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INTERGRANULAR STRESS-ASSISTED CORROSION CRACKING
OF AUSTENITIC ALLOYS IN WATER-COOLED NUCLEAR REACTORS

by

Craig F. Cheng

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ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

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ABSTRACT

Corrosion-cracking incidents of structural components and fuel-element cladding in 30 water-cooled reactors have been reported. All pressure boundary penetrations have small seeping leaks, but pipe severance has not occurred. Failures of the nonpressurized boundary components have been limited to small fasteners and screws, none of which have constituted potential nuclear safety hazards, directly or indirectly, nor damage involving other components.

The cracking incidents involving Types 304, 304L, and 347 stainless steel, Incoloy 800, and Inconel 600 were attributed to intergranular stress-assisted corrosion cracking associated with (a) oxygen, (b) caustic, or (c) hydrogen. Sensitization increases the susceptibility to cracking in dissolved oxygen, but it is not a factor in caustic solution. Other material conditions, corrective actions, and mechanisms of intergranular stress-assisted corrosion cracking are discussed.

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1. Introduction

Water-cooled nuclear reactors have been a source for generating electric power in commercial central stations for more than a decade. In these reactors, the reactor vessel, vessel components, primary piping, steam generators, and fuel-element cladding are subject to cyclic thermal and mechanical stresses due to the high pressure and temperature of the coolant and fuel during operational cycles. When austenitic alloys are used as reactor materials, intergranular corrosion failures due to the combined action of stress and corrosion occur under specific conditions. However, none of the failures constituted a potential nuclear safety hazard, directly or indirectly, nor did they lead to damage to other components. In the present paper, the cracking incidents involving structural components and fuel-element cladding constructed of Types 304, 304L, and 347 stainless steel (SS), Incoloy 800, and Inconel 600 in 30 water-cooled reactors are discussed. The failure analyses include the (a) operating conditions, (b) fabrication method, (c) cause of failure, and (d) corrective action taken.

The failures are identified as intergranular stress-assisted corrosion cracking (SACC) associated with (a) oxygen, (b) caustic, and (c) hydrogen. Classical stress-corrosion cracking (SCC) is only observed in halide-contaminated water. The mechanism of intergranular SACC is discussed.

*Work supported by the U. S. Atomic Energy Commission.

2. Cracking incidents involving austenitic alloys in water-cooled nuclear reactors

Corrosion-cracking incidents involving Types 304, 304L, and 347 SS, Incoloy 800, and Inconel 600 in water-cooled nuclear reactors may be separated into five groups in terms of the coolant condition. These include structural components in (a) preoperational hydrostatic water, (b) contaminated water, (c) semistagnant primary water, (d) free-flowing primary water, (e) semi-stagnant secondary water, and (f) fuel-element cladding in free-flowing primary water.

2.1. *Structural components in preoperational hydrostatic water*

SACC at the crevices of Type 304 SS water-cooled reactor-vessel components and steam-generator (SG) tubing in preoperational hydrostatic tests in commercial central stations have been reported (table 1). The predominantly intergranular and branching cracks are attributed to the combined effect of sensitization and localized stress in oxygenated water containing some halides and iron oxide.

2.1.1. *New Production Reactor, Richland, Washington, USA*

During hydrostatic testing in August 1963 [1], primary-to-secondary piping leaks were found in one SG in the New Production Reactor (NPR). The cracking was predominantly intergranular, and branching started from the inside diameter (ID) surface. Some cracks near the stainless steel tube-to-tube sheet weld initiated transgranularly.

Stress-relieving the ferritic steel shell at 565-620°C severely sensitized the tubing and produced a thermally stressed region near the tube-to-tube sheet weld. Both factors promoted intergranular SACC. The halides from an acid pickling solution (HF-HNO_3) used in fabrication probably are responsible

for the transgranular cracking. One badly cracked SG was replaced with Inconel 600 tubing. The other defective SG was covered with Incoloy 800 sleeves over the original tube-sheet area by rolling and welding the sleeves to the tube sheet.

2.1.2. *Oyster Creek-1, Forked River, New Jersey, USA*

During the hydrostatic testing of the Oyster Creek-1 Boiling Water Reactor (BWR) in October 1967 [2], intergranular SACC was found at the stub tubes located at the lower head region in the reactor vessel. The stub tubes serve as a transition between the pressure vessel and the tubes that contain the control rods (fig. 1). Subsequent examination revealed that most of the control-rod-drive (CRD) stub tubes exhibited intergranular SACC. The deepest cracks (8 mm) were above the toe of the weld, where high residual stress (beyond initial yield) was measured. Similar cracking was also observed at the safe ends of nine of the ten recirculation nozzles (fig. 2). Also, surface cracks were detected in the vessel cladding and in the head-spray nozzle flanges. The carbon steel reactor vessel was fabricated with Type 304 SS components prior to stress relieving (4 h at 620°C, air cooled). The corrective action taken was to replace or grind off the defective areas on the stub tubes and nozzle safe ends. The fabrication method was also modified. Originally, the Type 304 SS nozzle was welded to a ferritic steel forging with Inconel weld metal, and the entire unit was subsequently stress relieved. The present method of fabrication consists of cladding the transition surface of the safe end with weld-deposit 308L SS, then tying in with an Inconel electrode (InCo 82) to the existing Inconel butt weld, and finally connecting a new field weld of 308L SS to the mating pipe (fig. 2). A similar approach was employed in cladding the outside surface of the stub tube.

The cracking incidents were attributed to the exposure of sensitized SS to either one or a combination of the following factors: (a) oxygen in the test water, (b) chloride pickup in the test water due to the proximity of the reactor to the Atlantic Ocean, (c) fluoride surface contamination, and (d) iron oxide surface contamination. Surface contamination in (c) and (d) occurs during the stress-relieving operation. Fluoride contamination originates from the welding rod coatings. The iron oxide contamination is due to the contact with the carbon steel. In a mock-up fabrication test in 82°C demineralized water, the uncontaminated nozzle safe end exhibited surface cracking when 70-ppm chloride was added. Similar cracking was produced when the nozzle safe end was contaminated with fluoride or iron oxide, with no chloride addition.

2.1.3. *Tarapur, Tarapur, Bombay, India*

The Tarapur BWR Power Station uses sea water for the condenser and operates on a dual-cycle system. During hydrostatic testing in 1967, intergranular SACC was found in the stub tubes, nozzle safe ends, brackets, and vessel cladding [3]. The cause of failure was thought to be similar to that experienced at Oyster Creek-1, and a comparable repair program was instituted.

2.2. *Structural components in contaminated water*

Chloride or fluoride SCC of structural components in the reactor vessel and SG tubing constructed of Types 304 and 347 SS have been reported (table 2). The halides were leached from gaskets, lubricants, pickled surfaces, or faulty demineralizers. Typical transgranular and intergranular SCC cracking was observed. Table 2 summarizes these cracking incidents at Oyster Creek-1 [4], Kahl Karl Würgassen (KKW) [5], Millstone-1 [6], Savannah River [7,8], Agesta [9], and USS Nautilus [10].

