (28-in.) long billet. The billet was nonmagnetic and had a crusty brown surface scale, which suggested that it was a low-grade, low-chromium austenitic stainless steel (e.g., alloy 501). The SS angle was slightly discolored in spots but generally had a bright finish. It was characterized by a flame-cut edge (from removal operations) and an abbreviated double-T welded joint (approximately 1/2 of a box end joint).

From Figure 4, it appears that a manual-shielded metal arc welding technique was used with a coated electrode as filler material (commonly 308 SS). The middle macrograph shows slag entrapment at the weld-base metal fusion line and local grinding marks. The weld spatter observed in the right macrograph is caused by excessive welding current or a variation in the metal transfer from nonhorizontal welding positions.⁽⁷⁾ A heat-tinting effect is also seen in the photographs, indicating areas of the plates that were exposed to high temperatures during the joining process. The weld beads and fusion line areas appear structurally sound with no visual pits or surface cracks.

YANKEE ROWE MICROSTRUCTURAL ANALYSIS

A section of the 304 SS angle, including two weldments, was prepared for optical microscopy. Microstructural examinations were performed at the points

THIS SECTION OF FLOOR
WAS EXPOSED TO WATER
FROM 12/75 TO 5/77

1/4" = 1'

(3) PCS 1 3/4 CYPRESS +
1/16 STAINLESS X 8' X 16'

FIGURE 3. Schematic of Top View of Yankee Rowe Spent Fuel Pool

TABLE 7. Rack Support Specimen Compositions and Pool Conditions(a) for
Yankee Rowe Spent Fuel 304 SS Rack Support

Sample Location	Material	Rack Support Specimen Composition, wt%													
		C ^(c)	Mn	P	S	Si	Ni	Cr	Mo	V	Cu	Sn	Ti	Al	Co
Horizontal Plate	304 SS	0.07	1.63	0.028	0.016	0.48	9.69	18.18	0.41	0.033	0.28	0.14	0.008	0.007	0.130
Vertical Plate	304 SS	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

(a) Pool Conditions

pH: 6.8
 Boron: 800 ppm (maximum)
 Cl: <0.15 ppm
 Conductivity: 50 ohms-l
 Activity: 3.5×10^{-3} μ Ci/ml (nominal)
 Temperature: 24 to 35°C (75 to 95°F)
 (seasonal variation)
 Residence Time in Pool: 17 months

(b) Analysis performed by Jorgensen Steel Co., Seattle, Washington.

(c) Performed by LECO method.

FIGURE 4. Metallographic Examination of a Yankee Rowe (PWR) Spent Fuel Pool Rack Support Section (304 SS). Exposed to 800-ppm B as boric acid for 17 months at 24 to 35 °C

shown on the lower micrograph in Figure 4 and are shown in Figures 5, 6, and 7. The major points of interest are:

- a grain boundary penetration zone at points A and D on the vertical plate; other surface locations (points B, E, F, L) lacked this zone
- a plate microstructure characterized by equiaxed grains (~ 0.033 -mm diameter ASTM #7) with elongated inclusion stringers depicting rolling direction and by transgranular twins
- a nonuniform distribution of grain boundary precipitates as shown in Figure 8
- a dendritically solidified weld metal (austenite with delta ferrite constituents) free of visual defects with a small precipitate-free zone extending inward from the fusion line
- an entrapment of slag 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 5 reveals the following:

- a zone of shallow intergranular attack ~ 1 mm in length that initiates at the fusion line and extends ~ 25 to $80\text{ }\mu\text{m}$ into the base metal
- a decreasing depth of grain boundary penetration and surface grain size with increasing distance away from the fusion line
- an increasing density of grain boundary precipitates in surface regions with increasing distance from the fusion line
- removal of surface grains in areas adjacent to the fusion line
- grain boundary widening, with grain boundary precipitates residing along the affected boundaries.

FIGURE 5. Metallographic Examination of Yankee Rowe Vertical Plate and Weldments (304 SS). Exposed to 800-ppm B as boric acid for 17 months at 24 to 35°C (see Figure 4).

FIGURE 6. Metallographic Examination of Yankee Rowe Left Weldment (304 SS). Exposed to 800-ppm B as boric acid for 17 months at 24 to 35°C (see Figure 4).

FIGURE 7. Metallographic Examination of Yankee Rowe Right Weldment and Horizontal Plate (304 SS). Exposed to 800-ppm B as boric acid for 17 months at 24 to 35°C (see Figure 4).

FIGURE 8. Schematic of Yankee Rowe Weldments Showing Microstructural Zones (see Figure 4)

YANKEE ROWE WELDING THERMAL HISTORY

A numerical analysis was performed to characterize the thermal history of the plates with respect to welding parameters and design considerations. Table 8 and Figures 9 and 10 show peak temperatures, cooling rates, and heat-affected zone results. It is important to note that these values represent calculations for an initial weld pass on one side of the plate only. A second pass on the opposite side of the first weld bead could lower the cooling rates and peak temperatures significantly, depending on the interpass temperature between welding operations. Values of the above variables for three-leg heat transfer are shown, assuming that the opposite flange of the SS angle contributes to the heat removal process. These values would apply for a single pass near the 90° bend area of the SS angle.

A summary of heat transfer equations, thermal properties, design considerations, and assumptions for the thermal analysis is included in Appendix C.

DISCUSSION OF YANKEE ROWE SPENT FUEL POOL RACK SUPPORT SECTION

The welding of austenitic stainless steels with relatively low thermal conductivity can introduce localized areas of high heat input to the base

TABLE 8. Temperature History Analysis(a)

Distance from Weld Fusion Line, mm (in.)	Three-Leg Heat Transfer	
	Peak Temperature, °C	Cooling Rate, °C/s
0	Melting Point	
0.5 (0.002)	1323	94.1
1.0 (0.04)	1247	83.3
1.5 (0.06)	1179	73.9
3.0 (0.12)	1015	53.1
6.4 (0.25) ^(b)	773	26.2
8.9 (0.35)	659	19.0
11.9 (0.47)	560	10.3
15.9 (0.63)	468	5.5
20.0 (0.79)	402	3.5

Sequence: Select distance from weld fusion line, calculate peak temperature, and calculate cooling rate using equations in Appendix C.

(a) References 8, 9, and 10.

(b) Thickness of plate.

metal.⁽⁸⁾ The 304 SS is microstructurally altered when chromium diffuses to grain boundaries and reacts with carbon to form stable chromium carbides (see Figure 11). This process is called sensitization. The susceptibility of austenitic stainless steels to sensitization at moderately high (550 to 800°C) and relatively low (~300°C) temperatures is well known.⁽¹¹⁾ Sensitized, and to a much lesser extent, unsensitized austenitic materials are prone to intergranular corrosion and stress corrosion cracking (SCC) under certain environmental conditions.⁽¹²⁻¹⁵⁾ Several mechanisms have been proposed to account for this effect,⁽¹⁶⁾ including chromium-depleted grain boundary attack, grain boundary residual strain energy, precipitate-base metal electrochemical effects, and impurity segregation theories. Of these, the first and last theories appear to have the most support.

FIGURE 9. Peak Temperature Versus Distance from Weld Fusion Line

The chromium depletion theory accounts for reduced corrosion resistance if the chromium concentration falls to less than 13 weight percent (wt%) in the vicinity of the grain boundaries. This depletion may result in intergranular attack in the chromium-depleted metal adjacent to the grain boundaries. Prolonged heating or slow cooling in the 550 to 800°C temperature range allows for carbide formation. Immunity of high-purity alloys to corrosion restricts the strain energy theory since all alloys contain strained grain boundaries to a certain extent.

It has been suggested that an impurity segregation model involving sulfur, phosphorus, and nitrogen deposited from nonsensitizing heat treatment (1050°C for 2 h) at grain boundaries can lead to enhanced intergranular attack.⁽¹⁷⁾ Sulfide inclusions in stainless steels have been observed to dissolve selectively because of thermodynamic instability within the passive potential range for stainless steels.⁽¹⁸⁾ The dissolution of these Mn-Cr-Fe sulfides took place at the boundary between the sulfide and metal matrix. Sulfide

FIGURE 10. Peak Temperature Versus Distance from Fusion Line

HEAT-AFFECTED ZONE	SOLUTION ANNEAL ZONE	HIGH-TEMPERATURE ZONE	WELD METAL	FUSION ZONE
STRUCTURAL CHANGES: CHROMIUM CARBIDES PRECIPITATE CHROMIUM NITRIDES PRECIPITATE POSSIBLE PROBLEMS: INTERGRANULAR CORROSION IGSCC	STRUCTURAL CHANGES: CHROMIUM CARBIDES DISSOLVE CHROMIUM NITRIDES DISSOLVE NIOBIUM CARBIDES PRECIPITATE TITANIUM CARBIDES PRECIPITATE	STRUCTURAL CHANGES: CHROMIUM CARBIDES DISSOLVE NIOBIUM CARBIDES DISSOLVE TITANIUM CARBIDES DISSOLVE POSSIBLE PROBLEMS: KNIFE LINE ATTACK		STRUCTURAL CHANGES: GRAIN BOUNDARY LIQUATION POSSIBLE PROBLEMS: HOT CRACKING
550 TO 800°C 1000 TO 1450°F	800 TO 1250°C 1450 TO 2250°F	1250 TO 1350°C 2250 TO 2450°F		

FIGURE 11. Possible Mechanisms for Intergranular Attack on Welded Austenitic Stainless Steels

dissolution freed Fe and Cr ions that can undergo hydrolysis, contributing to a higher concentration of hydrogen ions or a lower pH. This higher acidity can restrict the repassivation of the matrix and allow for selective base metal dissolution.

