M3LW-23OR0402039

Light Water Reactor Sustainability Program

Preparation for Stress Corrosion Crack Initiation Testing of Austenitic Stainless Steels in PWR Primary Water

June 2023

U.S. Department of Energy
Office of Nuclear Energy

M3LW-23OR0402039

Preparation for Stress Corrosion Crack Initiation Testing of Austenitic Stainless Steels in PWR Primary Water

Ziqing Zhai

June 2023

Prepared for the
U.S. Department of Energy
Office of Nuclear Energy

ABSTRACT

Operating experience of Type 304(L)/316(L) austenitic stainless steels (SS) in PWR primary circuits have generally been excellent, but increasing intergranular stress corrosion cracking (IGSCC) incidents have been reported in free-flowing PWR primary water in recent years, posing a potentially serious emerging issue affecting nuclear power plants availability. A better understanding of the IGSCC initiation mechanism and its threats to plants are required to inform utility and regulatory on a proactive management strategy. In preparation for devising a detailed testing plan to address this need, this report reviews available field experience and laboratory studies on SCC initiation of austenitic SS in normal PWR primary water environments. Knowledge and technical gaps are identified, and a near-term action plan is proposed.

ACKNOWLEDGEMENTS

The author gratefully acknowledges the financial support from the Office of Nuclear Energy, U.S. Department of Energy, through the Light Water Reactor Sustainability Program Materials Research Pathway. The author would also like to acknowledge Dr. Mychailo Toloczko from Pacific Northwest National Laboratory, Dr. Peter Andresen (formerly with GE Global Research Center and currently with Andresen Consulting), Dr. Tyler Moss from Naval National Laboratory, and Mr. Frank Gift from Electric Power Research Institute for helpful discussions regarding austenitic stainless steel research in PWR primary water and information exchange.

CONTENTS

FIGURES

TABLES

ACRONYMS

ASN	Autorité de Sûreté Nucléaire (the Nuclear Safety Authority of France)
BWR	Boiling water reactor
CASS	Cast stainless steel
CBB	Creviced bent beam
CEDM	Control element drive mechanism
CERT	Constant elongation rate test
CGR	Crack growth rate
CRDM	Control rod drive mechanism
CRUD	Chalk River unidentified deposits
CT	Compact tension
CVCS	Chemical and volume control system
DCPD	Direct current potential drop
DO	Dissolved oxygen
EPRI	Electric Power Research Institute
GB	Grain boundary
GE	General Electric
HAZ	Heat affected zone
IDSCC	Inner diameter stress corrosion cracking
IG	Intergranular
IGA	Intergranular attack
IGSCC	Intergranular stress corrosion cracking
LAS	low alloy steel
LWR	Light water reactor
MHI	Mitsubishi Heavy Industries
MRP	Materials Reliability Program
NNL	Naval Nuclear Laboratory
NPP	Nuclear power plant
PH	Precipitation-hardened
PNNL	Pacific Northwest National Laboratory
PW	Primary water
PWR	Pressurized water reactor
RHR	Residual heat removal
SCC	Stress corrosion cracking

SCCGR	Stress corrosion crack growth rate
SI	Safety injection
SG	Steam generator
SMAW	Shielded metal arc welding
SS	Stainless steel
SSRT	Slow strain rate testing
TG	Transgranular
TGSCC	Transgranular stress corrosion cracking
TIG	Tungsten inert gas

Preparation for Stress Corrosion Crack Initiation Test of Austenitic Stainless Steels in PWR Primary Water

1. Field Experience of Austenitic Stainless Steels in PWR Primary Circuit

1.1 Stainless Steel Application in PWRs

Stainless steels have been widely used in light water reactor (LWR) coolant systems since the first commercial reactors were commissioned in the late 1950s, primarily because of cost, good general corrosion resistance, high ductility (in the absence of high fast neutron irradiation), ease of fabrication and wide practical experience in other industries [1]. Figure 1 illustrates the material selection for typical pressurized water reactor (PWR) components where stainless steels are used in plate and piping, castings, cladding, and weldments.

A wide variety of mechanical properties and corrosion resistance can be achieved in stainless steels by compositional control and heat treatment. As shown in Figure 2, the most commonly used stainless steel (SS) family is based on the iron (Fe)-18% chromium (Cr)-8% nickel (Ni) composition, i.e., Type 304 SS. Further adjustments of alloying elements are made to produce new variants in this family for specific properties such as reduced grain boundary (GB) sensitization, higher strength, better machinability, improved corrosion resistance, etc.

Depending on the manufacturing method, stainless steels can be classified into cast stainless steels (CASS) and wrought SS. The latter can be further classified in terms of their dominant crystal structure: austenite, ferrite, martensite, and duplex. All these types of SS have found use in LWRs, with their main characteristics and applications summarized in Table 1. Among them, Type 304 and 316 austenitic SS are the most widely used grades in PWR, mainly for structural supports of the reactor core and pressure-retaining boundary components such as piping in the primary circuit. The widespread SCC issues in the thermally sensitized weld heat affected zone (HAZ) of these austenitic SS in boiling water reactors (BWRs) in the 1970s led to the introduction of their low carbon grades Type 304L and 316L in both newer BWRs and PWRs. Some countries adopted an alternative approach that uses niobium or titanium stabilized grades such as Type 347 and Type 321. The carbon in these SS is trapped as stable niobium or titanium carbides, thus greatly reducing their susceptibility to sensitization. Type 308 and 309 SS are commonly used as weld overlay claddings on the internal surfaces of the less corrosion-resistant low alloy steel (LAS) pressure vessel components. High-strength SS such as A286 precipitation hardened (PH) austenitic SS, martensitic SS A410 and 17-4PH are used in PWRs for fasteners (nuts and bolts), springs, and valve elements. Operating experience with these SS has generally been very good given their extensive use, with the major concern being raised on the irradiation effects on core support structures.

Nevertheless, stress corrosion cracking (SCC) cases of unirradiated austenitic SS in PWR primary water have occurred in the field since the early 1980s [2] and continued to this day, with the most recent incidents reported in several French nuclear power plants (NPPs) [3], causing at least twelve French nuclear power reactor units to go offline in 2021-2022 for inspection, resulting in a profound impact on people's daily life and significant financial loss. These incidents suggest that SCC of unirradiated austenitic SS can be a performance-limiting degradation mode to in-service PWRs, and high-quality laboratory data are needed to provide key information for future degradation evaluation. To devise an adequate test plan to address such a need, relevant operating experience and associated laboratory research on the SCC of unirradiated austenitic SCC in PWR primary water will be reviewed in the following sections.

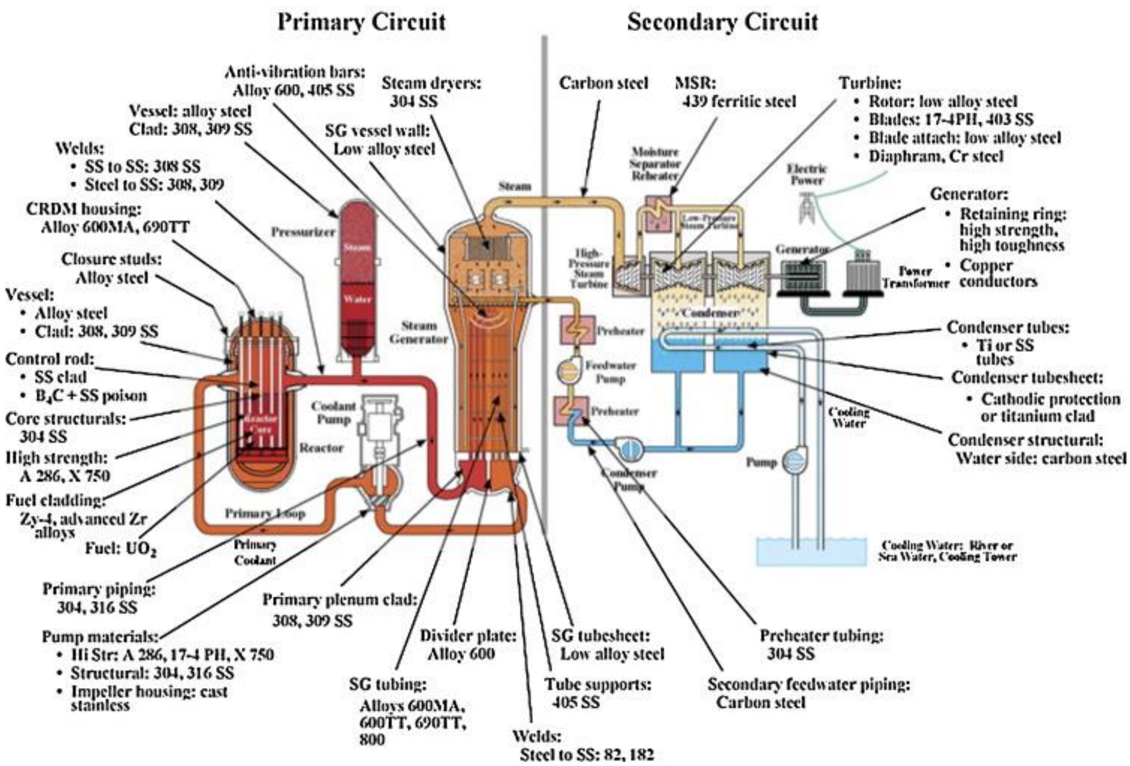

Figure 1. Typical design and materials in a PWR (Courtesy: R. Staehle) [4].