2.3. *Structural components in semistagnant primary water*

Intergranular SACC of Type 304 SS reactor-vessel components and small recirculating piping (under 20-cm dia.) in the primary reactor-water circuit has occurred at the semistagnant flow areas (table 3). The combined effects of (a) the relatively higher oxygen content (0.3-3.0 ppm) in the semistagnant water compared with the bulk solution (<0.2 ppm), and (b) the localized stresses beyond initial yield strength are responsible for the cracking of Type 304 SS materials.

2.3.1. *Dresden-1, Morris, Illinois, USA*

Dresden-1, a dual-cycle BWR, was the first full-scale, privately financed nuclear power plant to go into operation (1960) in the United States. Five to ten years later, SACC incidents were reported in the recirculating piping of the secondary SG systems. Some hairline leaking cracks propagated from the ID surface through the pipe wall in and near the heat-affected zone (HAZ) of the weld joints and in the straight-run sections. The defective piping sections included 10-cm cleanout fittings and demineralizer supply lines, 15-cm bypass and equalizer lines, a 7.5-cm steam-vent line, and a 10 to 5-cm steam-vent reducer. The locations of these cracked areas are indicated in chronological order of occurrence in fig. 3 [11]. These branching circumferential cracks were intergranular and exhibited localized ductile tearing. Cracking initiated from the ID surface edge of the disturbed metal layer (machined, ground, or otherwise plastically deformed). Cracks near a weld joint usually started at the nonsensitized area, then propagated cross sectionwise toward the sensitized zone and arrested at the weld metal. The breakthrough to the outside diameter (OD) surface can only be in the single-phase austenite or either an annealed or sensitized area and never at the austenoferritic duplex phase of the weld metal.

Cheng and Potter [11] concluded that, at times, these defective piping sections were exposed to semistagnant water containing relatively higher oxygen content (0.3-3.0 ppm) than under normal reactor operating conditions (<0.2 ppm). A cold-worked layer (32-34 Rockwell C) of disturbed metal is a prerequisite for the initiation of cracking, whereas sensitization is not, but the latter causes greater susceptibility to crack propagation. Figures 4-6 show the ID cold-worked layer at (a) the counterbore-relieved joints produced by grinding prior to welding, and (b) the mandrel sizing marks in the straight-run sections. When cold working was eliminated, no cracking was reported in the replaced piping.

2.3.2. *Elk River Reactor, Elk River, Minnesota, USA*

The Elk River Reactor facility is an indirect-cycle BWR. Unlike most BWR systems, the saturated steam from the heat exchangers is further heated in a fossil-fired superheater before it passes to the turbine. Intergranular SACC was found in the nozzle extension ends of the upper liquid-level line, the core-spray safe ends, and a steam nozzle [12]. All had been furnace sensitized at 621°C prior to welding onto the straight-run field pipe. The cracks initiated near the weldment at the rough-ground ID surface, but not always in the HAZ. These defective areas had been exposed to stagnant steam and/or droplets of steam condensate containing 3 ppm oxygen. Cracking was attributed to the combined effects of localized stress, sensitization, and higher oxygen content in the steam.

2.3.3. *LaCross Boiling Water Reactor, Genoa, Wisconsin, USA*

In October 1969, a leak occurred in the Type 304 SS safe end attached to the feedwater inlet nozzle No. 2 of the LaCross Boiling Water Reactor (LACBWR) [13]. Subsequent ultrasonic inspection and metallurgical examination

of the four feedwater inlet nozzles indicated major cracking on the inside surfaces of safe ends attached to nozzles 1 and 2, but not on the corresponding safe ends attached to nozzles 3 and 4. The branching cracks in the nozzle safe ends were distinctly intergranular and occurred within 1 yr. The four safe ends were furnace sensitized at 621°C. The two cracked safe ends were more severely sensitized than the uncracked safe ends. In addition, the safe end of the emergency core-spray nozzle 1 was as severely sensitized as that of nozzle 2, but less cracking was noted in the former. Furthermore, copper concentrations as high as 44% were found in the crevice area of the cracked safe ends, whereas ~3 to 6% was observed in the crevice areas of the uncracked safe ends. The cause of failure was attributed to the combination of (a) sensitization of the safe end-to-nozzle weld, (b) a corrosion reaction possibly sustained by the reduction of cupric ions in semistagnant water, containing an oxygen concentration of about 0.5 ppm, and (c) tensile stresses in excess of the initial yield strength imposed by a piping support failure early in the operation of the reactor. The piping support was redesigned, and all sensitized nozzle safe ends and thermal sleeves were replaced with Inconel 600.

2.3.4. *Nine Mile Point-1, Seiba, New York, USA*

Nine Mile Point-1 BWR Station suffered an intergranular SACC at both core-spray safe ends after 11 thermal cycles at full-power operation [14]. The Type 304 SS safe ends had been sensitized at 620°C for 10 h during reactor-vessel fabrication prior to welding the nozzles to the outside safeguard spray system. The cracks were both circumferential and longitudinal. They were located between the safeguard spray piping weld and the thermal sleeve weld and propagated from the inner surface to the outer surface (Fig. 7). The massive cracking of the sensitized safe ends was caused by high tensile stress

(90 ksi -- six times the allowable stress at temperature) due to the cantilever effect of the improperly supported attachment piping and the presence of relatively higher oxygenated (>0.3 ppm) semistagnant water in the closed and inverted loop. The situation was resolved by completely replacing both core-spray safe ends, the outboard section of the thermal sleeves, and the adjacent external attachment piping section with annealed Type 304L SS. The two core-spray lines were redesigned to reduce the operating stresses far below the yield point.

2.3.5. *Dodewaard GKN Reactor, near Nijmegen, Netherlands*

The Netherlands BWR reactor at Dodewaard suffered intergranular SACC at the transition surface of the core-spray nozzle safe ends in February 1972. Some hairline cracks were 8 mm long. The cracking was caused by plastic deformation at the counterbore surface as a result of grinding, the formation of oxygen cells from corrosion deposits, and sensitization from stress relieving the reactor vessel after fabrication [15]. The reactor-vessel stress-relieving procedure was similar to other BWRs fabricated in the mid-1960s.

2.3.6. *Argonne Experimental Boiling Water Reactor, Argonne, Illinois, USA*

The experimental boiling water reactor (EBWR) operated by Argonne National Laboratory exhibited intergranular cracking in the stitch-welded cladding of Type 304 SS [16]. The cracks were located in or adjacent to the steam zone and propagated from the inner surface through the outer transition surface but not into the carbon steel vessel. Microexamination revealed the cracked area was severely sensitized. The cracking was probably associated with localized induced stress, sensitization, and oxygen in the steam (3 ppm). The reactor was subsequently decommissioned.