In certain environments, small concentrations of chloride and fluoride ions are known to affect sensitized austenitic stainless steels adversely under stressed and unstressed conditions.⁽¹⁴⁻²⁰⁾ Fluoride ion attack tends to be intergranular, and decreased susceptibility has been observed for fine-grained and low-carbon content materials.⁽¹⁹⁾ A potential source of fluoride contamination is the slag residue from compounds considered necessary for weld metal atmospheric protection. However, stress corrosion of welded austenitic stainless steels in SFP environments is a rare phenomenon.

The welded joint from the Yankee Rowe SFP support rack shows an unexpected selectivity for areas of intergranular attack—namely, on only two of the four heat-affected weld zones. This observation suggests either a slightly different chemical composition (e.g., carbon, sulfur, phosphorous, or nitrogen)

or a welding procedure that could adversely affect one plate. A chemical analysis was performed on samples from the vertical and horizontal plates of the pool racks; the compositions were almost identical with only a minor variation in nickel, which would not account for impurity variations. Therefore, it appears that the thermal history from the welding operation adversely affected one plate, allowing for preferential impurity and/or carbide segregation to the attacked grain boundaries of the attacked regions (see Figure 5).

The thermal analysis described previously has shown a peak temperature of $\sim 1350^{\circ}\text{C}$ close to the weld fusion line, decreasing to $\sim 600^{\circ}\text{C}$ ~ 15 mm (0.6 in.) from the fusion line. Figure 10 suggests that the cooling rate is initially very high for areas adjacent to the fusion line, indicating a short time at these elevated temperatures. A zone of intermediate temperatures and cooling rates occurs at a distance of ~ 4 to 15 mm from the fusion line, suggesting a cumulative time/temperature history (Figure 12) conducive to carbide precipitation.

FIGURE 12. Intergranular Corrosion of 304 SS Tested in Acid Copper Sulfate Solution. Shaded areas indicate treatments resulting in intergranular corrosion. The effect of carbon content in the steels is shown as are common sensitizing and nonsensitizing thermal treatments (see Reference 21).

A possible sequential process correlating the thermal analysis of the material with microstructural observations immediately after passage of the heat source is shown below.

- 1) At areas close to the fusion line, high temperatures and cooling rates dominate. The cooling from high temperatures corresponds to a progressively decreasing high-temperature anneal accompanied by carbide dissolution (see Figure 11) and impurity (e.g., sulfur) segregation to grain boundaries; this area corresponds to the larger grained material adjacent to the weld bead (areas A and D in Figure 5) and is characterized by a local intergranular attack with low density of grain boundary precipitates. This essentially carbide-free zone extends around each weldment for a finite distance from the fusion line.
- 2) Lower temperatures and cooling rates characterize the area adjacent to the area described above. Under these thermal conditions, certain precipitates form (depending on time-temperature history) and act as effective obstacles for grain growth. The resultant area is very fine-grained with a slight intergranular attack.
- 3) Far removed from the above area is a region of relatively unaffected material (area B in Figure 5) with no noticeable precipitates. The grain size in this area is comparable to that of the unaffected base metal.

Since the zones of intergranular attack at points A and D on the vertical plate were within an area of relatively clean grain boundaries and the thermal analysis indicated very high peak temperatures and cooling rates in these regions, a nonsensitized microstructure is indicated. Recalling that fluoride-induced intergranular attack depends on a sensitized microstructure,⁽¹⁹⁾ this effect does not appear to be responsible for the observed intergranular attack.

It has previously been shown that grinding operations on the inside surfaces of 304 SS pipe can introduce concentrated areas of surface cold work and a rough surface topography⁽²²⁾ (similar to that observed in areas A, B, D, E,

F, and L in Figures 4 through 7). These effects allow for a strain-induced martensite transformation that displays poor corrosion and oxidation resistance. The combined effect of a cold-worked surface with reduced corrosion resistance and segregated impurities on grain boundaries and surface intersections could serve as potential sites for corrosion initiation.

SUMMARY AND CONCLUSIONS - YANKEE ROWE

The Yankee Rowe SFP environment consisted of a slightly acidic borated water (800-ppm B) environment at 24 to 35°C. Areas of intergranular attack in two of four heat-affected weld zones were observed. This attack extended ~1 mm from the fusion line and ~25 to 80 μm into the base metal. No other significant attack was observed. The T-joint weldment was exposed to the above conditions for ~17 months. A corrosion data survey conducted by the National Association of Corrosion Engineers (NACE)⁽²³⁾ listed a corrosion rate of <25 $\mu\text{m}/\text{yr}$ for 304 SS in a very dilute boric acid-water environment; the maximum depth of attack observed was ~75 μm in a very localized area. The local intergranular attack can be attributed to a combination of high heat input welding and austenitic stainless steel with a relatively high carbon content. Of the 20 SS welds inspected by metallography (Table 1), two of the four Yankee Rowe welds were the only ones where shallow intergranular corrosion occurred.

EXAMINATION OF A POINT BEACH (PWR) SPENT FUEL STORAGE RACK SECTION

A 304 SS section of a SFP rack from the Point Beach Nuclear Plant, Two Creeks, Wisconsin, was received at PNL for examination on May 30, 1979. The rack section and a SS nut and square washer used to anchor the racks to the floor were also supplied.

Figure 13 shows the location of the rack in the SFP; Figure 14 shows where the samples originated. Figure 15 is a view of the rack removed from the reactor pool in a reracking operation and indicates the section where the sample was removed (see arrow). Figure 16 shows the welded area from the rack section and the SS nut and square washer; four weld areas were examined metallographically.

Table 9 shows the elemental analysis of the two metal plates joined at the weld. The rack section, nut, and washer were exposed to the pool water conditions shown in Table 9 for 6.7 yr (from September 7, 1972, to May 8, 1979). The concentration ranges for radioactive isotopes in the pool water are summarized in Table 10.

POINT BEACH RADIOACTIVITY ANALYSIS

The section of Point Beach fuel storage rack had a total β/γ surface reading of 165 μCi ; the principal isotopes were ^{58}Co , ^{60}Co , ^{106}Ru , ^{134}Cs , ^{137}Cs , and ^{54}Mn .

On June 1, 1978, radiation readings from a hand-held meter indicated ~ 2.5 mR/h (shielded) on the inner plastic package and 7 to 8 mR/h on the same package with no shield for low-energy beta radiation.

Figure 17 shows the section of SFP rack with a detailed account of amounts and locations of radioactivity. Activities on the nut and washer are shown in Figure 18.

FIGURE 13. Plan View of Spent Fuel Storage Racks in Point Beach (PWR) Pool

FIGURE 14 PLAN AND ELEVATION VIEWS OF RACK FROM WHICH SAMPLE WAS TAKEN
(SAMPLE LOCATION INDICATED BY CIRCLES)

FIGURE 14. Plan and Elevation Views of Point Beach Fuel Storage Rack from Which Specimen Was Taken (specimen location indicated by circles)

POINT BEACH MICROSTRUCTURAL ANALYSIS

Figure 19 shows the storage rack specimen as it was sectioned (five locations). Pieces 1 and 3 were analyzed for radioactivity using a germanium-lithium detector.

The major element compositions (Ni, Cr) are approximately equal on the two plates with slight variations in minor alloying elements (Mn, Si). The carbon content and impurity elements (P, S) are within ASTM specifications for

**LOCATION
WHERE
SPECIMEN
REMOVED**

Neg. 7912715-1A

FIGURE 15. Type 304 SS Spent Fuel Pool Rack After 6.7 yr
in Point Beach (PWR) Pool

nominal 304 SS (see Appendix B). Metallographic examination of two weld areas (see Figure 20) indicated no significant evidence of intergranular attack or other degradation.

The microstructural analysis suggested different processing histories for the two plates. The top plate in Figure 20 exhibited equiaxed austenitic grains indicating a mill-annealed condition. The lower angle plate had a cold-worked microstructure, characterized by inclusions and elongated austenitic grains aligned with the rolling direction.

The welds appear to be structurally sound with a dendritically solidified structure of delta ferrite islands within an austenite matrix. Areas adjacent to the fusion line show a fine-grained matrix of recrystallized grains next to the base metal; the base metal is characterized by large, carbide-free austenite grains formed by rapid cooling through the sensitization range.

Neg. 79D465-4

Neg. 79D465-5

FIGURE 16. Views of Rack Section, Nut, and Washer from Point Beach Rack (304 SS). Exposed to ~2000-ppm B as boric acid for 6.7 yr at 16 to 45°C.

TABLE 9. Rack Specimen Compositions and Pool Conditions^(a) for Point Beach Spent Fuel 304 SS Rack

Heat	Material	Rack Specimen Composition, ^(b) wt%													
		C ^(c)	Mn	P	S ^(d)	Si	Ni	Cr	Mo	V	Cu	Sn	Ti	Al	Co
PB1	304 SS	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	304 SS	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

(a) Pool Conditions

pH: 4.7 to 4.8

Boron: 1900 to 2100 ppm

Cl: <0.05 ppm

Temperature: 16 to 45°C (60 to 113°F)

(normally maintained between 21 and 27°C)

Storage Rack Radiation Readings: 20 mR/h at ~30 cm (1 ft) after rinsing rack

Fuel Assemblies Stored at Rack Position: A12 for 653 days and C16 for 222 days (locations near specimen location)

Residence Time in Pool: 6.7 yr.