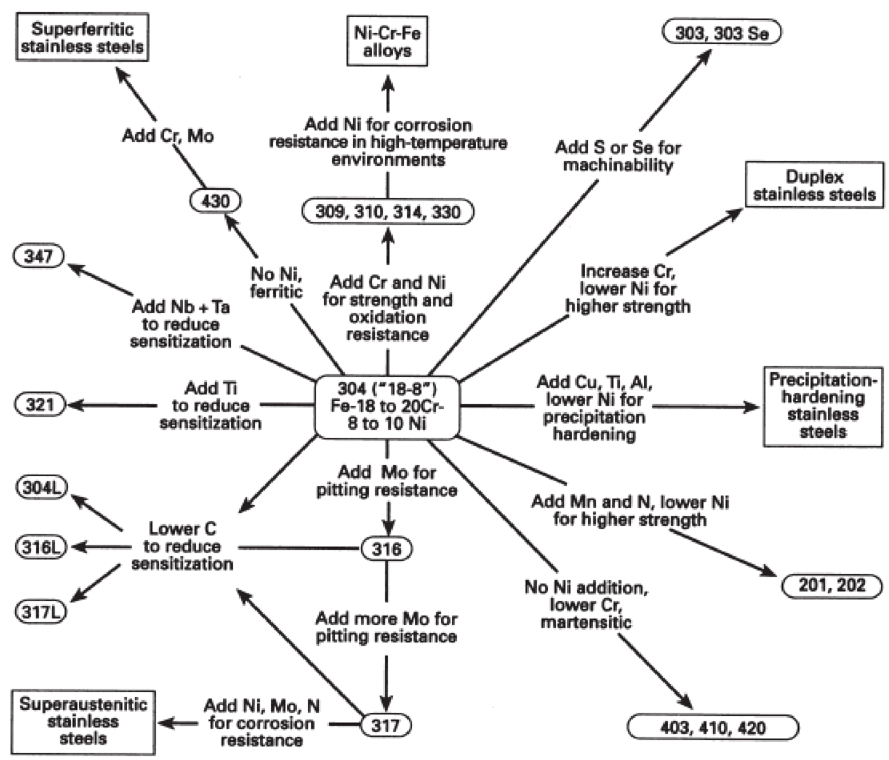

Figure 2. Composition and property linkages in the stainless steel family of alloys [5].

Table 1. Stainless steel applications in PWR primary circuit with their key properties and known degradation issues.

SS types used in LWRs (typical commercial grades)	Main properties	Main applications in LWRs	Key concerns for degradation
Austenitic SS (304(L), 316(L), 321, 347, A286)	Good combinations of strength and ductility at both low and high temperatures, good general corrosion resistance	• Primary circuit piping • Core support structures • Control rod claddings • Pump body • Preheater tubing in secondary circuit • PH austenitic SS A286 is used for fasteners and springs	Environmentally assisted cracking, including SCC and corrosion/thermal fatigue Irradiation assisted SCC
Duplex SS (308(L), 309, 310)	Good weldability, good general corrosion resistance	• Weld filler metals for joining wrought SS components, or for dissimilar metal joints between 304SS and LAS • Cladding on all carbon and LAS surfaces exposed to the reactor coolant to minimize radioactive CRUD formation	SCC due to weld defects or grain boundary sensitization – rare cases, usually happens only in atypical CASS with ferrite content <7.5%, severe sensitization, and impure oxidizing conditions in BWR coolant
Martensitic SS (410, 17-4 PH)	Increased strength	Turbine blades, valve stems, springs, jet pump rams, bolts	SCC when heat treated improperly; in particular, 17-4PH showed increased susceptibility to SCC and reduction in fracture toughness after aging at reactor operating temperature [1]
Ferritic SS (430, 405, 444)	Good general corrosion resistance, high resistance to SCC in chloride-containing environments	Not widely used for passive components in LWRs due to known issues listed on the right	Localized corrosion Embrittlement
CASS (CF8, CF8M, CF8C)	Good strength, good casting capability, and weldability	• Pump body and valve casings • Large diameter primary piping, elbows, nozzles	Thermal embrittlement at temperatures and times relevant to extended LWR service

1.2 Operating Experience

1.2.1 Inside Diameter SCC Incidents Documented in MRP-236

In 2017, the Electric Power Research Institute (EPRI) published a Materials Reliability Program (MRP) technical report, "SCC of Stainless Steel Components in Primary Water Circuit Environments of PWRs" (MRP-236, Rev.1 [6]), which reviewed all known incidents of SCC of austenitic SS exposed to primary water environments in PWRs up to the Year of 2016. A total of 183 SCC events initiated from vessel and pipe internal surfaces, termed inside diameter SCC (IDSCC), were documented from eight countries: China, France, Germany, Japan, Sweden, Switzerland, the United Kingdom, and the United States. Depending on the location they took place, these IDSCC events can be classified into two groups: 1) SCC associated with occluded/stagnant/off-chemistry environments, and 2) SCC under free-flowing, non-contaminated primary water (i.e., water which meets the EPRI PWR Water Chemistry Guidelines). For a region to be considered occluded, the flow condition must be stagnant or near stagnant or in an area where the flow is restricted. Figure 3 shows the IDSCC event frequency to component service life (i.e., plant age) and whether or not they occurred in regions with off-normal PWR primary water chemistry. ~78% of the events occurred within the first 30 years of service, peaking from 26 to 30 years of service life. However, the case number did not consistently increase, suggesting that these events are unrelated to plant aging. In Figure 4, the 183 IDSCC events are classified with regard to the components where they were detected. The majority of the events (~84%) occurred in low-flow regions or occluded/stagnant primary water conditions, with the highest frequency reported on control element drive mechanism (CEDM)/control rod drive mechanism (CRDM) housings and canopy and omega seal welds. The mechanism of SCC in such regions is reasonably well understood. These cases usually involve localized departures from the normal PWR primary water condition as these regions are likely to be exposed to oxygen and/or contaminated environments with various ionic species. Such conditions raise local corrosion potential and alter local pH, often leading to SCC dominated by a transgranular (TG) morphology. Consistent observations were made with operating experience. As shown in Figure 5, 80% of the IDSCC events that occurred under occluded conditions are purely TGSCC, with another 11% featuring a combination of TGSCC and pitting, intergranular attack (IGA), or intergranular SCC (IGSCC).

In comparison, IDSCC occurred in free-flowing PWR primary water conditions constitutes ~16% of the entire population, and is dominated by IGSCC. Since cold work and material sensitization were the most frequently reported influencing metallurgical factors on SCC of austenitic SS, EPRI also investigated whether these factors played a role in the reported IDSCC cases based on event documentation provided by utilities. The results are summarized in Figure 6 for sensitization and Figure 7 for cold work. While material sensitization due to Cr depletion at GBs was the primary cause for IGSCC in austenitic SS observed in BWRs, it only contributed to a small number of the IDSCC events occurred in free-flowing primary water (Figure 6). In comparison, besides the cases in which cold work condition is unknown, all the other IDSCC events in free-flowing primary water were associated with severe cold work, indicated by hardness values in excess of 300 HV, a rough equivalent to >20% cold work [6]. When looking at the location where IGSCC occurred in free-flowing PWR primary water (Figure 4), the majority of the cases were associated with pressurizer heater sheaths (~41%) and the chemical and volume control (CVCS) heat exchanger tubes (33%). It should be noted that most IGSCC events found in pressurizer heater sheaths are located close to the support plates. Although such locations are not directly defined as creviced location, crevice conditions may still be implicated [7]. In addition, among all these IDSCC events under free-flowing conditions, there is only one event occurred in the main reactor coolant loop SS piping, which was found in the steam generator (SG) nozzle safe end weld in Mihama Unit 2 in Japan.