2.3.7. *Shippingport Atomic Power Station, Shippingport, Pennsylvania, USA*

The Shippingport Atomic Power Station is a pressurized water reactor (PWR),

which started operation in 1957, and is a prototype for light-water reactors. Intergranular SACC of a sensitized Type 304 SS canopy seal was reported [17]. The canopy seal was an integral part of a housing, which had been sensitized during the fabrication procedure. Cracking was attributed to the exposure of the highly sensitized canopy seal to oxygen that was entrapped under the seal during filling operations.

2.3.8. *Yankee Nuclear Power Station, Rowe, Massachusetts, USA*

The Yankee (Rowe) PWR Station reported severe intergranular SACC of the Type 304 SS that was welded onto the pressurizer [18]. The cladding was severely sensitized, and the cracking was similar to the EBWR experience. It was concluded that the high oxygen levels in the pressurizer early in life promoted the cracking. Similar spot-welded cladding in the reactor vessel, not exposed to high oxygen conditions, was not cracked.

2.4. *Structural components in free-flowing primary water*

Intergranular SACC of Type 304 SS and Inconel 600 vessel components and auxiliary piping systems has been observed in free-flowing primary water of BWRs (table 4). The components away from the core, in a normal oxygen concentration below 0.3 ppm, failed in a cold-worked plus sensitization condition. Components near the core, with a normal oxygen concentration of more than 3 ppm and a high level of radiation, failed even in the cold-worked condition.

2.4.1. *Big Rock Point, Charlevoix, Michigan, USA*

The Big Rock Point BWR Station reported, in April 1970, intergranular SACC of Type 304 SS cold-headed bolts that were used as fasteners on the top guide assembly in the core area [19]. Cracking was initiated by the high residual stress from cold working, and possibly irradiation might have aided the process. Careful metallographic examination confirmed that sensitization was not involved. Solution-annealed bolts were used as replacements.

2.4.2. *Humbolt Bay Unit #3, Eureka, California, USA*

An inspection in May 1970 at the Humbolt Bay BWR Plant revealed intergranular SACC in a 4-cm Type 304 SS (Schedule 80) instrument line [20]. The cracking initiated from the ID surface longitudinally and circumferentially around a U-bend. Examination of the microstructure showed the ID surface had been cold worked and severely sensitized during the U-bending operation. The defective line was replaced with annealed Type 304 SS.

2.4.3. *Dresden-1*

Intergranular SACC of cold-headed Type 304 SS cap screws in the fuel bundles of the Dresden-1 BWR (Sect. 2.3.1) occurred after 10 years of service [21]. The measured hardness was equivalent to 32 Rockwell C. Apparently severe cold working, and possibly irradiation in the core, initiated the cracking.

2.4.4. *Garigiano, Senn, Italy*

Six years after operation, intergranular SACC of the Inconel 600 non-condensable outlet tube (4-cm dia, Schedule 80) was reported in the Senn BWR [22]. The cracks were near the weld joint. The combination of localized stress from grinding and carbide precipitation in the HAZ is the suspected cause of crack initiation. The defective portion was replaced with an annealed section.

2.5. *Structural components in semistagnant secondary water*

Intergranular caustic cracking occurred in the crevices of the Type 304 SS and Inconel 600 SG tubing in PWR and Inconel 600 inspection tubes in the Heavy Water Reactor (HWR). Residual stress and alkalinizing species in the secondary water were responsible for the cracking (table 5). At present, two major water-treatment concepts are used in modern LWR: pure water or "volatile amine" treatment and phosphate treatment [23].

The volatile amine treatment concept uses vaporizing amines for pH control

throughout the steam cycle and hydrazine for scavenging oxygen. The common amines such as ammonia, morpholine, and cyclohexylamine provide adequate pH control at the low temperatures found in condensers or low-pressure feedwater heaters. Recent experiences have indicated that utilizing only amines for pH control does not permit the buffering capability required to protect against the formation of free caustic on the SGs. Free caustic can enter the SG through condenser leakage or make-up water impurities (see Benzau-1, Sect. 2.5.6).

The phosphate treatment is another method of controlling pH through the buffering action of sodium phosphate and, at the same time, the treatment minimizes the chances of free-caustic formation on the SG. Two types of phosphate treatment exist: coordinated phosphate-pH [24] and congruent [25]. In fig. 8, the coordinated phosphate-pH is regulated by maintaining the water pH at a given phosphate concentration below the curve of pure trisodium phosphate (Na_3PO_4). The congruent method attempts to maintain the molar ratio of sodium-to-phosphate at 2.6 or lower. This approach is more reliable than the coordinated phosphate-pH method, since it has been found [26] that, at 300°C, alkaline phosphate containing a higher (2.8) sodium-to-phosphate ratio can precipitate a salt richer in disodium phosphate than the starting mixture, with a resultant increase in the boiler water alkalinity. The recommended operating range for the congruent method that will ensure the absence of free caustic in SGs is illustrated in fig. 9. However, recent data have been reported on new problems arising from localized wastage in Inconel 600 tubes. The wastage has been attributed to a localized concentration of sodium nickel phosphates and sodium hydrogen nickelites [27]. Consequently, the major U.S. manufacturers of SGs for PWRs are now recommending the volatile amine treatment.

2.5.1. Shippingport Atomic Station

The Shippingport PWR Station (see Sect. 2.3.7) experienced Type 304 SS tube leaks in one of its four horizontal U-tubes after 150 h of operation [28].

The faulty tubes were plugged, and the boiler chemistry, based on the coordinated phosphate-pH control of 50-80 ppm phosphate at pH 10.7-10.9, was changed to 100-300 ppm phosphate at pH 10.6-11.0 to minimize the formation of free caustic. However, subsequent metallurgical examination revealed cracks at or near the tube sheets, which were attributed to caustic cracking. The cracks were intergranular in the sensitized material and transgranular in the annealed material.

2.5.2. *Indian Point-1, Buchanan, New York, USA*

Starting at the end of 1968, numerous leaks of SG tubing have occurred at the Indian Point-1 PWR Station. The coordinated phosphate-pH treatment was used. In one case, leaks were observed in three of four SGs that contained Type 304 SS horizontal U-tubes [29]. The intergranular cracks were located at or near tube sheets and were attributed to caustic cracking.

2.5.3. *Yankee (Rowe)*

The Yankee (Rowe) PWR plant (see Sect. 2.3.8) experienced leaks in each of four Type 304 SS SGs. The leaks occurred within 1.5 cm of the upper surface of the tube sheet. These leaks started early in 1966, and 12 tubes were repaired in March 1967. A few additional leaks occurred in 1969. The cracking was probably caused by free caustic; therefore, the secondary-water chemistry was changed from the phosphate to the volatile amine treatment with morpholine and hydrazine [29,30].