(b) Racks supplied by Bechtel to Specifications ASTM-A240 and A276. Analysis performed by Jorgensen Steel Co., Seattle, Washington.

(c) 0.045 and 0.065, respectively, by LECO method.

(d) 0.029 and 0.024, respectively, by LECO method.

TABLE 10. Concentration Ranges for Radioactive Isotopes in Point Beach (PWR) Spent Fuel Pool Water, May 1979

<u>Isotope</u>	<u>Concentration, $\mu\text{Ci/ml}$</u>
^{57}Co	1×10^{-5} to 9×10^{-6}
^{144}Ce	2×10^{-4} to 5×10^{-6}
^{51}Cr	7×10^{-5}
^{103}Ru	1×10^{-5}
^{134}Cs	3×10^{-4} to 2×10^{-5}
^{137}Cs	3×10^{-4} to 2×10^{-5}
^{95}Zr	6×10^{-5} to 2×10^{-5}
^{95}Nb	1×10^{-4} to 3×10^{-5}
^{58}Co	6×10^{-4} to 4×10^{-4}
^{54}Mn	1×10^{-4} to 1×10^{-5}
^{60}Co	2×10^{-3} to 7×10^{-4}
^{141}Cs	5×10^{-6}
^{125}Sb	2×10^{-5} to 7×10^{-6}

The weld microstructure suggests that the weld was made with a single pass. The depth of penetration into the parent metal was not excessive.

The nut and washer formed several crevice areas. Figure 21 shows a cross section of the thread area of the nut. Although some mechanical working can be seen, no corrosion is evident. The washer has three crevice areas: between the nut and washer, the washer and bolt, and the washer and rack brace (see Figure 22). Although the washer surface in contact with the nut is scratched and gouged, no significant attack or corrosion was observed even in the damaged area. This is also the case between the washer and bolt.

A mild (up to $16\text{-}\mu\text{m}$) grain boundary penetration was observed between the washer and rack brace.

SUMMARY AND CONCLUSIONS - POINT BEACH

Four rack section welds, a nut, and washer were examined after 6.7 yr in the Point Beach (PWR) fuel storage pool. No significant attack or corrosion

FIGURE 17. Radioactivity Distribution of Point Beach Rack Section (304 SS)

was observed. Some grain boundary penetration was observed on the washer, but it was mild ($<16 \mu\text{m}$). The structural integrity of each component did not appear to be significantly changed during pool residence.

Neg. 79D465-5

FIGURE 18. Radioactivity Distribution of Point Beach Nut and Washer (304 SS)

Neg. 81F573-1

FIGURE 19. Fuel Storage Rack from Point Beach (PWR) Spent Fuel Pool (304 SS, 0.05 wt% C). Exposed to ~2000-ppm B as boric acid for 6.7 yr at 16 to 45°C.

FIGURE 20. Metallographic Examination of Point Beach Fuel Storage Rack Specimen PB2 (304 SS, 0.05 wt% C). Exposed to ~2000-ppm B as boric acid for 6.7 yr at 16 to 45°C.

FIGURE 21. Metallographic Examination of Point Beach Fuel Rack Nut (304 SS). Exposed to 2000-ppm B as boric acid for 6.7 yr at 16 to 45°C.

FIGURE 22. Metallographic Examination of Point Beach Fuel Rack Washer (304 SS).
Exposed to 2000-ppm B as boric acid for 6.7 yr at 16 to 45°C

EXAMINATION OF ZION (PWR) SPENT FUEL POOL RACK SECTIONS

Three sections of a 304 SS SFP rack from the Zion Nuclear Power Station (Zion, Illinois) were received at PNL for examination in December 1980. The three sections were cut from the upper edge of one rack. Figure 23 shows the location of the rack within the SFP, and Figure 24 shows one of the sections as it was received at PNL.

Section 2 (see Figure 23) was decontaminated by electropolishing, which was performed at PNL and sponsored by Quadrex Corporation. An electropolished specimen was subjected to metallographic examination; all other examinations were performed on as-received rack sections.

Each section exhibited an untarnished surface with some discoloration at the welds. The inside (fuel-side) surfaces (shown in Figure 25) were covered with a brown deposit that became thinner toward the top of the fuel storage cell. The deposit could be wiped off to reveal an untarnished surface similar to the outside of the sections.

Rack component specimens were decontaminated by electropolishing and analyzed by emission spectrometry. Results of the elemental analysis are presented in Table 11. The rack was exposed to pool water for approximately 4.8 yr (early 1976 to November 5, 1980). Pool chemistries and conditions are shown in Table 11.

ZION RADIOACTIVITY ANALYSIS

Radiation readings for two of the sections were taken as the sections were unpacked. The section sent for decontamination was 10,000- to 23,000-cpm β/γ on inside (fuel-side) surfaces with 1,500-dpm α . The corresponding readings on the outside surface were 5,000-cpm β/γ and <200-dpm α . After decontamination by electropolishing, inside and outside readings were below 200-cpm β/γ and <200-dpm α . During electropolishing, less than 1 mil of metal was removed.

The as-received rack section used for microstructural analysis had 75,000-cpm β/γ and 300-dpm α on inside surfaces and 5,000-cpm β/γ and <200-dpm α on outside surfaces.

SECTION 1- NOT DECONTAMINATED
 SECTION 2- NOT DECONTAMINATED
 SECTION 3- ELECTROPOLISHED

16 - 7 X 7 RACKS = 784 ASSEMBLIES
 4 - 3 X 7 RACKS = 84 ASSEMBLIES
 TOTAL = 868 ASSEMBLIES
 IN ORIGINAL RACKS

FIGURE 23. Zion Spent Fuel Pool Arrangement. Replacement racks have storage locations for 2112 assemblies as compared to 868 assemblies for original racks.

Neg. 81F373-3

FIGURE 24. Fuel Storage Rack Section from Zion (PWR) Spent Fuel Pool (304 SS). A complete module of the type shown above accommodates a PWR fuel assembly for storage. Exposed to 2000- to 4000-ppm B as boric acid for 4.8 yr at 21 to 27°C.

Neg. 81F573-2

FIGURE 25. Inside of Fuel Storage Rack Section from Zion (PWR) Spent Fuel Pool (304 SS, 0.06% C). Exposed to 2000- to 4000-ppm B as boric acid for 4.8 yr at 21 to 27°C.

ZION MICROSTRUCTURAL ANALYSIS

Ten welds on one rack section were visually examined; four of these welds were examined metallographically. One metallographic sample was also examined by SEM and Auger spectrometry. A total of 17 welds were examined visually. None of the welds examined visually showed signs of degradation or corrosion, and all appeared structurally sound.

TABLE 11. Rack Specimen Compositions and Pool Conditions^(a) for Zion Spent Fuel
304 SS Rack

		Rack Specimen Composition, ^(b) wt%													
Heat	Material	C ^(c)	Mn	P	S ^(c)	Si	Ni	Cr	Mo	V	Cu	Sn	Al	Ti	Co
Z1D	304 SS	0.048			0.012										
Z3B	304 SS	0.058	1.57	0.031	0.013	0.50	8.36	18.25	0.46	0.040	0.20	0.019	0.010	0.006	0.122
Z3D	304 SS	0.057	0.48	0.020	0.010	0.54	8.15	18.73	0.39	0.028	0.19	0.013	0.016	0.007	0.094
Z5B	304 SS	0.059	1.61	0.031	0.019	0.93	8.89	18.80	0.48	0.032	0.14	0.013	0.009	0.005	0.163

(a) Pool Conditions

pH: 4.0 to 4.7
 Boron: 2000- to 4000-ppm B as H₃BO₃
 Cl⁻: <0.15 ppm
 F⁻: <0.15 ppm
 Mg: <1.0 ppm
 Ca: <1.0 ppm
 Li: 0.2 to 2.2 ppm
 Temperature: 21 to 27°C
 Total β/γ Activity: 2 x 10⁻³ μCi/ml in pool water
 Principal Isotopes: ⁵⁸Co, ⁶⁰Co, ¹³⁴Cs, ¹³⁷Cs
 Residence Time in Pool: 4.8 yr.

(b) Analysis performed by Jorgensen Steel Co., Seattle, Washington.

(c) Performed by LECO method.

Figure 26 shows one of the four welds examined by optical metallography and is representative of the other welds examined. Oxide films were too thin to resolve. Minor pitting in the crevice between the brace and the section wall did not exceed 20 μm in over 4.8 yr and was observed only in the brace metal. The differing microstructure of the brace (Z51B) suggests possible sensitization. Despite this, the pitting attack was well within the corrosion rates reported for stainless steel in dilute boric acid environments.⁽²⁴⁾ Some grain boundary widening at the metal/water interface is seen, but it is less than 10 μm in the worst case. The grain boundary widening is seen on the inside of the rack wall and is not seen to be a function of distance from welds or the amount of deposit. This widening may have been related to the manufacturing of the rack rather than to a corrosion process during service. Some dark grain boundaries seen near the weld in sample 1B (see Figure 27) were analyzed by SEM-EDAX and Auger techniques. SEM-EDAX studies showed that the grain boundaries were eroded and free of contaminants. Auger analysis confirmed the SEM-EDAX analysis and indicated that the grain boundary material was identical to the weld material with the oxygen concentration in both being the same. This weld material penetrates down the grain boundaries during the welding process.