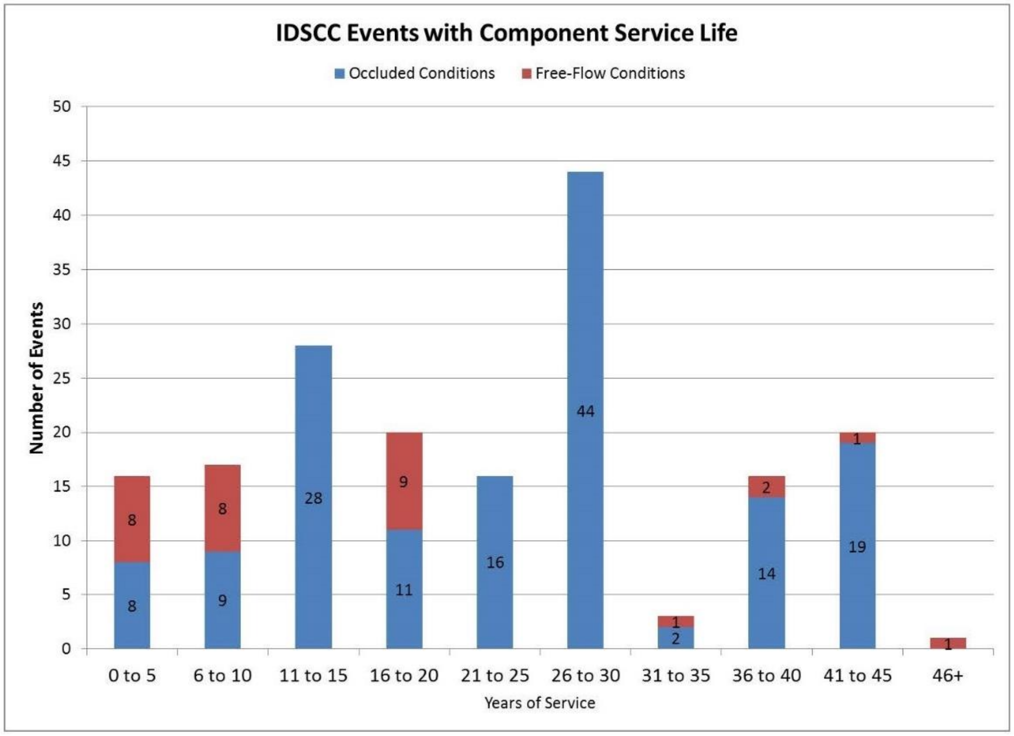

Figure 3. Distribution of IDSCC events in austenitic SS by component service life (i.e., plant age) [6].

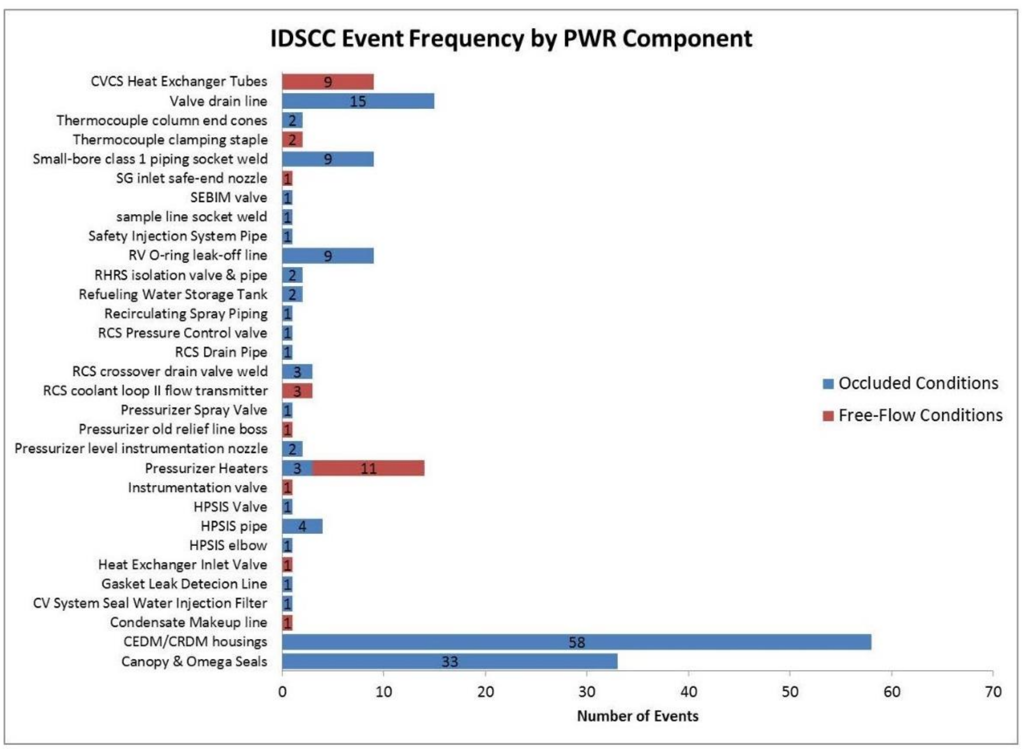

Figure 4. Distribution of 183 IDSCC events as a function of affected SS components in PWR primary water under occluded and free-flow conditions up to 2016 [6].

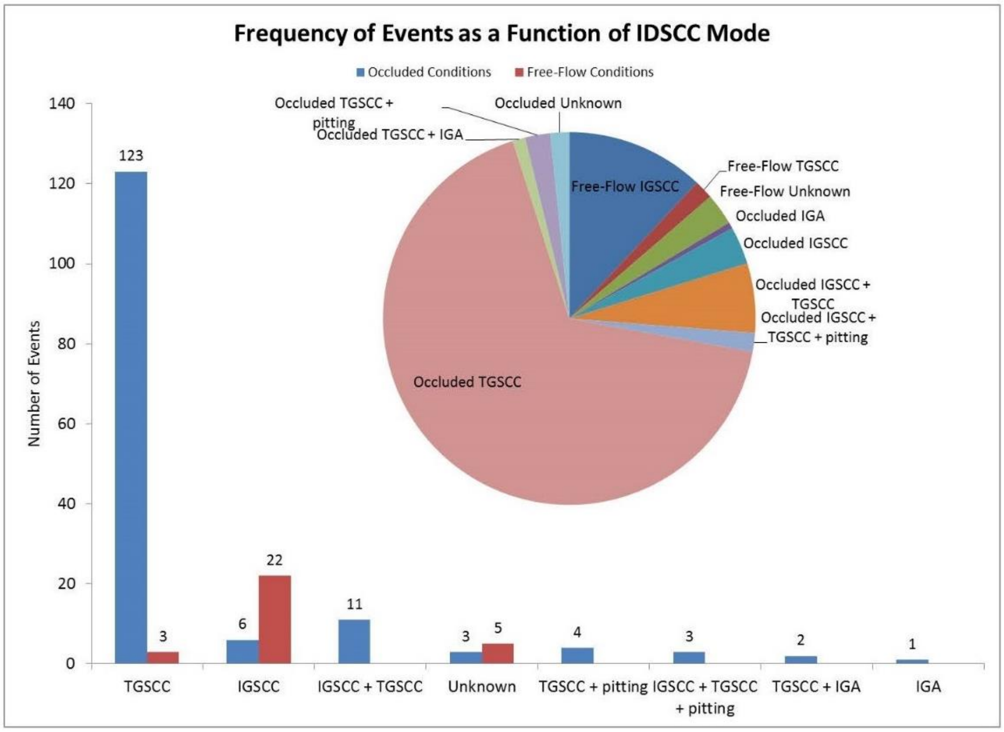

Figure 5. Distribution of 183 IDSCC events in SS components as a function of cracking mode in PWR primary water under occuluded and free-flow conditions [6].

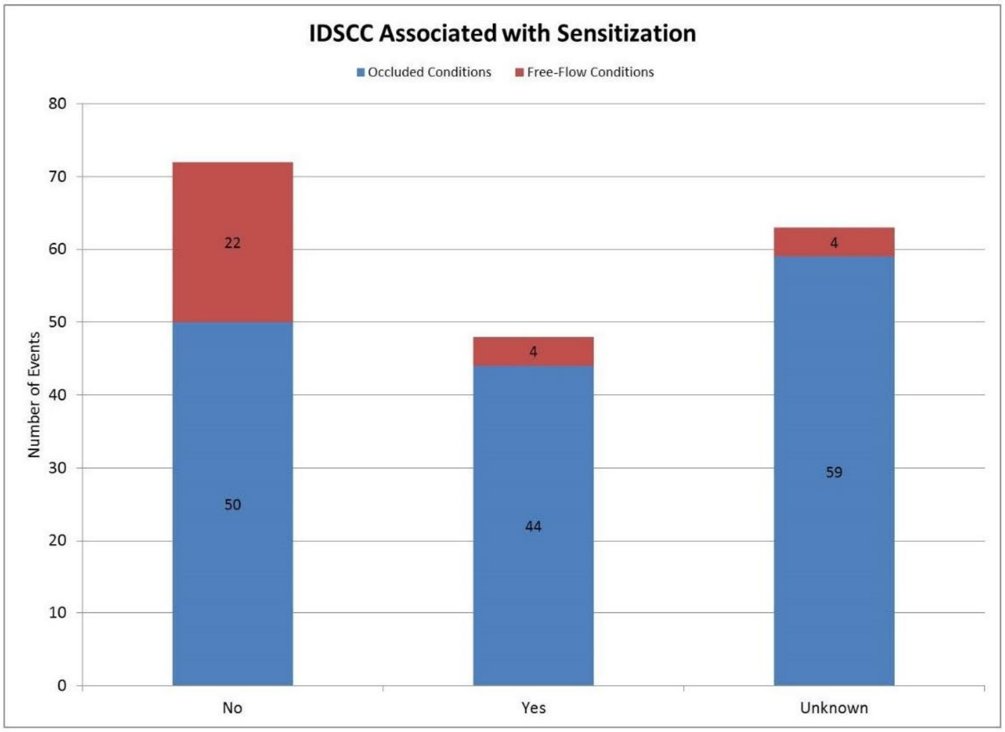

Figure 6. Distribution of 183 IDSCC events in SS components showing the proportion of events in which material sensitization was identified as a contributor to IDSCC in PWR primary water under occluded and free-flow conditions [6].