2.5.4. *Connecticut Yankee, Haddam Neck, Connecticut, USA*

The Connecticut Yankee PWR Station experienced leaking or cracking of Inconel 600 SG tubing in eight SGs during 1970-1972, 3-5 yr after starting operation [21,30]. One tube failed completely, whereas in another the crack penetrated 98% of the wall. The cracks were intergranular and were located in a narrow band 3-15 cm above the tube sheet on the inlet (hot-leg) side of the tube bundle (fig. 10) [23]. The cracking was attributed to free caustic. No cracking has been reported since the changeover to the congruent

phosphate control in the operating range recommended by Westinghouse [23] in March 1973 (fig. 9).

2.5.5. *San Onofre, San Clemente, California, USA*

The San Onofre PWR plant experienced leaks in the Inconel 600 SG at the U-bends and support baffles. The leaks were located by eddy-current techniques (fig. 10) [23]. The leaking tubes were plugged. Cracking associated with free caustic was suspected, and the congruent phosphate control was instituted.

2.5.6. *Beznau-1 NOK, Dettmolding, Switzerland*

The Beznau-1 PWR plant suffered major intergranular caustic cracking of Inconel 600 SG tubing from the secondary side in 2 yr of operation [21,23]. Some 500 tubes had to be plugged because of leaks and partial wall penetrations during two shutdowns in August and December 1971. The earlier defects in the first shutdown were located in the hot end of the tube-to-tube sheet crevice region (fig. 10). An investigation showed that the volatile amine treatment used in the secondary water provided no buffering or protective action to cope with the impurities introduced into the SGs by condenser leakage. The SG tubes developed a hard scale, largely composed of caustic impurities (NaOH or KOH) found in the river.

Intermittent addition of phosphate to the boiler was initiated in October 1971, when the station resumed operation. The condenser leakage was at a higher rate than anticipated; therefore, the caustic impurities in the river water depleted the phosphate content in the secondary water. Consequently, unbuffered (free) caustic was deposited on the tubing, which caused cracking 3-9 cm above the tube-to-tube sheet (fig. 10). After the introduction of continuous congruent phosphate control of the boiler water together with the continuous blowdown of the SG to reduce impurities, no tube-leakage problem

has developed since the plant returned to operation in the winter of 1972. The sister plant, Beznau-2, started operation in October 1971 at the same site. No leakage of SG tubing has occurred using continuous congruent phosphate control and blowdown.

2.5.7. *H. B. Robinson-2, Hartsville, South Carolina, USA*

The H. B. Robinson-2 PWR plant reported severe leaks in Inconel 600 SG tubing within 1 to 2 yr of operation [23,31]. The slug-feeding method of phosphate addition used since the initial operation in 1970 promoted erratic chemistry control with wide fluctuations in chemistry parameters, particularly when substantial impurities from condenser leakage occurred. The defects with intergranular cracks were located several centimeters from the tube sheet and were induced by caustic deposits (fig. 10). The plant resumed operation in June 1972 after the defective SG tubes were plugged. No tube leakage has been reported since the unit returned to service using continuous congruent phosphate control coupled with continuous blowdown.

2.5.8. *Point Beach-1, Two Creeks, Wisconsin, USA*

Leakage of Inconel 600 tubing in the SG has been reported in the Point Beach-1 PWR plant [23,30]. A number of tubes, located several centimeters above the tube-to-tube sheets, were cracked (fig. 10). In some, but not all cases, the leakage in the SG was identified with intergranular cracking. The presence of free caustic was indicated by poor control of the sodium-to-phosphate ratio when the slug-feed system was employed during the first 13 months of plant operation. No increase in leakage rate was noted in the next eight months, when the phosphate concentration in the slug-feed system was increased, with periodic blowdown, to remove solids. The plant has since returned to operation after retubing in March 1973 using the continuous congruent phosphate treatment with blowdown.

2.5.9. *Obrigheim KWO, Baden Würt, West Germany*

At the Obrigheim PWR Station, substantial leakage and cracks near the apex of a cold-worked U-bend area and near the tube-to-tube sheet were reported in Inconel 600 SG tubing in October 1971 and March 1972 [23,30]. The cracking was intergranular. Ammonia rather than an oxygen-scavenging amine was used to maintain the pH at 9.2-9.5. Also, condenser leakage was indicated by the calcium deposits on the SG tubing. Cracking is associated with high stress (greater than two times the initial yield strength) and unknown causative factors. Detailed information is not available.

2.5.10. *Ågesta, Stockholm, Sweden*

In the Ågesta HWR system, three Inconel 600 inspection tubes cracked at the OD surface. The intergranular cracking occurred after exposure to 217°C secondary water that had been alkalized to 10.0-10.8 pH with lithium hydroxide for 150 days followed by potassium hydroxide for a few days. The oxygen and chloride in the bulk solution were low (<0.2 ppm) [32]. The cracks were located in the crevices between the vessel lid and the tubes. The other factors that are thought to have contributed to the cracking include (a) residual tensile stresses approached the yield strength as a result of shop welding, (b) a relatively high carbon content of the alloy (0.08%) gave rise to heavy precipitation of carbides during subsequent overheating after welding, and (c) occasional local boiling of light water resulted in a concentration of hydroxides at the gas-liquid interface in the crevices.

2.6. *Fuel-element cladding in free-flowing primary water*

During the period 1960-1967, several austenitic alloys, less expensive than Circaloy, were investigated as candidates for fuel-element cladding in light-water reactors (BWR and PWR). However, this approach was unsuccessful,

primarily because of intergranular cracking (table 6) [33]. Fuel-element cladding of Types 304, 304L, and 347 SS, Incoloy 800, and Inconel 600 were tested in both the annealed and cold-rolled conditions. Localized stress and irradiation effects in conjunction with (a) oxygen in the BWR, and (b) hydrogen in the PWR resulted in intergranular cracking of the cladding.

2.6.1. *Experimental BWRs in the USA*

Type 304 SS cladding was tested in a government-sponsored program on the irradiation of fuel at high power densities at the Vallecitos Boiling Water Reactor (VBWR) [34]. The cladding exhibited intergranular cracking, which originated on the outer surface in 500 to 1000 h. The cracking patterns varied with the initial condition of the tubing. Cracks in the cold-worked tubing were, for the most part, longitudinal, whereas those in the annealed condition were mostly circumferential. However, no grain-boundary sensitization was found. It was concluded that irradiation increases the susceptibility to intergranular attack, but no significant difference in the failure rate was evident between solution-annealed and cold-worked cladding. The failure was associated with SACC in the presence of (a) oxygen from radiolytic decomposition of water in the BWR, and (b) irradiation corrosion in the core.

2.6.2. *Commercial BWR Stations in the USA*

High-density fuel-element cladding of cold-rolled Type 304 SS suffered intergranular SACC similar to that in VBWR, Dresden-1, Big Rock Point, and Humbolt Bay [35,36]. In addition, cladding of cold-rolled Inconel 600 and Incoloy 800 suffered gross intergranular cracking in fuel-element tests at Big Rock Point [37]. Microexamination showed neither sensitization nor excessive plastic deformation had occurred.