SUMMARY AND CONCLUSIONS - ZION

Ten welds from a rack section exposed for 4.8 yr in the Zion (PWR) Nuclear Power Plant SFP (>2000-ppm B) were visually examined, and four welds were examined metallographically. No atypical welds or evidence of excessive welding temperature were observed; all welds appeared to be structurally sound. Some minor corrosion (<20 μm in depth) was detected in the crevice between the brace and the section wall.

For a service exposure of 4.8 yr, the rack sections were in extremely good condition. There appeared to be no significant corrosion or degradation problems; the integrity of the components appeared to be unchanged from preservice conditions.

FIGURE 26. Metallographic Examination of Zion Sample 4B (304 SS 0.06% C). Exposed to 2000- to 4000-ppm B as boric acid for 4.8 yr at 21 to 27°C.

45407C

A

4R407A

4R407E

23609

A-SEM

23607

B-SEM

FIGURE 27. SEM Examination of Zion Sample 1B (304 SS). Exposed to 2000- to 4000-ppm B as boric acid for 4.8 yr at 21 to 27°C.

EXAMINATION OF LA CROSSE (BWR) SPENT FUEL STORAGE RACK SECTIONS

In April 1981 PNL received from Dairyland Power Cooperative's LACBWR Power Plant (La Crosse, Wisconsin) ten sections taken from the two types of SFP racks used at that facility. The older rack consisted of 304 SS with a 6.35-mm (1/4-in.) boronated SS plate (≥ 2 wt% B) between every other assembly. The newer rack used 304 SS with a boron sandwich (35 wt% B_4C dispersion between 1.04-mm Type 1100 aluminum sheets) on the outside faces. An elemental analysis of each material is given in Table 12.

The boronated SS rack was used for temporary storage from December 1968 to August 1972 when continuous storage of spent fuel was initiated. The new boron racks were placed in the pool in December 1976. Both types of racks were removed from the pool during reracking in July 1980. The racks were thus exposed to pool water for 11.5 and 3.5 yr, respectively.

Table 13 shows the pool chemistry and conditions that existed in the SFP and concentration ranges of radioactive isotopes in the pool water.

A visual inspection of the sections showed that the surfaces were generally untarnished with discoloration at the weld ranging from metallic to golden to black. No weld spattering was observed, suggesting that normal weld procedures were used. A brown deposit was present on the inner surfaces but could be easily removed to reveal a surface similar to the outer surfaces. All 36 welds that were examined on the La Crosse specimens appeared to be structurally sound.

LA CROSSE RADIOACTIVITY ANALYSIS

Data received from LACBWR indicated the activities and principal isotopes that were present. These data are summarized in Table 14 for samples L4, L9, and L10.

The activity was associated with the brown deposit and was smearable.

TABLE 12. Rack Specimen Compositions and Pool Conditions^(a) for La Crosse Spent Fuel Storage Racks^(b)

Heat	Material	Rack Specimen Composition, ^(b) wt%									
		C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Co
AC1	304 SS	<0.060	1.52	0.020	0.12	0.65	9.10	18.36	0.24	0.23	0.06
AC2	Borated SS	0.045	1.52	0.019	0.020	0.57	9.43	18.93			
NES	304 SS	0.046	1.23	0.027	0.011	0.42	8.27	18.05	0.21	0.09	0.22
NES	Boral	← Type 1100 Al alloy with 35% B ₄ C (nominal) by weight →									

(a) Pool Conditions

pH: 5.00 to 7.10, normal 6
Boron: None
Cl⁻: <0.02 ppm
Temperature: 13 to 27°C, normal 21°C
Conductivity: 0.33 to 2.10 μmho/cm, normal 0.52 μmho/cm
Water Activity (borated SS): 1.19×10^{-4} to 7×10^{-1} μCi/ml
(boral sandwich): 1.19×10^{-4} to 4.86×10^{-2} μCi/ml
Normal Water Activity: 3.2×10^{-3} μCi/ml
Total Alpha Activity: $<0.37 \times 10^{-7}$ to 5.6×10^{-6}
Normal Alpha Activity: 2.1×10^{-7}
Residence Times in Pool: Borated SS = 11.5 yr
304 SS and boral = 3.5 yr.

(b) Analysis performed by Jorgensen Steel Co., Seattle, Washington.

TABLE 13. Concentration Ranges for Radioactive Isotopes in La Crosse (BWR) Spent Fuel Pool Water

<u>Isotope</u>	<u>Concentration, $\mu\text{Ci/ml}$</u>
^{58}Co	9.30×10^{-4}
^{60}Co	1.7×10^{-4} , $4 \times 10^{-5}(\text{a})$
^{99}Tc	8.8×10^{-4}
^{131}I	2.7×10^{-3}
^{133}Xe	4.9×10^{-3}
^{134}Cs	2.5×10^{-3} , $1 \times 10^{-4}(\text{a})$
^{137}Cs	3.9×10^{-3} , $3 \times 10^{-3}(\text{a})$

(a) Readings taken in August 1979; all others taken during refueling outage in 1977.

TABLE 14. Radioactivity Measured on Surfaces of La Crosse Spent Fuel Storage Rack Specimens in Early 1981

<u>Type of Specimen</u>	<u>Sample</u>	<u>Rack Surface Activities, μCi</u>					
		<u>^{137}Cs</u>	<u>^{134}Cs</u>	<u>^{60}Co</u>	<u>^{57}Co</u>	<u>^{144}Ce</u>	<u>^{54}Mn</u>
SS/SS Borated Steel	L4	2.16	0.20	0.31	0.007	0.07	-
Boral	L9	0.04	0.00	0.30	0.001	0.01	0.002
SS/SS	L10	13.43	1.96	1.04	0.050	1.54	0.017

LA CROSSE METALLOGRAPHIC ANALYSIS

Two representative specimens with a total of four welds and a section of boral plate were metallographically examined.

Sample L4 was taken from the older rack about halfway up and on the end of the rack and had been exposed to pool conditions for 11.5 yr. Samples L9 and L10 were taken from the new rack that had been exposed to pool conditions for 3.5 yr. Sample L9, the boral plate, was taken from the center region of the rack, while sample L10 was taken from the middle of the top edge of the rack.

Sample L4 is shown in Figure 28 and consists of boronated stainless steel welded to 304 SS welded to another 304 SS plate. Figure 29 shows the micrographs of sample L4. No evidence of significant intergranular attack or other degradation can be seen. The microstructure of each plate indicated similar processing histories. The boral plate, sample L9, is shown in Figure 30 while Figure 31 shows the micrographs of the cross section. The boral plate was difficult to polish due to B_4C pullout. Although scratched and dented, a uniform 6- μm oxide layer covered the aluminum-clad surface. No evidence of other types of corrosion could be seen on the aluminum.

Figure 32 shows a view of specimen L10; Figure 33 shows metallographic sections of the specimen. No intergranular attack or other degradation is evident.

SUMMARY AND CONCLUSIONS - LA CROSSE

A total of 36 welds and two boral plates from Dairyland Power Cooperative's LACBWR Nuclear Power Plant were examined visually; five welds and a section of boral plate were examined metallographically. Exposure time was 3.5 to 11.5 yr. All welds were structurally sound, and none were atypical. No evidence of significant intergranular attack or corrosion was found. The integrity of the components did not appear to be significantly degraded by pool residence.

Neg. 810321-8-7

FIGURE 28. La Crosse (BWR) Fuel Storage Rack Specimen L4C (304 SS, <0.06% C). Exposed to deionized water for 11.5 yr at 13 to 27 °C.

FIGURE 29. Metallographic Examination of La Crosse (BWR) Fuel Storage Rack Specimen L4C (304 SS, <0.06% C). Exposed to deionized water for 11.5 yr at 13 to 27°C.

Neg. 8103218-25

FIGURE 30. La Crosse Boral Plate Cross Section - Specimen L9. Exposed to deionized water for 3.5 yr at 13 to 27°C.

FIGURE 31. Metallographic Examination of La Crosse Boral Plate Cross Section – Specimen L9. Exposed to deionized water for 3.5 yr at 13 to 27°C; 35 wt% B₄C in an aluminum matrix clad with aluminum plates.

Neg. 8103218-30

Neg. 8103218-29

FIGURE 32. Two Views of La Crosse (BWR) Fuel Storage Rack Specimen L10 (304 SS). Exposed to deionized water for 3.5 yr at 13 to 27°C.

FIGURE 33. Metallographic Examination of La Crosse (BWR) Fuel Storage Rack Specimen L10 (304 SS, 0.048% C). Exposed to deionized water for 3.5 yr at 13 to 27°C.

EXAMINATION OF RBOF STAINLESS STEEL AND ALUMINUM COMPONENTS

PNL received three SS specimens in April 1981 and one aluminum specimen in July 1981 from the Receiving Basin for Offsite Fuel (RBOF), Savannah River Plant, near Aiken, South Carolina. RBOF is operated for DOE by E. I. du Pont de Nemours and Company; its principal function is to store fuel from university research reactors.