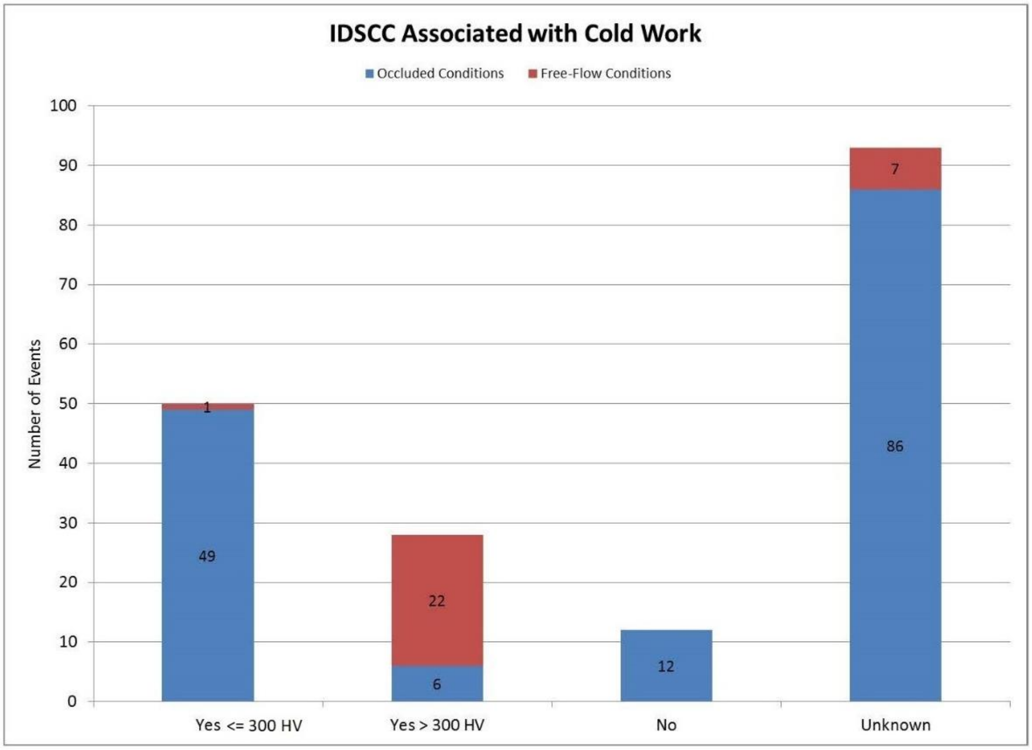

Figure 7. Distribution of 183 IDSCC events in SS components showing the proportion of events in which cold work was identified as a contributor to IDSCC in PWR primary water under occluded and free-flow conditions [6].

While the MRP-236, Rev. 1 [6] summarized IDSCC events in austenitic SS reported in the field up to the Year of 2016, new IGSCC cases in free-flowing PWR primary water conditions have since occurred in the past couple of years, notably in Japan and France. A short summary of these new IDSCC events are provided below.

1.2.2 Ohi-3: SCC in 316SS HAZ in Pressurizer Spray Line

In August 2020, a crack was detected in Japanese Ohi NPP Unit 3 (online since 1991) during the in-service inspection in the pressurizer spray line piping 316 SS weldment (Figure 8). The crack was located in the inside diameter of the weldment exposed to flowing PWR primary water at 290°C. As shown in Figure 9, the crack exhibits an IGSCC morphology. It was initiated from the inside surface in the weld HAZ closely adjacent to the fusion line, extended ~60 mm circumferentially and reached a maximum depth of 4.4 mm, but did not grow into the weld. A series of analyses were performed to determine the root cause of this SCC event. No surface damage layer, ionic contaminants, weld defects, or indication of sensitization were found, but hardness values as high as ~350 HV were measured in near-surface regions with a distance of ~0.2–0.33 mm from the fusion line on the elbow side [8]. These values exceeded the ~300 HV that has been shown to be responsible for most IDSCC events documented in MRP-236 Rev. 1, indicating this region was subjected to high weld residual stress, which likely played a key role in this SCC event. Comparison of the hardness values obtained in field weldments produced by different welding procedures at different locations of similar pipings suggest that the high hardness may be a result of unusual high heat input by a specific SMAW + TIG welding procedure and the elbow geometry [8].

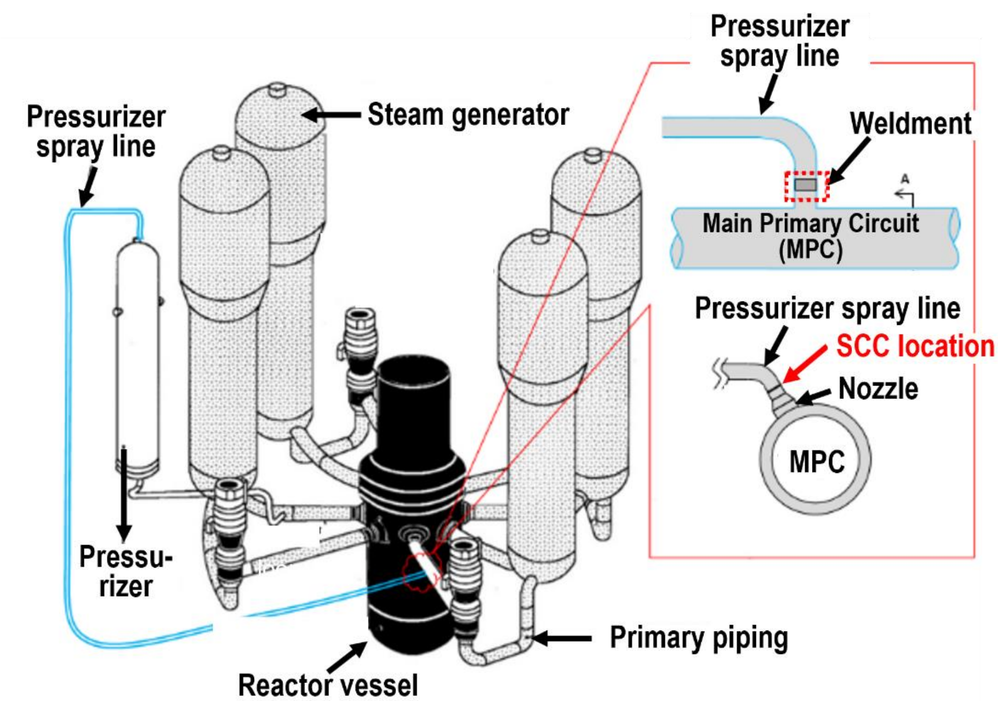

Figure 8. Location of the SCC in a pressurizer spray line 316 SS weldment detected in the Ohi 3 in August 2020, redacted for English based on [9].