2.6.3. PWR Test Loops in the USA

One experiment conducted in the Engineering Test Reactor (ETR) loop was prematurely terminated because of the excessive release of fission products [38]. It was determined that a pump failure with subsequent loss of coolant was responsible for the severe thermal conditions that cracked the cladding of a fuel rod and produced the "fission break." The fuel rods had operated for 76 days in 600°F lithiated water at pH 10 and at a peak surface heat flux of 93 W/cm^2 ($296,000 \text{ Btu/h-ft}^2$). The UO_2 pellets, clad in 304L stainless steel tubing, suffered intergranular cracking. The severe thermal condition also changed the cladding from a solution-annealed to a sensitized condition. The failure was attributed to stress rupture in conjunction with hydrogen embrittlement under the severe thermal conditions. The hydrogen was generated by the reaction of the UO_2 fuel and water.

Likewise, fairly extensive longitudinal intergranular cracking occurred in Type 347 SS cladding of the PM-3A Reactor in the region of the core where the cladding stresses and fuel burnups were maximum [39]. Cladding surface temperatures in this region may have been sufficiently high to produce hydrogen embrittlement through boiling.

Intergranular cracking of vacuum-annealed 304L SS cladding on poison elements was observed in the West Milton PWR test loop at Knolls Atomic Power Laboratory [40]. The failure occurred during irradiation in 600°F ammoniated water at pH 10 when the cladding was stressed in excess of the yield point. The hydrogen content of the irradiated cladding increased from 4-6 to 8-14 ppm in 700 h at effective full power. Stress-rupture cracking aided by hydrogen was indicated.

3. Discussion

All cracking incidents of austenitic alloys, Types 304, 304L, and

347 SS, Incoloy 800, and Inconel 600, in water-cooled reactors, as listed in tables 1 through 6, are associated with (a) oxygen, (b) free caustic, (c) hydrogen, and (d) halides as the major causative factors under high stress. The cracking incidents in (a)-(c) were predominantly intergranular, whereas, in (d), the cracking was a mixed mode (transgranular and intergranular). The composition at the grain boundaries may be quite different from the bulk grains. If the corrosive attack prefers the grain boundary, the mode of cracking is intergranular; however, if the attack selects the grain, the mode is transgranular. Factors (a)-(c) are associated with intergranular SACC, and its mechanism will be discussed. The halide cracking mechanism (SCC) has been extensively reviewed elsewhere [6].

3.1. *Mechanism of intergranular SACC*

The mechanism of intergranular SACC is not clearly understood. The cracking, in the water-cooled reactors discussed above, occurs in two stages, initiation and propagation. The initiation stage requires excessive stress (beyond initial yield strength) concentrated at surface imperfections to facilitate an initial site for preferential tunneling or grain-boundary attack. The propagation stage proceeds along the grain boundary in two steps: metallic dissolution, oxidation or hydrogen absorption at the crack tip until the process is stilled by mass-transport or diffusion limitations; and mechanical tearing at the yielding crack tip by localized stresses. These stresses are momentarily built up by internally or externally applied loads that cause the localized stresses to exceed the yield strength at each burst cycle. The process repeats in alternating steps. The corrosive action is attributed to oxygen, caustic, hydrogen, carbide precipitation and irradiation.

3.1.1.1. *Oxygen*

In the first step of crack propagation, dissolved oxygen promotes the metallic dissolution, or the oxidation at the crack tip. In metallic dissolution, a potential difference develops between the oxygen-depleted solution at the crack tip and the relatively oxygen-rich bulk coolant. As a result of the hydrolysis of the cations, the solution in the crevice increases in acidity until the reaction is stifled by mass-transport limitations. This behavior applies to austenitic stainless steel and Incoloy 800. The intergranular oxidation by diffusion control is preferred for Inconel 600 [41].

In the second step, mechanical tearing, the rate of propagation depends on the magnitude of the localized stresses at the crack tip. If the stresses are dependent on the wedging action, the propagation rate is expected to be slow since the stresses must be rebuilt after each burst cycle. Dresden-1 piping (see Sect. 2.3.1) exhibited localized ductile fracture in this manner (fig. 11). Crack propagation by the wedging action of oxides without externally applied loads or residual stresses has been demonstrated on Type 304 SS [42] and Inconel 600 [43]. If externally applied stresses are added, the rate is much faster. The fast brittle fracture of the core-spray-nozzle safe ends (fig 12) in Nine Mile Point-1 (Sect. 2.3.4) was attributed to externally applied stresses (~90 ksi).

The amount of oxygen required for intergranular SACC is not clearly defined. However, the lack of similar intergranular cracking in FWR primary water (<0.1 ppm oxygen) suggests the minimum concentration is probably above 0.1 ppm. According to Clarke and Gordon [44], a definite oxygen acceleration exists between 0.2 and 3-5 ppm in sensitized Type 304 SS. The degree of

acceleration between 0.2 and 3-5 ppm oxygen is considerable; 1.5 to 2 orders of magnitude at a stress level of about twice the yield strength. Between 5 and 100 ppm, little additional acceleration is due to oxygen. Data in table 7 indicate that both Type 304 SS and Inconel 600 are susceptible to intergranular SACC in oxygenated water, particularly when the materials are sensitized. In the case of Type 304 SS, the susceptibility is also enhanced by surface cold working (<10%) in which induced martensite platelets accelerate the oxide film growth and subsequent film rupture (Dresden-1 Sect. 2.3.1 and fig. 7). Normally, annealed Type 304 SS is satisfactory in the medium-range oxygen (0.3-3 ppm), cold-worked material is more resistant than the annealed, and the weld metal with an austenoferritic duplex phase appears immune to cracking (Dresden-1, Sect. 2.3.1 and fig. 4). Laboratory data confirmed the observation for Type 304 SS [44] and Inconel 600 [42]. At high-range oxygen (3-8 ppm) plus irradiation, the three alloys failed in the cold-worked condition. It has been observed [45] that irradiation can greatly increase the rate of grain-boundary corrosion in fuel-element cladding. In general, the susceptibility to intergranular SACC in dissolved oxygen decreases in the following order: Type 304 SS, Incoloy 800, and Inconel 600 [44].

3.1.2. Caustic

In the first step of propagation, caustic solution promotes metallic dissolution. This is related to the solubility of the film as dihydroferrite (HFeO_2^-) species for Type 304 SS and as dinickelite (HNiO_2^-) for Inconel 600. In the second step, the stress required for mechanical fracture at the yielding crack tip must be externally applied. The rate of propagation depends on

the magnitude of the external stress and the time required to build up a new protective film after each burst cycle.

As in the case of dissolved oxygen, the susceptibility to intergranular caustic cracking decreases in the following order: Type 304, Incoloy 800, and Inconel 600 [22]. However, unlike dissolved oxygen, sensitization per se does increase the susceptibility to cracking [23,46]. It should be pointed out that Type 304 SS exhibits intergranular cracking in 10% sodium hydroxide at 600°F and transgranular cracking when the concentration is increased to 50% [46], whereas Inconel 600 exhibits only intergranular cracking in caustic solutions. All SG tubing failures of these materials (table 5) are intergranular, except for the annealed Type 304 SS tubing cracks in the Shippingport SG. The reason for the transgranular cracking in the latter case may be the higher concentration of caustic in the crevice area.