The specimens included two sections of a 304 SS roller table, a section of a SS ruler that had been mounted on the table, and an aluminum spacer from a spent fuel rack. The RBOF pool chemistry and conditions are summarized below:

pH = 5.5 to 7.0

B = None

Cl⁻ = <10 ppb

Temperature = 28 $\pm$ 3°C

Activity = 1 to 3 x 10⁻⁴ - 10⁻³ μ Ci/ml

Isotopes = 90% ¹³⁷Cs.

The 304 SS alloy specimens were exposed to pool water for 15 yr (1965 to September 1980). The location of the sample examined from the roller table is shown in Figure 34. The ruler was mounted on the top of the table, forming a stainless-to-stainless crevice. Figure 35 shows one of the samples as removed.

By visual examination the SS surfaces were clean and untarnished except for areas within and adjacent to the weld that were oxidized during welding. The outer surfaces were bright where the weld had been ground flush with the base metal. The inner surfaces of the weld show no spatter or evidence of excessive welding conditions. There were a number of <0.5 mm diameter light brown spots on one weld and in the markings on the ruler. No other evidence of corrosion or degradation was observed. All welds appeared structurally sound. A total of 14 welds on the RBOF table specimens were visually examined, and 2 welds were metallographically investigated.

The aluminum rack spacer (13.5 x 8.3 x 0.635 cm) was manufactured from 6061-T6 aluminum alloy and exposed to pool water for 16 yr (March 1965 to

Neg. 81F573-4

FIGURE 34. Roller Table Section (304 SS) from Receiving Basin for Offsite Fuels (RBOF), Savannah River Plant. Exposed to deionized water for 15 yr at $28 \pm 3^\circ\text{C}$; box shows section examined.

April 1981). The spacer was in contact with irradiated Zircaloy-clad fuel from the Heavy Water Components Test Reactor (HWCTR) at the Savannah River Plant, Aiken, South Carolina. The rack spacer is shown in Figure 36 as it was received at PNL. The aluminum spacer had a dull, tarnished appearance. The more heavily tarnished areas run vertically and may have developed at points of contact with the fuel cladding. An unidentified gray paint-like deposit covered <5% of the spacer surface. No evidence of gross corrosion or degradation was observed in visual examinations.

Neg. 8103218-33

FIGURE 35. RBOF Fuel Storage Roller Table Specimen RIC (304 SS). Exposed to deionized water for 15 yr at $28 \pm 3^\circ\text{C}$.

RBOF METALLOGRAPHIC ANALYSIS

Two SS welds and a cross section of the SS ruler were examined metallographically (see Figures 37 and 38, respectively). No abnormal or atypical welds were found. In Figure 37 grain boundary widening was observed in the worked inner side of the bend, possibly due to very mild corrosion or to the manufacturing process. If due to corrosion, the widening is $<75 \mu\text{m}$ in 15 yr, which would be considered minor. There was no other significant evidence of intergranular attack or other degradation.

The aluminum specimen was sectioned as shown in Figure 36 to include the following features: the crevice region between the spacer and the other

FIGURE 36. Front and Back Views of RBOF Aluminum Fuel Rack Spacer. Exposed to deionized water for 16 yr at 28 $\pm 3^{\circ}\text{C}$.

aluminum plates in the fuel rack, the tarnished region at the possible Zircaloy cladding contact, and an area that was open to the pool water. Figure 39 shows the oxide layer at various points on the sample; at some points the oxide layer was pulled away from the base metal by the mounting material. The oxide layer varied from 8 to 20 μm . The thickest oxide was found near the middle of the sample and in the tarnished region; the layer was thinnest in the crevice region.

FIGURE 37. Metallographic Examination of RBOF Fuel Storage Roller Table Specimen RIC (304 SS). Exposed to deionized water for 15 yr at $28 \pm 3^\circ\text{C}$.

FIGURE 38. Metallographic Examination of RBOF Fuel Storage Roller Table Ruler Specimen R3B (304 SS). Exposed to deionized water for 15 yr at $28 \pm 3^\circ\text{C}$.

Figure 40 shows the aluminum specimen after a 0.5% HF etch. The elongated grain structure indicates that the aluminum was rolled or extruded. Figure 40B shows some evidence of local mechanical damage. Some mild pitting occurred in the crevice, tarnished, and open regions. The pitting depth was less than $20\ \mu\text{m}$ and showed no preference for grain boundary attack. No significant difference in pitting or other corrosion was observed in the crevice or tarnished regions compared to the open regions of the sample. No other degradation or corrosion was observed on the aluminum spacer.

FIGURE 39. Oxide Layers on RBOF Aluminum Fuel Rack Spacer. Exposed to deionized water for 16 yr at $28 \pm 3^\circ\text{C}$.

FIGURE 40. Metallographic Examination of RBOF Aluminum Fuel Rack Spacer Alloy 6061-T6. Exposed to deionized water for 16 yr at $28 \pm 3^\circ\text{C}$.

SUMMARY AND CONCLUSIONS - RBOF

After 15 yr of exposure to RBOF deionized pool water, the 14 alloy 304 SS welds and the SS ruler-crevice section showed no evidence of significant intergranular attack or other corrosion. After 16 yr of exposure in the RBOF pool in close proximity and probable contact with Zircaloy-clad spent fuel, the aluminum spacer had developed an 8- to 20- μm oxide layer and some minor pitting ($<20\ \mu\text{m}$).

The integrity of the stainless steel and the aluminum specimens was not significantly changed by corrosion after 15 to 16 yr of pool residence.

THREE MILE ISLAND (TMI) PIPE EXAMINATION

During the period from November 1974 to February 1977 several reactors with boric acid systems developed SCC in pipes at locations near welds.^(a) In response to the appearance of these cracks the NRC issued Circular 76-06.⁽²⁴⁾ The circular indicated that:

- cracks were adjacent to weld zones in thin-walled low-pressure pipes, which were not part of the reactor coolant system
- cracking occurred in normally stagnant pipe segments
- analysis of surface products indicated that chloride ions are a probable factor, together with a stress state in the pipe due to welding and/or fabrication.

During the refueling outage of TMI-Unit 1 in February 1979, six^(b) through-wall cracks were observed near welds in the SFP cooling system piping and one at a weld in the decay heat removal system. The cracks were detected due to local crusts of boric acid that developed when the solution seeped through the cracks and the small amount of water evaporated. Radiation levels on the crusts were low due to low concentrations of radioactivity in the pool water.

The TMI-Unit 1 SFP system has two cooling loops: loop A, which is redundant and therefore normally stagnant; and loop B, which is normally circulating. All but one of the pipe cracks developed in the stagnant loop A piping, and all cracks occurred in weld heat-affected zones in pipe sections with carbon levels exceeding 0.07 wt%.

An extensive nondestructive testing (NDT) program was instituted at TMI-Unit 1, and ultrasonic testing identified several hundred "indications." Subsequent calibrations and metallographic correlation suggested that only 39 of

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(a) For example, see entries in Appendix A for Arkansas Nuclear No. 1, 9-26-76, 12-1-76, 5-5-77, 11-25-77; Millstone, 4-22-76; San Onofre, 1-17-77; Trojan, 10-10-77; 3-8-78. Cracks have also developed in boric acid system piping at three Japanese PWRs.

(b) A seventh crack was later discovered.⁽⁴⁾

the indications represented partial through-wall cracks. Discovery of the TMI-Unit 1 pipe cracks prompted NRC to issue IE Bulletin No. 79-17.⁽²⁵⁾

The Electric Power Research Institute (EPRI), Palo Alto, California, commissioned studies to provide insight into the nature of the cracking phenomenon; they can be summarized as follows:

- There has been no evidence that 304 and 304L SS will crack in uncontaminated oxygen-saturated boric acid environments characteristic of PWR pools.
- Stainless steel in boric acid environments with 15-ppm Cl^- added has cracked in one test but not in another. The incidence of cracking may be related to the degree of sensitization, although this has not been confirmed.

The cracking phenomenon is not currently regarded as a safety problem. However, costs associated with replacing cracked piping have prompted EPRI to investigate a program to conduct mechanism studies and identify countermeasures.

The Spent Fuel and Fuel Pool Component Integrity Program at PNL has conducted a study of the pipe cracking phenomenon to provide a perspective relevant to spent fuel component integrity. A section of TMI-Unit 1 pipe (Figure 41) with a through-wall crack was subjected to metallographic and analytical examination, and sections of the pipe have undergone SCC tests in various media. Results of the examination were presented at the Corrosion 81 Conference in Toronto.^(26,27) Intergranular SCCs (IGSCCs) found in the weld heat-affected zone of the pipe have been evaluated by a series of analytical tests. The intergranular nature of the cracks was identified by optical metallography and scanning electron microscopy (SEM). The presence of Cl^- was observed by energy dispersive x-ray analysis (EDAX) and Auger electron spectroscopy (AES). Sulfur was observed by EDAX and AES both as particles and uniformly on the fracture surface.