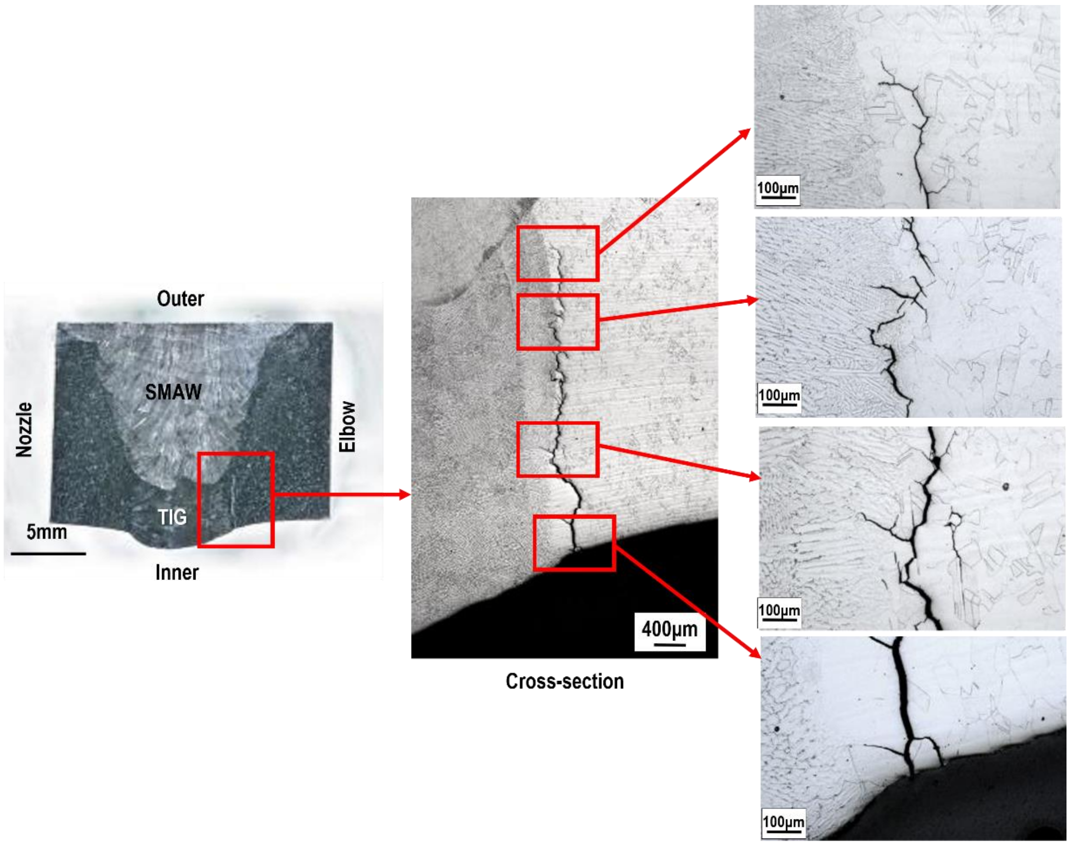

Figure 9. Optical images of one cross-section of the SCC crack found in the 316 SS weldment highlighted in Figure 8 [9].

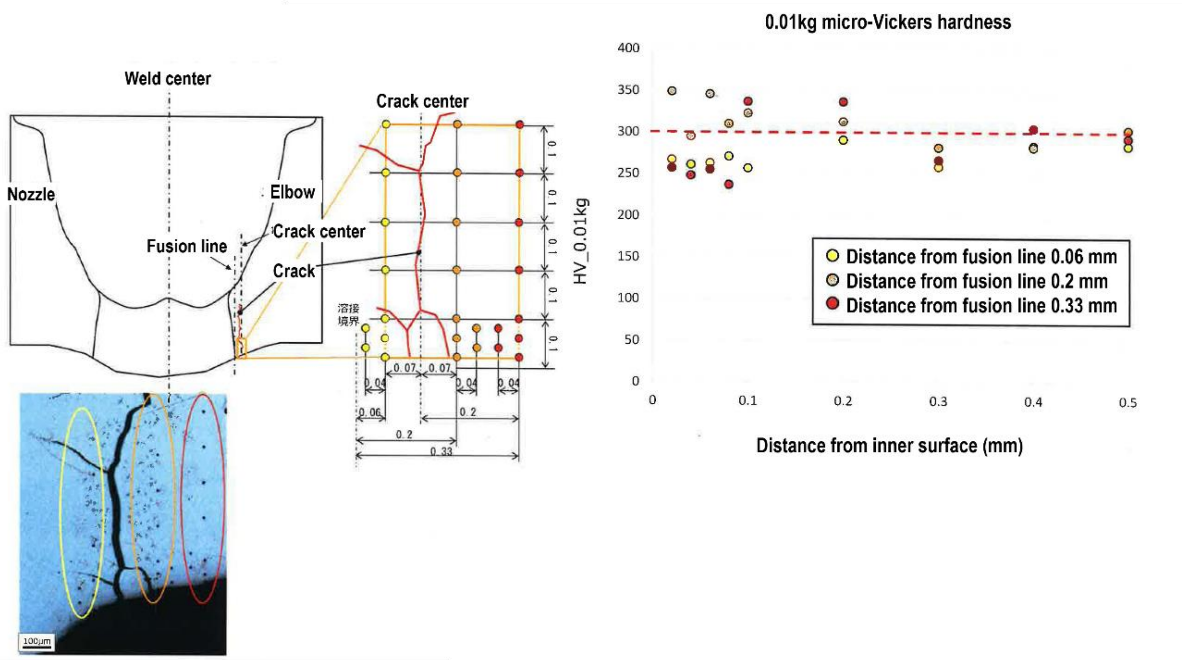

Figure 10. Hardness measurements in close vicinity of the SCC crack [9].

1.2.3 Multiple French N4 and P4 units: SCC in Safety Injection (SI) Lines and Residual Heat Removal (RHR) Lines

In October 2021, flaw indications were detected in the SI lines connected to the cold leg of the primary circuit at Civaux-1 during the 2nd 10-year inspection [10]. The SI system is used to control accidents in which coolant escapes from the primary system due to a leak [11]. In French NPPs, it consists of a low-pressure and a medium-pressure injection systems as well as the pressure accumulators (Figure 11). It is connected in several places to the loops of the primary system, through which the water heated in the reactor core is fed into the SG and then cooled and returned to the reactor core [11]. The flaw was detected in an elbow weld joint made of non-sensitized 316LN, exposed to free-flowing, non-polluted hydrogenated primary water at ~300°C. Since thermal fatigue cracking is a known issue in welds in such components, the regular ultrasonic inspection was specifically optimized to screen for thermal fatigue cracking. However, destructive examination confirmed this is a fully intergranular crack (Figure 12c) - a piece of unambiguous evidence for SCC - because thermal fatigue cracking is usually transgranular and fairly straight. The crack exhibits a circumferential orientation with a large aspect ratio (surface length >> depth) in the weld HAZ. This weldment has a very large root pass (Figure 12b), but the subsequent passes were annealed during welding, which is believed to introduce compressive residual stress in the adjacent base metal [12]. This welding procedure may help explain the wider profile of high hardness values (≥270 HV) measured from the inner face (Figure 13) [13]. The fact that the crack did not grow deeper beyond the root pass also seems consistent with the quick drop in hardness measured from the inner surface near the subsequent passes. Although these results suggest that crack propagation strongly depends on weld residual stress in the HAZ region, questions remain on the cause of the initiation of this SCC crack. According to the Autorité de Sûreté Nucléaire (ASN), elevated stresses caused by cyclic loading due to thermal stratification is suspected to be the primary contributor [12], followed by weld residual stress and potentially elevated oxygen due to dissolved oxygen in make-up water [14].

Upon detection of the above IGSCC incident in Civaux-1, EdF subsequently shutdown multiple NPPs for investigations of defect alike, and additional indications were detected in multiple units at the time of this report: Civaux-2, Chooz-1 & -2, Penly-1, Chinon-3, Cattenon-3, Flamanviller-2, and Golfech-1. These crack indications mostly occurred in SI lines and RHR lines in N4 (1450 MW) and P4 (1300 MW) type reactors, but not in the older, 900 MW plants [3]. The largest SCC crack detected thus far was found in Penly-1 in the HAZ of a weld in a SI line connected to the hot leg of the primary circuit. As shown in Figure 14b, this weld was fully emerged in ~300°C deaerated primary water insusceptible to thermal stratification effects but was repaired twice during manufacturing [10]. By the time it was detected, the crack had grown to a circumferential length of 155 mm (~1/4 circumference) and a maximum depth of 23 mm (85% through-wall thickness) [10]. To date, additional inspections and investigations are still ongoing at EdF.

Table 2. provides an overview of most of the IDSCC events in free-flowing PWR primary water documented in MRP-236, Rev. 1 [6], together with the new cases in Japan and France summarized above.