3.1.3. *Hydrogen*

In the first step of propagation, hydrogen diffuses into the lattice and migrates to the grain boundary ahead of the crack tip. This agrees with the experiment conducted by Gilpin et al. [47]. They showed that radioactive hydrogen (tritium) does migrate to the regions of maximum triaxial stresses ahead of a notch. In the second step, mechanical fracture at the yielding crack tip is due to the buildup of hydrogen pressure that momentarily exceeds the yield strength and then drops after each burst cycle. The process repeats as more hydrogen diffuses into the metal and collects at the grain boundaries, as illustrated by Beachem [48]. The rate of propagation depends on the hydrogen pressure plus externally applied stresses, if any, and the time required to concentrate the hydrogen at the grain boundary.

In general, austenitic stainless steel fuel-element cladding has performed well in PWR without evidence of cracking. The Type 304 SS in the

Yankee (Rowe) Reactor and Type 304 SS in the Saxton Reactor performed well [49]. Both fuel elements were at a high neutron dose and had a longer burnup than the elements in the three PWRs (EBR, PM-3A, and West Milton), as indicated in table 6. Apparently, the cladding-surface temperature was sufficient to produce boiling and thereby more hydrogen was charged into the cladding.

4. Conclusions

Corrosion-cracking incidents of structural components and fuel-element cladding in 30 water-cooled nuclear reactors have been reported. All pressure-boundary penetrations detected have been small seeping leaks, but pipe severance has not occurred. Failures of nonpressurized boundary components have been limited to small fasteners and screws, none of which have constituted potential nuclear safety hazards, directly or indirectly, nor damage involving other components. The cracking of Types 304, 304L, and 347 SS, Incoloy 800 and Inconel 600 was attributed to intergranular SACC associated with oxygen, caustic, or hydrogen. Also, in the case of austenitic stainless steel, halide SCC was observed in contaminated water. The susceptibility of intergranular SACC decreases in the following order: austenitic SS, Incoloy 800, and Inconel 600. Sensitization promotes cracking in primary water containing dissolved oxygen (>0.1 ppm), but it is not a factor that relates to the free caustic in the secondary SG.

The mechanism of intergranular SACC occurs in two stages, initiation and propagation. The initiation stage requires stresses, above the initial yield stress, concentrated at surface imperfections. The propagation stage proceeds along the grain boundary in alternate steps (a) metallic dissolution, oxidation or hydrogen absorption at the crack tip, and (b) mechanical tearing at the yielding crack tip, momentarily caused by stress beyond yield at each burst cycle. The process then repeats.

The history of cracking incidents has led to changes in fabrication procedures, material, and overall component design. Great care is taken to (a) prevent the formation of a cold-worked surface and avoid subsequent sensitization during fabrication, and (b) ensure that loads due to applied or residual stresses are kept below the yield stress in BWR structural components. The caustic cracking problems in PWR SGs may be resolved by the combination of the (a) selection of the more resistant Inconel 600, (b) reduction of the possibility of caustic carry over in secondary water, and (c) provision for continuous blowdown of the SGs. Also, the recent chloride-intrusion incident in a BWR led to the lowering of the chloride and conductivity operating limits in the coolant water to increase the margin of safety.

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Table 1
Cracking of structural components in preoperational hydrostatic water.

Nuclear station; Date of operation	Date of incident	Defective component			Possible cause of failure	Corrective action
		Description	Alloy	Failure ^a		
Hanford NFR, Richland, Washington, USA, GWR - 1966 (Jan) 790 MW(e) net	1963 (Aug)	SG tubing	Type 304 SS	Circumferential, mostly IC	Sensitization residual stress, oxygen (3 ppm), halides	Retubed with Inconel 600; welded Incoloy 800 sleeve over tube-to-tube sheet
Oyster Creek-1, Forked River, New Jersey, USA, BWR - 1969 (Aug) 560 MW(e) net	1967 (Oct)	Stub tubes; nozzle safe ends and flanges; vessel cladding	Type 304 SS	Severe at crevices, mostly IC	Sensitization residual stress, oxygen (3 ppm), halides, iron oxide	Replaced cracked parts or removed cracks; buttering transition with Inco 62; weld SS component to buttered region
Tarapur-1, Tarapur, Bombay, India, BWR - 1969 (Oct) 190 MW(e) net	1968	Stub tubes near welds; nozzle safe ends; brackets; vessel cladding	Type 304 SS	Severe at crevices, mostly IC	Sensitization residual stress, oxygen (3 ppm), halides, iron oxide	Replaced cracked parts or removed cracks; buttering transition with Inco 62; weld SS component to buttered region

^a IC denotes intergranular crack.

Table 2
Cracking of structural component in contaminated water.

Nuclear station; Date of operation	Date of incident	Defective component			Possible cause of failure	Corrective action
		Description	Alloy	Failure ^a		
Oyster Creek-1, Forked River, New Jersey, USA, BWR - 1969 (Aug) 560 MW(e) net	1971 (Oct)	Main steam-valve bushings	Type 304 SS	Circumferential and longitudinal, annealed, TC	Chloride source not identified	Washed valves; re- placed cracked bushings; added drain holes; used chloride-free lubricant
Kahl Karl, Würgassen (KKW), West Germany, BWR - 1962 (June) 15 MW(e) net	1969 (outage)	Control-rod ex- tension shaft	Type 304 SS	Circumferential at HAZ, longi- tudinal at cold work, TC	Residual chloride from previous chemical treat- ment	Replaced cracked shaft
Millstone-1, Waterford, Connecticut, USA, BWR - 1971 (Apr) 652 MW(e) net	1972 (Sept)	Lower power radi- ation monitors; sheaths, handles, holder; control- rod cladding	Type 304 SS	Annealed, IC cold work, TC	Chloride from condenser sea- water leak	Replaced cracked parts; lower limits for chloride and conductivity in reactor water
Savannah River, Savannah, Georgia, USA, HWR - 1955 Plutonium plant	1960	Heat-exchanger tubing (OD); outlet nozzles	Type 304 SS	Tubing, annealed, TC; nozzle, TC and IC	Leachable chlorides in corrosion pro- ducts, fluorides from previous HF-HNO ₃ pickling	Replaced cracked parts; removed halide source
Agesta, Stockholm, Sweden, HWR - 1964 (Mar) 10 MW(e) net	1964	Drain tube	Type 304 SS	Annealed, TC	Chloride from fleximetallic gasket	Replaced cracked drain tube; changed to low chloride gasket
USS Nautilus, U.S. Navy, PWR	-	Steam generator	Type 347 SS	Inner tube sheet and adjacent tube area, an- nealed, TC	Chloride seeped through from secondary water	Replaced with In- conel 600 heat exchanger

^aTC and IC denote transgranular and intergranular cracks, respectively.