Constant extension rate tests (CERT) performed on specimens removed from the pipe and containing the weld heat-affected zones demonstrated that 15-ppm Cl^- in a 1.3 wt% H_3BO_3 solution was sufficient to cause IGSCC when the

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APPENDIX A

INCIDENT REPORTS

APPENDIX A

INCIDENT REPORTS

<u>Date of Event</u>	<u>Reactor (Type)</u>	<u>Occurrence</u>	<u>Description</u>	<u>Consequence</u>	<u>Probable Cause</u>	<u>Corrective Action</u>	<u>Reference Locket No.</u>
5/5/81	San Onofre2 (PWR)	Leak from refueling water to primary makeup water	Refueling water pump discharged back into primary makeup tank contaminating radio-chemistry lab water	Erroneous boron results for reactor refueling cavity and SFP	Leakage through a seal flushing line check valve	Revising procedures to prevent opening the primary seal valves during cavity filling. Remove and cap out-of-service line.	50-206
1/17/80	Salem 1 (PWR)	Leak in spent fuel pool	A leak of about 1.9 gpm was discovered at tube drain	Not given	Unknown	The level of SFP has been maintained according to technical specifications ongoing testing	50-272

SUMMARY OF EVENTS INVOLVING SFP EQUIPMENT

<u>Date of Event</u>	<u>Reactor (Type)</u>	<u>Occurrence</u>	<u>Description</u>	<u>Consequence</u>	<u>Probable Cause</u>	<u>Corrective Action</u>	<u>Reference Docket No.</u>
2-79	Three Mile Island-1 (PWR)	7 Through-wall pipe cracks.	Through-wall cracks @ welds in the spent fuel cooling system piping and one at a weld in the decay heat removal system.	Local boric acid build-up on outer pipe surface; no significant water leakage.	Intergranular stress corrosion cracking originating on the pipe. I.D. Cracking localized to HAZ where the type 304 s/s was sensitized during welding. Stagnant pipe conditions may have also contributed.	Sections of cracked piping replaced with 304 S/S. Broad-based UT exam of potentially affected systems utilizing an approved technique was initiated.	50-289
9-11-78	Turkey Pt. 4 (PWR)	Mechanical seal failure.	S.F. pool cooling pump mechanical seal failed.	~150 gal. of radioactive contaminated water leaked from pumproom floor to outside area.	Not determined at time of report.	Exposed area decontaminated. Pump overhaul scheduled.	50-251 Report date: 10-11-78
3-17-79	Plant Hatch (BWR)	Crack in fuel pool transfer canal.	A 6-8" long crack penetrating the 1/4" liner plate was discovered.	Concrete structure beneath liner plate exposed. Since canal had never been filled with water, there was never any contamination risk.	Impact on the floor plate by a sharp pointed object.	To weld a plate over the crack and test for leakage.	50-321 Report date: 4-18-79

EVENTS INVOLVING IRRADIATED FUEL BUNDLES

Date of Event	Reactor	Operation	Event	Consequence	Probable Cause	Corrective Action	Reference Docket No.
5-28-77 (Report date)	Oyster Creek (BWR)	Lowering of fuel assembly and mast into SFP racks.	Fuel assembly and mast dropped during movement.	Movement arrested by cable drum brake without further damage when operator released grapple lower level.	Shearing of 6 bolts coupling mast speed reducer to cable drum. (All but 2 had failed at earlier date.)	Not given	50219-1144
1-28-78	Peach Bottom #3 (BWR)	Fuel Element Insertion into core.	Fuel assembly released from grapple and fell across core.	Element settled 20' from horizontal with nosepiece under handle of an element and supported by handle of another. No radioactive release.	Fuel grapple opened due to being bumped by operator's elbow, followed by a slack cable signal when the fuel assembly nose cone contacted the core as it was being lowered, thereby satisfying interlocks for the grapple to open.	Safety guard has been installed around switch to prevent accidental actuation.	50278-1160 - supp. info. 50219-1144
1-20-74	Pilgrim (BWR)	Transfer of irradiated fuel bundle from storage position to channel inspection facility.	During movement, fuel bundle became detached, dropping 20 ft. Bundle came to rest with nose piece on bottom of SFP with handle resting against side of storage tank.	Refueling floor immediately evacuated. No activity release or personnel exposure. Reactor shut down for 23 days. Exposure of dropped fuel at time of shutdown was 3600 mRad/MTU. No leakage of SFP liner observed.	Under investigation	Dependant on evaluation of probable cause	50293-227
10-77 (Report date)	Oconee 1 and 2 (PWR)	Fuel bundle transfer in SFP.	Fuel Bundle damaged during handling.	Upper ten inches of a corner rod bent outward at 45°. This rod kept the bundle from being placed in a rack or shipping cask.	Bent fuel rod removed from assembly.		50-269, 270 and 287

a) See Table of References for additional fuel handling events in the same time period.

Date of Event	Reactor (Type)	Occurrence	Description	Consequence	Probable Cause	Corrective Action	Reference Docket No.
10-27-78	Calvert Cliffs (PWR)	Buckled floor plates and ruptured three plug welds.	Air pressure used in an attempt to clean obstruction in chase channels. Applied pressure raised and buckled floor plates in Pool #2 while raising plates and causing plug weld to rupture in Pool #1	Repairs required before pool filled with water	Incorrect procedure, air pressure should not be used to clear chase channels.	Chase channels cleared, air leaks and floor plates repaired.	50317-77 Report Date 11-6-78
6-16-78	Pilgrim 1 (BWR)	Crack in fuel pool cooling system weld.	Investigation revealed crack on saddle plate near weldment for drain valve on pump discharge tie line.	Fuel pool cooling area subject to water spray.	Cyclic stress fatigue of metal.	System secured, leak isolated. Weld ground out and rewelded. Evaluation to determine if installation of new pipe section will reduce stresses.	50293-1034
5-23-78	Turkey Point 4 (PWR)	SFP cooling pump shaft failure.	Shaft failure caused mechanical seal faces to partially open, resulting in water leakage at seal.	~50 gal. of water leaked from pump, contaminating two operators and small adjacent area. No off-site release.	Cause not clear - pump shaft and bearings sustained major damage.	Pump shaft and bearings replaced - tested satisfactorily.	50251-967
4-28-78	Connecticut Yankee (PWR)	Pressure buildup in poison cavity of spent fuel racks.	Significant gas generation and pressure buildup in annular poison cavity of racks presently storing spent fuel.	Swelling of annular neutron absorber cavity resulting in reduction in internal cell dimensions and water gap between adjacent cells.	Design deficiency: decomposition of moisture in plates and/or decomposition of binder material used in plate manufacture.	Not given	50213-837
4-26-78	San Onofre 3 (PWR)	SFP liner damaged by air overpressurization.	Air pressure used to remove debris from wall leak chase channels. Air pressure mistakenly applied to floor chase channels (not provided with venting areas).	Pressure buildup, due to lack of venting, bulged floor plates and ruptured limited number of concrete plug weld anchors.	Failure to obtain procedure.	Liner plates with deformations exceeding 1/8" replaced.	50-362
3-8-78	Trojan (PWR)	Small defect in weld in SFP cooling piping.	Weld defect found in a section of the SFP cooling system piping at a hanger attachment. (Small accumulation of boric acid crystals observed on pipe hanger to SFP cooling system return line)	Leak detected with no spent fuel in SFP. No effect on plant or public safety. Water leakage from defect nonradioactive.	Improper welding: burn through of pipe wall during initial welding of pipe hanger gusset to pipe.	Defective section of piping replaced. Visual inspection of cooling system resulted in no other defect discoveries.	50344-753
2-3-78	Dresden 1 (PWR)	Contaminated water release.	During plant shutdown, a 1/4" copper fuel pool filter pressure sensing line was found broken.	~200 gallons of contaminated water covered part of the rad-waste control room floor and ran under the door onto the ground outside.	Line froze and broke, due to 4 hour heating system outage.	Leak isolated and repaired, water samples taken, area decontaminated.	50010-917

Date of Event	Reactor (Type)	Occurrence	Description	Consequence	Probable Cause	Corrective Action	Reference Docket No.
12-14-77	Millstone 2 (BWR)	Failure of SRAS during filling of refueling pool.	During filling of refueling pool for a refueling outage from the refueling water storage tank (RWST), a sump recirculation activation signal (SRAS) did not occur.	Problem detected during filling operations - no hazards presented. Manual initiation of SRAS available in event of LOCA.	RWST level transmitter current output was too high to cause an SRAS trip. Offset in transmitter output caused by presence of air in unvented vertical section of U-tubing below transmitter.	Incident under review for corrective action.	50336-942
11-25-77	Arkansas 1 (PWR)	Two pinhole leaks in SFP cooling system.	Inspection found 2 pinhole leaks in SFP cooling system. Both were in HAZ, one in cooler outlet piping, the other downstream of discharge check valve. Pin head size boron buildup visible.	No water detected.	(Unknown)	Leaks monitored and repaired.	50313-967
10-10-77	Trojan (PWR)	Defect in pipe.	A small defect was found on the outer radius of an 8-inch stainless steel elbow on the discharge piping of the SFP cooling pump. (Pair of short, transverse, interconnected cracks adjacent to longitudinal seam weld on outer bend of elbow.)		Manufacturing defect revealed by metallurgical analysis.	Elbow replaced.	50344-602
8-10-77	Sequoyah 1, 2 (PWR)	Damage to piping	Inspection showed some key supports were missing on SFP cooling pipe that was attached to suction side of pump.	Preoperational testing postponed pending installation of permanent supports and hydrostatic testing. During hydrostatic testing, pump bellows were damaged and the supports temporarily failed.	Installation deficiency: ties not adequately designed for system overpressure.	Damaged bellows and supports replaced.	50327-427,379
8-2-77	Pilgrim 1 (PWR)	Radioactive material spill.	During change out of fuel pool demineralizer, a valve was opened in the radwaste truck lock area to facilitate venting of spent resin storage tank resulting in tank overflow.	~200 gallons of radioactive liquid released in lock area.	Personnel error: Consequences of leaving valve open and unmonitored was overlooked.	Not given	50293-924
6-23-77	San Onofre 2 & 3 (PWR)	Cracking of stainless steel liner plate material.	Final report on investigation concerning cracking discovered on 8 SS liner plates of SFP handling and storage facilities for these units.		Intergranular stress corrosion cracking.		50361-274 (interim report: 50361-198)