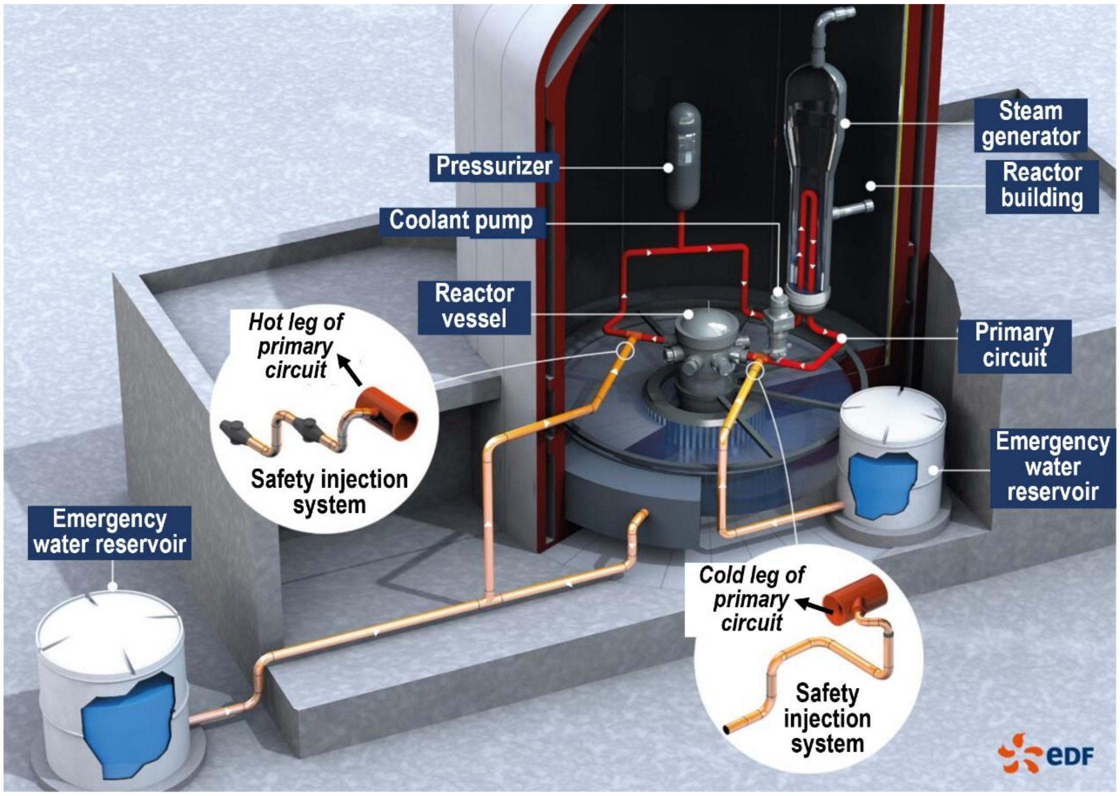

Figure 11. Illustration of the safety injection system location in the French N4 NPP design, redacted for English based on [15].

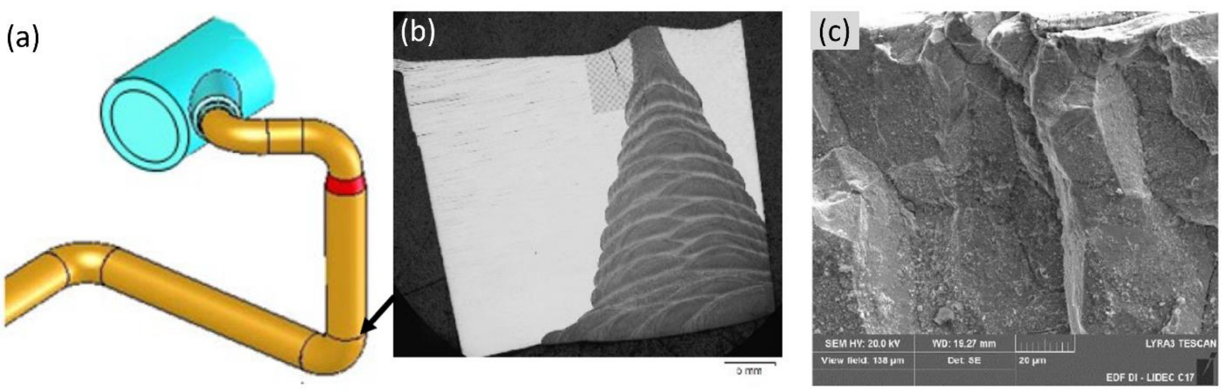

Figure 12. SCC found in the safety injection lines in Civaux-1 [13].

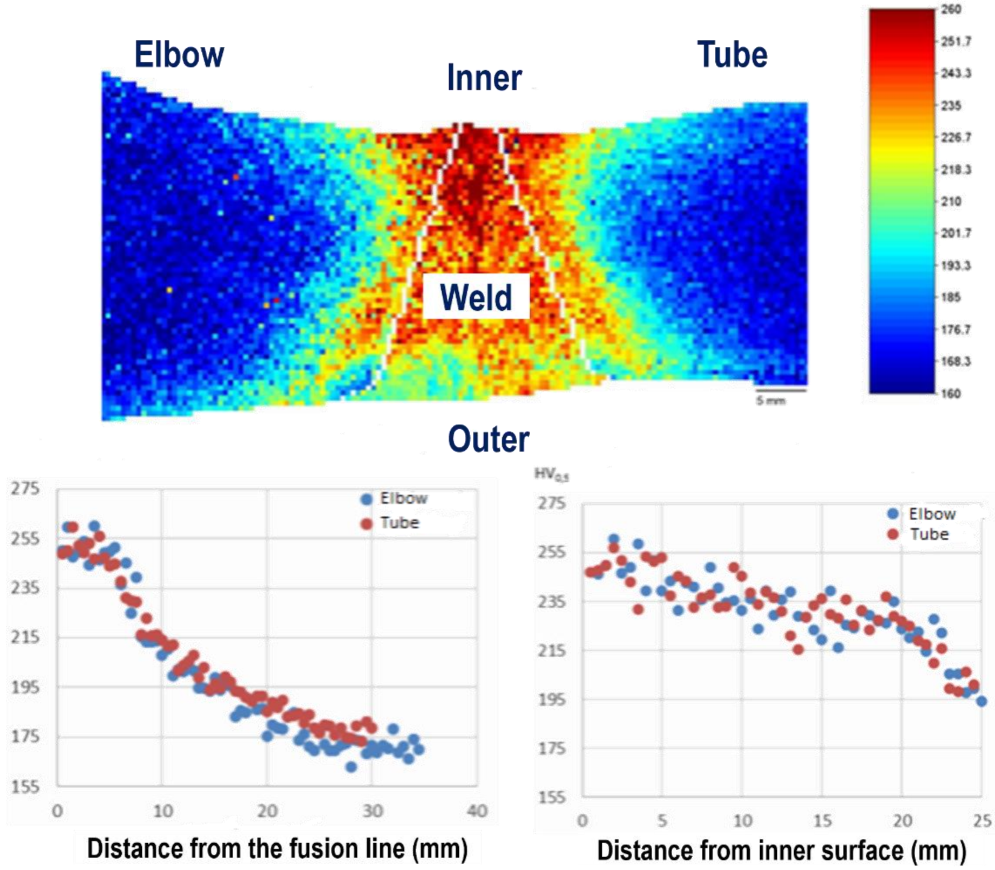

Figure 13. Hardness measurements of a cross-section containing the SCC crack found in Civaux 1 [13].

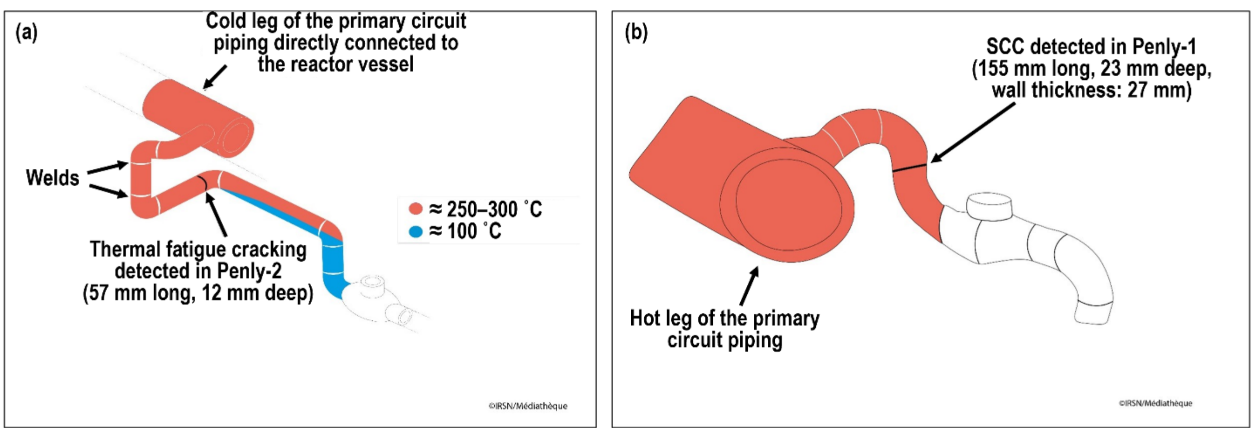

Figure 14. Schematic illustration of the location of cracks detected at Penly-1 and 2 during supplementary inspections of the SI lines connected to the main primary circuit: (a) a thermal fatigue crack detected in Penly-2 and (b) an SCC crack detected in Penly-1, redacted for English based on [10].

Table 2. Summary of IDSCC events occurred in nominal free-flowing PWR primary water conditions.