Table 3
Cracking of structural components in semistagnant primary water.

Nuclear station; Date of operation	Date of incident	Defective component			Possible cause of failure	Corrective action
		Description	Alloy	Failure ^a		
Dresden-1, Morris, Illinois, USA, BWR - 1960 (Aug) 250 MW(e) net	1965 (Dec) to 1972 (Jan)	10-cm cleanout fitting and de- mineralizer supply line; 15-cm bypass lines; 7.5 cm steam-vent lines; 10 to 5-cm steam-vent reducer	Type 304 SS	Cold-worked area near weld joints and straight runs, circumfer- ential, IC	Localized stress and oxygen (0.3- 3 ppm)	Replaced defective parts; eliminated cold working to re- duce localized stress
Elk River, Elk River, Minnesota, USA, BWR-1964 (Aug) 22 MW(e) net	1968 (Jan)	Nozzle extension end and safe ends, steam nozzle	Type 304 SS	Cold-worked area near weld joints, IC	Localized stress, sensitization, and oxygen (3 ppm) in steam	(Reactor operation terminated)
LaCrosse BWR, Genoa, Wisconsin, USA, BWR - 1964 (Aug) 50 MW(e) net	1969 (Oct)	Feedwater inlet- nozzle safe ends	Type 304 SS	Near weld joints, IC	Applied stress (>yield stress), sensitization, copper deposits, and oxygen (>0.3 ppm)	Reduced applied stress; replaced safe ends and thermal sleeves with Inconel 600
Nine Mile Point-1, Scitoba, New York, USA, BWR-1970 (Jan) 500 MW(e) net	1970 (Mar)	Core-spray- nozzle safe ends	Type 304 SS	Near weld joints, IC	Applied stress (>yield stress), sensitization, and oxygen (>0.3 ppm)	Reduced stress; replaced defective parts with annealed Type 304L SS
Bolswaard GKN, near Nijmegen, Netherlands, BWR-1969 (Mar) 54.5 MW(e) net	1972 (Feb)	Core-spray- nozzle safe ends	Type 304 SS	Near weld joints, IC	Sensitization and oxygen (>0.3 ppm), other details not available	Details not avail- able
Argonne EBWR, Argonne, Illinois, USA, BWR - 1957 (Feb) 4.5 MW(e) net	1967 (July)	Spot welded cladding	Type 304 SS	Near welded seam, IC	Residual stress, sensitization, and oxygen in steam (3 ppm)	Reactor decom- missioned

Table 3
Cracking of structural components in semistagnant primary water (contd.).

Nuclear station: Date of operation	Date of incident	Defective component			Possible cause of failure	Corrective action
		Description	Alloy	Failure ^a		
Shippingport, Shippingport, Penn- sylvania, USA, PWR - 1957 (Dec) 150 MW(e) net	-	Canopy seal cladding	Type 304 SS	Near welded joints, IC	Residual stress, sensitization, and entrapped oxygen (1 ppm)	Change filling operation pro- cedure
Yankee (Rowe), Rowe, Massachusetts, USA, PWR - 1960 (Aug) 125 MW(e) net	-	Spot welded cladding	Type 304 SS	Near welded joints, IC	Residual stress, sensitization, and entrapped oxygen (1 ppm)	Change filling operation pro- cedure

^a IC denotes intergranular cracks.

Table 4
Cracking of structural components in free-flowing primary water.

Nuclear station; Date of operation	Date of incident	Defective component			Possible cause of failure	Corrective action
		Description	Alloy	Failure ^a		
Big Rock Point, Charlevoix, Michigan, USA, BWR - 1962 (Dec) 71.3 MW(e) net	1970 (Apr)	Top guide bolts (core area)	Type 304 SS	Cold headed, IC	Residual stress, cold working, and oxygen (> 3 ppm), and irradiation	Replaced with annealed bolts
Humbolt Bay, Eureka, Cali- fornia, USA, BWR - 1963 (Aug) 63 MW(e) net	1970 (May)	3.8 cm instru- ment line at U- bend area	Type 304 SS	Longitudinal and circumfer- ential, IC at ID cold-worked surface	Residual stress, cold working plus sensitization, and oxygen (< 0.3 ppm)	Replaced with annealed line
Dresden-1, Morris, Illinois, USA, BWR - 1960 (Aug) 200 MW(e) net	1970 (Mar)	Fuel-bundle cap screws, 32 Rock- well C (core area)	Type 304 SS	Cold headed, IC	Cold working, re- sidual stress, and oxygen (> 3 ppm) and irradiation	Replaced with annealed screws
Garigiano, Senn, Italy, BWR - 1964 160 MW(e) net	1970 (Apr)	3.8 cm non- condensable out- let tubing	Inconel 600	ID cold-worked surface (near weld), IC	Localized stress, cold working plus sensitization, and oxygen (< 0.3 ppm)	Replaced with new annealed section

^a IC denotes intergranular cracks.

Table 5

Cracking of structural components in semistagnant flowing secondary water (contd.).

Nuclear station; Date of operation	Date of incident	Defective component			Possible cause of failure	Corrective action
		Description	Alloy	Failure ^a		
HE Robinson-2, Hartsville, South Carolina, USA, PWR - 1970 (Sept) 740 MW(e) net	1971 (Aug)- 1972 (May)	Steam generator tubing, vertical	Inconel 600	Few inches above tube sheet, IC	Free caustic and localized stress	Plugged cracked tubes; changed to continuous congruent phos- phate treatment and blowdown ↓
Point Beach-1, Two Creeks, Wisconsin, USA, PWR - 1970 (Nov) 740 MW(e) net	1971 - 1972	Steam generator tubing, vertical	Inconel 600	Few inches above tube sheet, IC	Free caustic and localized stress	
Obrigheim KW0, Baden Württ, West Germany, PWR - 1969 (Mar) 320 MW(e) net	1971 (Oct) 1972 (Mar)	Steam generator tubing, vertical	Inconel 600	U-bend and near tube sheet	Metallurgical ex- amination made; details not available	Plugged cracked tubes; other details not available
Agesta, Stockholm, Sweden, BWR - 1964 (Mar) 10 MW(e) net	1964 (Aug)	Inspection tube	Inconel 600	Top tube crevice area, IC	Residual stress, heavy carbide precipitation, free caustic, and oxygen	Replaced cracked tube

^aIC and TC denote intergranular and transgranular cracks, respectively.

Table 5
Cracking of structural components in semistagnant secondary water.