Date of Event	Reactor (Type)	Occurrence	Description	Consequence	Probable Cause	Corrective Action	Reference Docket No.
5-5-77	Arkansas 1 (PWR)	Leak in SFP discharge piping.	Leak discovered in pipe-to-pipe weld, in vertical run of piping, for the high discharge line to the SFP.	Contained leak, no hazard to health.	(Unknown)	Leak isolated, pipe will be repaired or if crack in HAZ of welded section of pipe, section of pipe will be replaced.	50313-727
4-21-77	Dresden 1 (BWR)	Ruptured vent line.	Fuel pool heat exchanger inlet vent line ruptured.	Spill of 2000 to 5000 gallons of fuel pool water onto sphere skirt and north end of warehouse floor. Fuel pool water level remained constant. Spilled water contained inside station with no release of activity.	Corroded and weakened short nipple on welded coupling failed.	Spill area isolated, dried and decontaminated. Surveys indicated surface contamination below Dresden contamination control limits. The nipple was replaced.	50010-740
2-28-77	Quad Cities 22 (BWR)	Bypass of "A" fuel pool radiation monitor.	Fuel pool radiation monitor bypassed in control room due to intermittent spiking causing initiation of standby gas treatment system (SBGTS).	(Not Given)	Faulty GM tube in the sensor converter assembly of Unit 2 - "A" fuel pool monitor.	Reactor Building Ventilation System isolated SBGTS initiated. Failed sensor converter assembly, faulty GM tube replaced.	50265-1056 (supplemental info). 50265-1005
1-20-77	Arkansas 1 (PWR)	Leak in pipe weld.	A leak was discovered at a weld on the 3-inch side of a 3- to 2-inch reducer in the resin transfer line for SFP demineralizer.	Small seepage of liquid that was contained in controlled area.	(Unknown)	Section of 2-inch pipe and reducer replaced.	50313-716
1-17-77	San Onofre 1 (PWR)	Flaw in pipe.	Through-wall flaw in portion of containment spray line (6-inch schedule 10 ASTM 312 type 304 SS). Boric acid crystals observed on portion of line. Dye penetrant examination revealed linear and rounded imperfections in the line.	(Not Given)	Stress corrosion cracking produced by neglected annealing of welds.	Pipe replaced.	50206-664
1-16-77	G. E. Morris	Temporary shutdown of	Power supply interruption which shut down circulation pump. Upon restart, cooling pipe froze leading to a broken line.	Cooling system was shut down for several weeks with 225 tons of fuel intact. Fuel pool came to an equilibrium temperature of 115°F.	Loss of Power.	System repaired.	701308
12-1-76	Arkansas 1 (PWR)	Water leak from valve in SFP cooling system.	Leakage detected on weld downstream of valve SF-11 between valve and T-section of pipe. Dye penetrant test of area revealed no penetrating cracks. Dye penetrant test on valve above SF-10 revealed crack in HAZ of valve SF-11.	Minute amount of leakage was contained, no hazard to health or safety of public.	(Unknown)	SF-11 valve replaced.	50313-693

Date of Event	Reactor (Type)	Occurrence	Description	Consequence	Probable Cause	Corrective Action	Reference Docket No.
11-12-76	Brunswick 2 (BWR)	SFP water level inadvertently lowered.	During shutdown, SFP water level lowered 5 inches below minimum required level. Air leak on SFP inner gate bellows allowed leakage.	Leakage of water through bellows entered drain. Drain valve shut and bellows repressurized.	Leak on pool gate bellows.	Bellows repaired.	50324-988
11-3-76	Peach Bottom 1 (HTGR)	Tube leak.	Heat exchanger tube leak on SFP cooling system allowed SF water to enter the discharge canal through the service water system. Valve leakage prevented isolation of heat exchanger. Adjustment of differential pressure across heat exchanger allowed for minimal release.	Maximum release was 1000 gallons per day for 4 days. This was diluted by 3500 gpm service water resulting in a contaminant activity of 4×10^{-11} $\mu\text{Ci/cc}$.	(Unknown)	Blank flanges installed in service water lines removing heat exchanger from service, and isolating leak. Total decay heat load minimal, pool temperature remained constant at 69 F.	50171-215
9-26-76	Arkansas 1 (PWR)	Leak in SFP cooling system.	Leaks discovered at weld on SFP outlet nozzle and in piping upstream of SFP valve. Three cracks (1.5 to 2 inches long) in nozzle and two in weld (1 to 1.5 inches long) HAZ above valve detected.	Leak minor, contained in reactor auxiliary building. No hazard to health or safety of public.	(Unknown) - probable cause is SCC (failure mode as in 50-313/74-118)	Outlet nozzle replaced. Ten inch schedule 10 piping replaced with 6-inch section between elbow and SFP valve.	50313-640
6-27-76	Calvert Cliffs (PWR)	Leak in pump valve.	Pinhole leak was discovered in bonnet of #11 boric acid pump suction relief valve.	#11 boric acid storage tank was isolated and drained.	Carbon steel components not compatible with hot boric acid solutions.	Carbon steel bonnet and spring replaced with stainless steel equivalents.	50317-654
4-22-76	Millstone (BWR)	Leak in pipe.	Leakage found in boric acid concentrates pump discharge line. Pinhole leak in 1.5-inch schedule 40 304 stainless steel pipe downstream of flanged connection to pump discharge lines.	Minimal leakage contained to radwaste system.	Possible chloride stress corrosion cracking (see LER-76-25/31)	Leak repaired with welded patch.	50336-450
11-6-75	Turkey Point (PWR)	SFP leak.	Unit #4 SFP and concrete wall leakage.	Leak rate of 2 gph, channeled to radwaste system.	Crack in liner.	Special SS drain trough utilized to divert the leakage to the waste holding tank.	50251-419

<u>Date of Event</u>	<u>Reactor (Type)</u>	<u>Occurrence</u>	<u>Description</u>	<u>Consequence</u>	<u>Probable Cause</u>	<u>Corrective Action</u>	<u>Reference Docket No.</u>
8-18-75	Turkey Point 3 & 4 (PWR)	Malfunction of SFP cooling pumps.	Bearing and pump shaft failures of SFP cooling pumps within four month period.	SFP heat exchanger outlet valve limited to 25% full volume. SFP heat transfer demands still met with reduced flow.	Excessive internal pump stresses caused by fluid cavitation.	After each event, pumps repaired, tested and reinstated.	50250-394
4-13-75	Turkey Point (PWR)	SFP water spill.	Emergency SFP pump discharge hose connection separated from piping.	7400 gallons of SFP water was released: -4440 gallons were recovered by the waste holding tank. -2960 gallons were absorbed by the ground.	Hose connection not strong enough to withstand pressure.	Larger pump with stainless steel pipe connections installed to replace the smaller emergency SFP pump.	50251-356
3-6-75	Turkey Point (PWR)	Broken restraint springs.	Six broken stainless steel restraint springs, one found on SFP bottom, others still attached to fuel racks.	None, pool was drained.	Under investigation	Springs replaced.	50251-338 (supplemental information: 50251-420
11-18-74	St. Lucie 1 (PWR)	Crack in concrete.	A hairline crack in the reinforced concrete refueling canal was detected. The crack was located near an embedded steel plate used to support a fan cooler platform.	Not given	(Unknown) - possibly misplaced rebar.	Cracked concrete repaired.	50335-142
10-26-74	Turkey Point (PWR)	Leak in SFP liner.	Leakage detected in #3 SFP liner.	Leak rate ~1 gpm compared to pre-operation leak rate of 2 gph. Additional leakage of 22-27 gpm due to either liner or vessel to cavity leak.	Under investigation - no indication of any detrimental effect from borated water or concrete.	A small head tank connected to the monitoring channels was installed to fill the spaces between the liner and concrete.	50250-324
8-2-74	Surry 1 (PWR)	SFP water spill.	Water was being transferred from the SFP to the boron recovery system using a temporary hose line. One end of the line connection separated from its fitting.	A maximum of 150 gallons of SFP water escaped to the storm drains.	Temporary hose connection could not withstand pump pressure.	To prevent future connector slippage, the fitting and connector were welded together.	50280-244/238