Location	# of cases /Country /Year of occurrence (of service)	Crack morphology	Material	Temp (°C)	Root cause based on failure analysis
Pressurizer heater sheaths	4/USA, 1980-2012 6/Framatome, 1997-2006 1/UK, 2010	IGSCC	316/316L	345	Surface cold work (hardness ≥295 HV) imparted by fabrication, with a possible elevation of pH due to lithium concentration in creviced regions and minor contribution by cyclic load due to the cycling of heaters. Only one case was related to partial sensitization, transient oxygen, and possible copper contamination.
CVCS heat exchanger tubes	9/France, 1986, 1994	IGSCC & TGSCC	304L	140 – 200	Severe cold work due to cold rolling (hardness >300 HV)
SG inlet nozzle safe end	1/Japan, 2007 (11 years)	IGSCC	316		Post-weld machining of the inner bore resulting in high surface hardness (>450 HV) and high tensile residual stresses
RHR system elbow piping	1/China, 2010 (15 years)	IGSCC	316		Surface grinding in the vicinity of the cracks
Pressurizer spray line	1/Japan, 2020 (19 years)	IGSCC	316L	290	High cold work (hardness ≥300 HV) produced in HAZ by high heat input during welding
SI & RHR lines elbow piping	9/France, 2021-2022 (15–23 years)	IGSCC	316LN	250–300	One case was associated with elevated stress due to thermal stratification and high cold work in HAZ by welding, plus potentially elevated oxygen due to dissolved oxygen in make-up water; Investigations ongoing for other cases.

1.3 Summary

The review of operating experience with austenitic SS IGSCC incidents that occurred in nominal free-flowing conditions revealed several key findings:

- Based on all cases where failure analyses were performed and documented, severe cold work has been a key contributor to IGSCC with hardness values commonly measured at 275 HV and above (typically ≥300 HV). The cold work was either produced as surface cold work by machining or grinding or bulk cold work produced by fabrication (e.g., cold rolling, swaging), installation (e.g., crimping, bending), and weld-induced shrinkage.
- While most cases occurred in nominally good-quality, free-flowing PWR primary water, some cases suggest out-of-specification primary water chemistry may have contributed to observed IGSCC cases:
  - Creviced conditions may exist under pressurizer heater support plates. Elevated pH can be introduced in such occluded environments when boron concentration is very low (e.g., at the end of the fuel cycle and particularly during cycle stretch-out [16]) and has been shown to increase IGSCC susceptibility [7].
  - Elevated oxygen can be present under certain scenarios. It can be caused by transient periods during plant shutdown and startup, or air-saturated make-up water ingress due to a growing practice not to deaerate the make-up water tanks of PWRs as the original vendors intended [16].
- For the cases observed in pressurizer heater sheaths, cyclic loading due to intermittent heater operation may contribute to IGSCC, but this could not explain the other IGSCC that occurred in static loading conditions.

The aforementioned new IGSCC cases from Japan and France represent the only confirmed cases of IDSCC events in non-isolable portions of branch piping [12] in the PWR primary circuit, suggesting that IGSCC of austenitic SS in PWR environments is a potentially serious emerging issue. While high hardness was reported in all examined IGSCC cases under free-flowing PWR primary water conditions, the root cause of IGSCC initiation remains unknown due to the apparent complexity in welding history, operation conditions, and metallurgical factors. Although no IGSCC incidents have been reported in analogous welds in U.S. PWRs, it is essential to have a better mechanistic understanding of the condition under which such IGSCC incidents could take place so that informed inspection and maintenance plan can be made to prevent such events from jeopardizing plant operation.

2. Status Review of SCC Initiation Research on Austenitic Stainless Steels in PWR Primary Water

SCC research on austenitic SS in PWR primary water has been carried out worldwide for decades, with a heavier focus on SCC propagation. In 2022, EPRI published a technical report titled Materials Reliability Program: Stress Corrosion Crack Growth Rates in Stainless Steels in PWR Environments (MRP–458) that established crack growth rate (CGR) disposition equations for assessing SCC propagation of austenitic SS in PWR primary water environments, which reflects the effects of stress intensity factor, temperature, Vickers hardness (to present retained deformation), with enhancement factors for oxygen-containing, high corrosion potential conditions, plus tolerance to chloride and sulfate impurities [17]. The Naval Nuclear Laboratory also developed an empirical equation for SCCGR of 304/304L in hydrogenated PWR primary water as a function of temperature, stress intensity factor, material cold work, orientation, and sulfur content [18]. In comparison, the SCC initiation mechanism of austenitic SS in nominally good-quality PWR primary water is not well understood, mainly because austenitic SS are generally very resistant to SCC initiation in such environments, often resulting in the incorporation of accelerants significantly far from realistic plant operating conditions for relevant investigation. However, useful information can be found from laboratory studies in the literature to guide planning for future SCC initiation research activities. The following sections summarize key information drawn from past SCC initiation research, focusing on test data obtained in normal PWR primary water conditions. When data on SCC initiation does not exist, relevant information obtained from stress corrosion crack growth rate (SCCGR) test will be summarized in the context of better understanding the SCC phenomenon in plant components.

2.1 SCC Initiation Testing Methods

Testing methods with both passive and active loading have been used to study SCC initiation of austenitic SS in PWR primary water. U-bend, notched, or humped tensile specimens, and creviced bent beam (CBB) specimens are often used for passively loaded tests. Actively loaded tests at target stress or strain were also employed to study this topic, but it is often very challenging to produce SCC initiation in a realistic timeframe using these methods. Therefore slow strain rate testing (SSRT), also called constant elongation rate testing (CERT), became the most commonly used active loading method. While these testing methods can rapidly screen material SCC initiation susceptibility regarding different environmental or metallurgical factors, the loading conditions are too severe to allow for any direct correlations between test results and plant operating conditions. For example, in an SSRT test, a tensile specimen is slowly and monotonically strained to a predefined load/strain, which tends to eliminate the precursor incubation period preceding crack initiation due to the continuous perturbation of surface oxide films imposed by the increasing load. In addition, the plastic strain introduced during loading will change the amount of cold work in the material unless the additional introduced amount is very small compared to the starting cold work condition.

On the other side, complex strain path has been identified by operating experience as an important factor affecting SCC initiation. Several laboratory studies tried to emulate this situation by using multi-axial loaded or humped specimens or testing specimens machined from materials prepared by complex machining/straining/shearing procedures. Nevertheless, the most commonly used testing method is still SSRT, again posing difficulty in gauging mechanistic understanding from acquired data due to the severity of the loading condition.

In addition, except for a few studies that attempted to measure SCC initiation response in situ via electrochemical and acoustic methods [19, 20], most studies did not employ any in-situ monitoring techniques, and SCC initiation was only confirmed by surface examinations carried out at planned test interruptions or after the conclusion of the tests.

To address the limitations of conventional SCC initiation studies, the Naval Nuclear Laboratory (NNL) developed a laboratory research program in recent years to test elbows and pipes that had been processed to include varying through-thickness cold work and residual stress. Laboratory ball-sizing techniques developed in-house were used to impart varying through-thickness cold work and residual stresses in multiple prefabricated elbow and pipe specimen sections (Figure 15). As illustrated in Figure 16, these sections were then welded into flanged test spools and bolted together in series, allowing simulated PWR primary water flow through during testing. The specimens were monitored in situ using direct current potential drop (DCPD) and periodic ultrasonic testing inspections to detect and monitor SCC initiation and growth. Such a test setup provides a viable option to create component scale samples containing residual stress and plastic strain that closely emulate in-plant conditions. Early results from this test program will be summarized in the following sections when applicable.

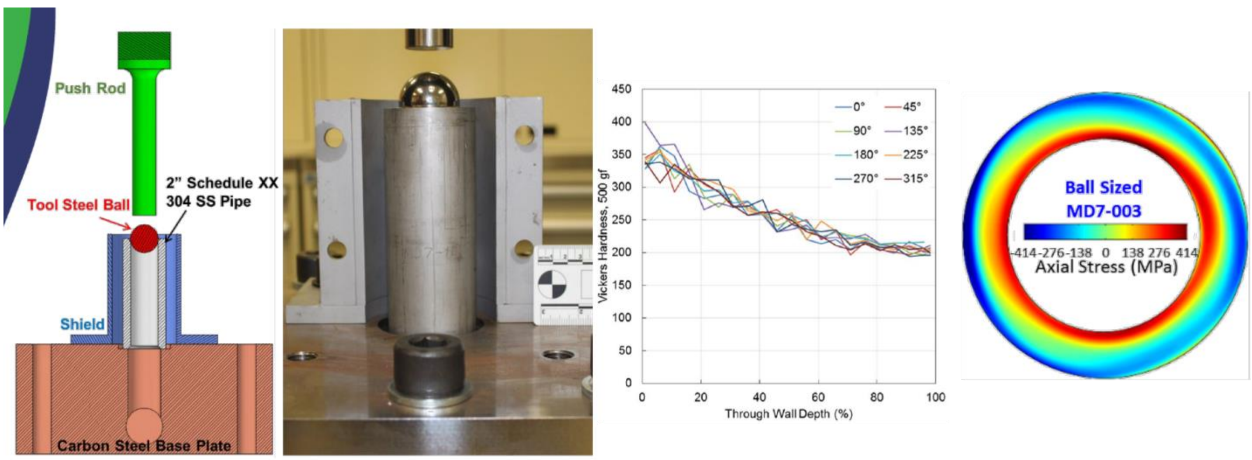

Figure 15. Ball-sizing technique and finite element simulations developed at NNL to impart through-thickness cold work and residual stresses in pipings [21].