Nuclear station; Date of operation	Date of incident	Defective component			Possible cause of failure	Corrective action
		Description	Alloy	Failure ^a		
Shippingport, Shippingport, Penn- sylvania, USA, PWR - 1957 (Dec) 150 MW(e) net	1958 (Jan)	Steam generator tubing, horizon- tal	Type 304 SS	At or near tube sheet, sensi- tized IC, an- nealed TC	Free caustic and localized stress	Adjusted pH/PO ₄ ratio
Indian Point-1, Buchanan, New York, USA, PWR - 1962 (Oct) 237 MW(e) net	1968 (Oct)	Steam generator, tubing, horizon- tal	Type 304 SS	At or near tube sheet, IC	Free caustic and localized stress	Plugged cracked tubes
Yankee, Rowe, Massachu- setts, USA, PWR - 1961 (July) 175 MW(e) net	1966- 1969	Steam generator tubing, vertical	Type 304 SS	1-1/2 in. above tube sheet, IC	Free caustic and localized stress	Plugged cracked tubes; changed to zero solid treat- ment
Connecticut-Yankee, Haddam, Connecti- cut, USA, PWR - 1967 (July) 125 MW(e) net	1970- 1972	Steam generator tubing, vertical	Inconel 600	1-6 in. above tube sheet, IC	Free caustic and localized stress	Plugged cracked tubes; changed to continuous concurrent phosphate treatment and blowdown
San Onofre, San Clemente, California, USA, PWR - 1968 (Jan) 430 MW(e) net	1970	Steam generator tubing, vertical	Inconel 600	Eddy-current in- dications, U- bends and support baffles	Free caustic and localized stress	↓
Beznau-1 NOK, Doettingen, Switzerland, PWR - 1969 (June) 350 MW(e) net	1971 (Aug- Dec)	Steam generator tubing, vertical	Inconel 600	Tube-sheet cre- vice, 1-6 in. above tube sheet, IC	Free caustic and localized stress	

Table 6

Cracking of fuel-element cladding in free-flowing primary water.

Reactor		Cladding alloy		Possible cause of failure
Name	Type	Name	Condition	
VBWR	BWR	Type 304 SS	Annealed Cold rolled	Oxygen (8 ppm.), stress, and irradiation
Dresden-1	BWR	Type 304 SS	Cold rolled	
Humbolt Bay	BWR	Type 304 SS	Cold rolled	
Big Rock Point	BWR	Type 304 SS	Cold rolled	
		Inconel 600 Incoloy 800		
ETR	PWR	Type 304L SS	Annealed	Hydrogen, stress, and irradiation
PM-3A	PWR	Type 347 SS	Annealed	
West Milton (KAPL)	PWR	Type 304L SS	Annealed	

Table 7
Intergranular stress-assisted corrosion cracking of austenitic alloys in water-cooled reactors.

Cracking component		Part name	Reactor involved	Reference	
Material	Condition ^a			Table no.	Section no.
<u>Low-range oxygen (< 0.3 ppm) -- Normal BWR primary water</u>					
Type 304 SS	CW & Sen	U-bend tube (3.8 cm)	Humbolt Bay	4	2.4.2
Inconel 600	CW & Sen	Tube (3.8 cm near weld)	Garigiano	4	2.4.4
<u>Medium-range oxygen (0.3-3 ppm) -- Semistagnant primary water and hydrostatic water</u>					
Type 304 SS	Ann. plus CW surface	Piping (<20 cm) near weld and straight run	Dresden-1	3	2.3.1
		Nozzle safe end, nozzle	Elk River		2.3.2
		Nozzle safe end	LaCross		2.3.3
		Nozzle safe end	Nine Mile Point		2.3.4
		Nozzle safe end	Dodewaard		2.3.5
	Sen		EBWR		2.3.6
		Transition cladding surface	Shippingport Yankee (Rowe)	1	2.3.7 2.3.8
		SG tubing	Hanford NPR		2.1.1
		Vessel internal	Oyster Creek-1	1	2.1.2
		crevices	Tarapur-1		2.1.3
<u>High-range oxygen (3-8 ppm) -- Core area (irradiation)</u>					
Type 304	CW	Top guide bolt	Big Rock Point	4	2.4.1
	CW	Fuel-bundle cap screw	Dresden-1		2.4.3
	CW		Dresden-1		2.6.2
	CW		Humbolt Bay		2.6.2
	CW		Big Rock Point	6	2.6.2
	CW and Ann.	Fuel-element cladding	VBWR		2.6.1
Incoloy 800	CW		Big Rock Point		2.6.2
Inconel 600	CW		Big Rock Point		2.1.2

^aCW, Sen, and Ann. denote cold worked, sensitized, and annealed, respectively.

Fig. 1. Stub-tube configuration in Oyster Creek-1.
Neg. No. MSD-60441.

Fig. 2. Typical repair of nozzle safe end in Oyster Creek-1. Neg. No. MSD-60439

Fig. 3. Cracking incidents in Dresden-1. Inspection: 1, December 1965; 2, February 1966; 3, April 1966; 4, 5 and 6, March 1967; 7, November 1970; 8, February 1971; and 9, January 1972. Neg. No. MSD-58653.

Fig. 4. Micrographs of cracking at inside diameter surface of 15-cm bypass line in Dresden-1. (a) Photograph of circumferential crack in counterbore relief. (b) Enlargement of crack in (a), as-polished. (c) Micrograph showing the arrest of longitudinal side crack in the weldment, 10% electrolytic oxalic. (d) Electron micrograph of replica from fracture of leaking crack in (b) showing discontinuous carbide precipitation. Neg. No. MSD-58699.

Fig. 5. Micrographs showing cold-worked layer at inside diameter surface of counterbore relief area in Dresden-1 piping. (a) and (b) 10-cm cleanout fitting (circumferential). (c) and (d) 10-cm supply line (longitudinal). (a) and (c) annealed; (b) and (d) sensitized, 10% electrolytic oxalic. Neg. No. MSD-58703.

Fig. 6. Micrographs showing cracking at inside diameter surface of 15-cm equalizer line in Dresden-1. (a) Macrograph of circumferential crack adjacent to pipe-to-elbow weld. Mag. $\sim 6\times$, as-polished. (b) Micrograph of intergranular cracking at heat-affected zone of rectangular area in (a); 10% electrolytic oxalic. (c) Macrograph of circumferential crack and mandrel marks in straight-run section, as-received. (d) Micrograph showing intergranular branching cracks and cold-worked surface of straight-run section; 10% electrolytic oxalic.

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Fig. 7. Schematic showing safe-end configuration in Nine Mile Point-1. Neg. No. MSD-60443.

Fig. 8. Boiler water pH vs PO₄ control to avoid free caustic.
Neg. No. MSD-60440.

Fig. 9. pH vs orthophosphate solutions with various sodium-to-phosphate mole ratios. Neg. No. MSD-60442.

Fig. 10. Vertical locations of cracks in SG tube. Neg. No. MSD-60438.

Fig. 11. Scanning-electron micrographs of fracture surface showing predominantly intergranular, occasionally transgranular (arrows), and localized ductile fracture in Dresden-1. (a) and (b) Crack surface of 10-cm cleanout fitting. (c) and (d) Crack surface of 10-cm supply line. Neg. No. MSD-58702.

Fig. 12. Scanning-electron micrographs of fracture surface of nozzle safe end in Nine Mile Point-1. Neg. No. MSD-58797.