Date of Event	Reactor (Type)	Occurrence	Description	Consequence	Probable Cause	Corrective Action	Reference Docket No.
11-1-73	Connecticut Yankee (PWR)	Valve diaphragm leakage.	RWST thermosiphon heater return isolation valve leaked through the diaphragm of a carbon steel valve.	A maximum of 270 liters leaked from the valve which was subsequently diluted by service water at 6000 gpm before leaving plant. Leakage activity within MPC limits by at least two orders of magnitude.		Stainless steel valve bonnet and travel stop installed. A different type of diaphragm employed (curved rubberized type with extended lifetime).	50213-243
6-5-72	Prairie Island (PWR)	Pitting of stainless steel tank	Pits were observed in SFP S/S tanks during cleaning and sand-blasting operation. (One pit in SFP #1, 500-1000 pits in SFP #2). Tanks constructed of 304 SS. Both experienced similar storage history. Largest percentage of pits adjacent to welds, rate of attack: 100-150 mils/year.	Not given.	Stainless steel sensitization not indicated, but initial corrosion proceeds along grain boundaries. Water chemistry was not detrimental and plate materials not defective.	Pits repaired with approved welding techniques.	50282-68 50306-65
1972	G. E. Morris AFR SFP	Small puncture in SFP liner.	A 18-100 shipping cask tipped against the loading pit wall puncturing the stainless steel liner.	Damage area was crescent-shaped, 4.1 ft long with a maximum parting of 41.5 inches. Pool water entered an area between the stainless steel liner and the concrete wall.	Tipped shipping cask.	Temporarily repaired within 27 hours. Permanent repair in 12 days.	NEOG-20969

APPENDIX B

FABRICATION PARAMETERS FOR STAINLESS STEEL
SPENT FUEL POOL COMPONENTS

APPENDIX B

FABRICATION PARAMETERS FOR STAINLESS STEEL SPENT FUEL POOL COMPONENTS

Use of Type 304 stainless steel (SS) in spent fuel storage module fabrication is in accordance with ASTM SA-240 specifications. These specifications include a minimum tensile strength of 515 MPa (75 ksi), a minimum yield strength of 205 MPa (30 ksi), a maximum Brinell hardness number of 183 (or Rockwell B number of 88), and a 40% minimum elongation of 50 mm (2 in.) for the stainless steel. The material must also conform to chemical composition requirements in the specifications; these requirements are specified in Table B.1.

TABLE B.1. Chemical Requirements for Type 304 Stainless Steel

<u>Element</u>	<u>Composition, %</u>
Carbon ^(a)	0.08
Phosphorus ^(a)	0.045
Silicon ^(a)	1.00
Nickel	8.00 to 10.50
Manganese ^(a)	2.00
Sulfur ^(a)	0.03
Chromium	18.00 to 20.00
Others	0.010 ^(a)

(a) Maximum.

The steel shall be heat treated to meet the mechanical property requirements of the specifications (above) and shall be of uniform quality. Grinding to remove surface imperfections is allowed for material with a No. 1 finish, provided such grinding does not reduce the thickness or width at any point beyond the permissible variations in dimensions.

To further increase the corrosion resistance of 304 SS, the following additions and modifications have been suggested by Brooks and Perkins:

- The maximum carbon content in the steel shall be 0.055% instead of 0.08%.
- The delta ferrite content number (an empirical number based on chromium and nickel equivalents that determines the percentage of delta ferrite in the material) for the steel shall be restricted to between 4 and 8.
- Additional care shall be taken during any welding operations; the tungsten inert gas (TIG) fusion process for welding shall be used; and the heat input during welding shall be limited to 15,000 J (less than half the current nuclear industry's recognized limit).
- A melting practice shall be followed in which the residual aluminum and calcium used for the melt deoxidation shall be analyzed so that the total percentage of aluminum and calcium in the melt shall not exceed 0.01%.

APPENDIX C

WELDING THERMAL ANALYSIS

APPENDIX C

WELDING THERMAL ANALYSIS

Welding parameters from Reference 1, recommended settings for shielded metal arc welding (covered electrode) on 1/4-in. 304 stainless steel (SS) plate:

I, welding current	160 amps
ts, travel speed	5.5 imp (2.3 mm/s)
deposited weld metal	0.262 lb/ft
v, assumed voltage	20 V
f, heat input arc efficiency	0.8 to 1.0

FIGURE C.1. Design Schematic of Heat Transfer Model

Application of above parameters in heat transfer equation allows for calculation of net heat input (H_{net}):

$$\text{3-leg, T-joint: } H_{\text{net}} = \frac{2}{3} \frac{fVI}{ts} = 835.2 \text{ J/mm}$$

MATHEMATICAL ANALYSIS

Peak Temperature Calculation

- Single pass, full-penetration butt weld in sheet or plate. Peak temperature distribution in base metal adjacent to weld given by:

$$\frac{1}{T_p - T_o} = \frac{4.13 \rho C_p t Y}{H_{NET}} + \frac{1}{T_{mp} - T_o}$$

where T_p = peak temperature at distance Y from weld fusion line

T_o = initial preweld temperature of plate

ρ = material density

C_p = material specific heat

Y = distance from weld fusion line

T_{mp} = material melting temperature

t = plate thickness.

- sample calculation, at $Y = 1.5$ mm:

$$\frac{1}{T_p - 20^\circ\text{C}} = \frac{4.13(7.9 \times 10^{-3} \text{ g/mm}^3)(0.38 \text{ J/g-K})(6.4 \text{ mm})(1.5 \text{ mm})}{835.2 \text{ J/mm}} + \frac{1}{1410 - 20^\circ\text{C}}$$

$$T_p = 1179^\circ\text{C}$$

Cooling Rate Calculation

- Two-dimensional, assuming no radiative or convective heat transfer.
- Check for applicability of thin or thick plate cooling rate equation:

$$\tau = t \left(\frac{\rho C_p T_{int} - T_o}{H_{NET}} \right)^{1/2}$$

where τ = dimensionless parameter

T_{int} = temperature of interest.

If $\gamma > 0.9$, use thick plate cooling rate equation: (heat transfer through plate thickness)

$$\frac{dT}{dt} = \frac{2\pi k (T_{int} - T_o)^2}{H_{NET}}$$

If $\gamma < 0.6$, use thin plate equation: (surface opposite weld at approximately same temperature as welded surface)

$$\frac{dT}{dt} = 2\pi k \rho C \left(\frac{t}{H_{NET}} \right)^2 (T_{int} - T_o)^3$$

where k = material thermal conductivity transition behavior for $0.9 > \tau > 0.6$.

- For specific case of 3-leg heat transfer (7-joint):
 $\tau = 0.24$; use of thin plate cooling rate equation is warranted.

Sample Calculations

- Width of zone experiencing temperatures (peak $\geq 600^\circ\text{C}$ (lower sensitization temperature for austenitic stainless steels)

$$\frac{1}{600-20^\circ\text{C}} = \frac{4.13(7.9 \times 10^{-3} \text{ g/mm}^3)(0.38 \text{ J/g-K})(6.4 \text{ mm})Y}{835.2 \text{ J/mm}} + \frac{1}{1410-20^\circ\text{C}}$$

where $Y = 10.6 \text{ mm}$ from weld fusion line.

- Cooling rate at temperature (peak) of 600°C of weld metal or at a distance of 10.6 mm from weld fusion line.

$$\begin{aligned} \frac{dT}{dt} &= 2\pi(0.023 \text{ J/mm-g-}^\circ\text{C})(7.9 \times 10^{-3} \text{ g/mm}^3)(0.42 \text{ J/g-K})\left(\frac{6.4 \text{ mm}}{835.2 \text{ J/mm}}\right)^2 \\ &\quad \times (600-20^\circ\text{C})^3 \end{aligned}$$

$$\frac{dT}{dt} = 5.5^{\circ}\text{C/s}$$

ASSUMPTIONS

The following assumptions (2-4) were made in this preliminary thermal analysis:

- a point heat source moving in a straight line at a constant travel speed
- peak temperatures and cooling rates equations for thin plate materials, restricting heat transfer to two dimensions (in the plane of the plate)
- no radiative or convective heat transfer effects
- cooling rates maximum at the weld centerline, but cooling rate distribution curve is such that centerline cooling rates are only 5 to 10% higher than actual weld heat-affected zone values; therefore, these rates assumed to be representative values for the immediate heat-affected zone.
- welding parameters taken from Reference 1 and apply to a covered electrode shielded metal arc welding process (manual technique) for 1/4-in. 304 SS plate.
- geometry of welded joint allows the assumption of a T-joint weld configuration on flat plates until heat transfer effects from the center hole and the other flange of the stainless steel angle take effect; center hole tends to restrict heat flow while the opposite flange promotes heat transfer; these competing effects appear to contribute to deviations from normal heat flow behavior at roughly the centerline of the backing plate.

TABLE C.1. Thermophysical Properties for Type 304 Stainless Steel

Room Temperature (20°C)	
Cp	~0.38 J/g-K ⁽¹⁸⁾
K	~0.15 W/cm-K (0.015 J/mm gC)
P	~7.9 x 10 ⁻³ g/mm ³
α	~3.5 mm ² /s
700°C	
Cp	~0.45 J/g-K
K	~0.25 W/cm-K (0.025 J/mm gC)
P	~const.
α	~5.0 mm ² /s
523°C 800K	
P	~const.
Cp	0.41 J/g-K
K	0.0229 J/mm SC
α	4.63 mm ² /s
For T >800°C	
Cp	~0.49 J/g-K = C
P	~const. - 7.9 x 10 ⁻³ g/mm ³
K	0.28 W/cm-K (0.028 J/mm gC) assumed constant
α	- 5 mm ² /s
600°C	
Cp	~0.42 J/g-K
K	~0.23 W/cm-K (0.023 J/mm gC)
P	- const.
α	- 4.75 mm ² /s
800°C	
Cp	~0.49 J/g-K
K	~0.28 W/cm-K (0.028 J/mm gC)
P	- const.
α	- 5 mm ² /s
T _{mp}	- 1370 to 1450°C
T _{mp}	(Aug.) to 1410°C

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