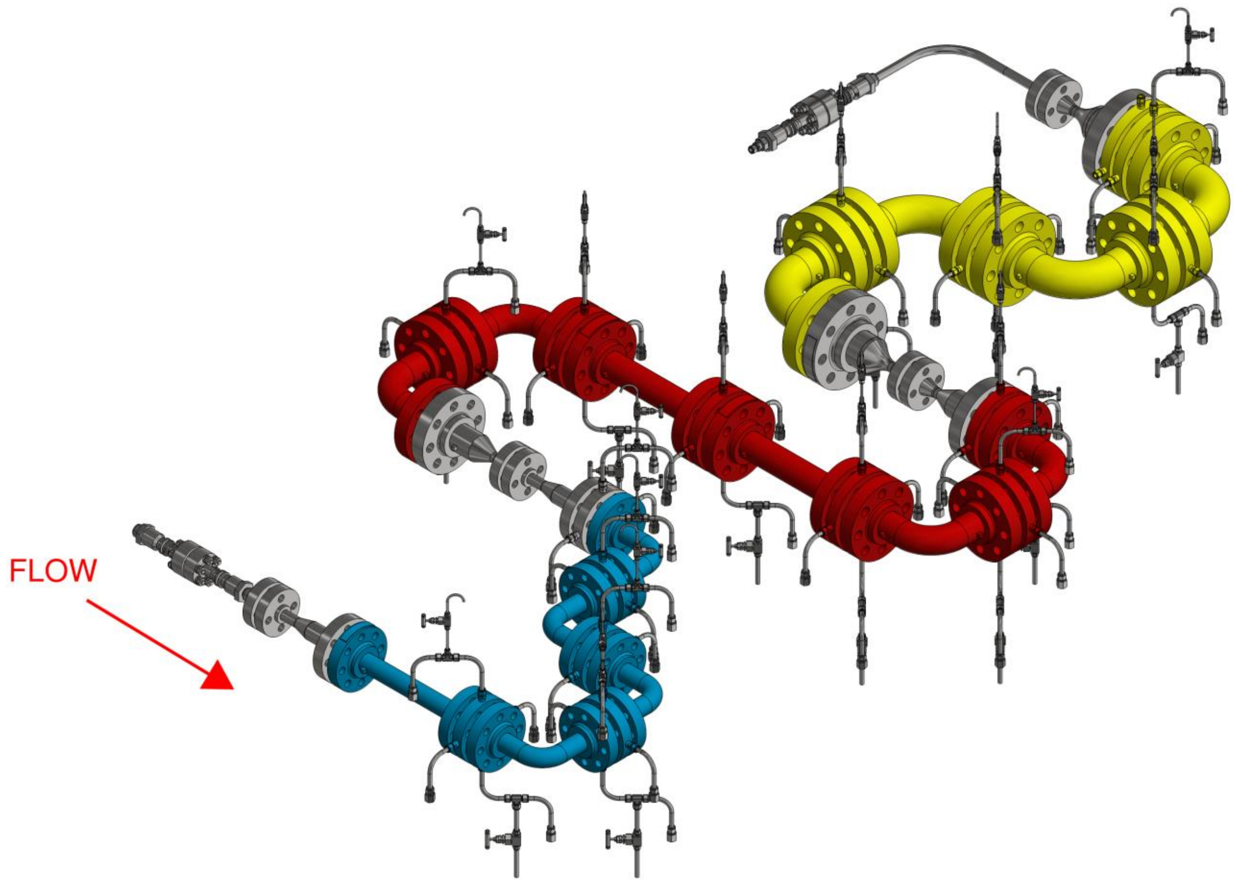

Figure 16. Schematic layout for the elbow test loop with 15 test sections at NNL. Blue: 2-inch elbow schedule XX. Red: 3-inch schedule 160. Yellow: 5-in schedule 120. Thermocouples at inlet, outlet, and mid-loop. The low-temperature reference electrode is located at the last 4-inch elbow (loop outlet). [22].

2.2 Key Influencing Factors

2.2.1 Cold Work

In plants, cold work can be induced by various industrial process operations such as turning, reaming, swaging, bending, bolting, cutting, thermal cycling of components, and hammering, just to name a few. Depending on the specific procedure, the imparted cold work can either be surface cold work, affecting only the first few grains to millimeters (mm), or bulk cold work [6]. Welding-induced shrinkage is another common source of cold work in HAZ. Its severity depends on welding parameters such as heat input, welding speed, and the number of welding passes and is often more prominent in pipings with larger diameters.

Observations of the effect of cold work on SCC were initially made for BWR components operating in relatively oxidizing environments, and, as a result, in General Electric (GE) BWRs, the amount of strain hardening is confined to < 5% [16]. In comparison, the use of cold worked wrought Type 3XX SS components is more widely allowed for PWR components where the environment is less oxidizing. As summarized in Section 1.3, the IDSCC cases observed in free-flowing PWR primary water were all associated with high local hardness values exceeding 300 HV. Such values are usually equivalent to a stress of ~700 MPa in L-grade SS materials, and indicate a cold work level of at least 20% [16].

Unlike the well-established SCCGR dataset obtained by fracture mechanics testing using compact tension (CT) specimens on austenitic SS as a function of cold work, no such body of data exists for SCC initiation time associated with cold work or hardness in literature. As mentioned earlier, this is mainly due to the high resistance of austenitic SS to SCC initiation in normal PWR primary water. At constant load or strain, it is extremely difficult to initiate SCC in a realistic testing timeframe even if the materials are excessively cold worked. In one study, crack initiation leading to sustained propagation in severely cold worked austenitic SS 304(L)/316(L) (with hardness up to 450 HV) has proved to be very difficult despite small TGSCC initiations up to 20 µm deep in notched tensile specimens, even after 17000 hours at 360°C [23]. In another case, the authors induced excessive plastic deformation by pressing a hump into 15 to 30% cold-rolled 316 SS strips before forming U-bends. Surface hardness analysis of the U-bend specimens with a humped section shows hardness values of 370–390 HV. The resulting straining was so severe that ductile tears were observed, from which IGSCC initiated after the specimens were exposed to PWR primary water. In comparison, less severely stressed specimens did not tear or crack after the test [24].

Consequently, the cold work effect was more commonly evaluated using SSRT. In one work, Couvant et al. investigated the effect of strain path variation on the SCC initiation in SS 304L in 360°C PWR primary water using SSRT on tensile specimens [25]. The test strain was performed along different directions with respect to the prior deformation process (Figure 17). Vickers micro-hardness measurements were performed near the initiation areas in specimens after the tests ended with the result summarized in Figure 18. Assuming that microhardness did not evolve near the edges of the cracks during the propagation stage except at the crack tip, the authors claimed a micro-hardness threshold for initiation at ~250 $HV_{0.1}$ and ~310 $HV_{0.1}$.

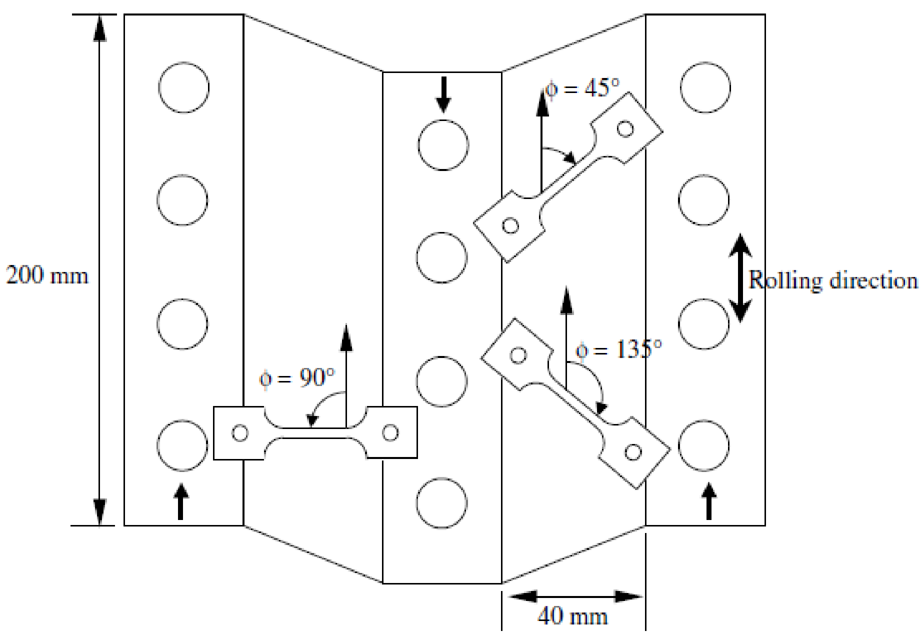

Figure 17. Machining of secondary specimens in pre-sheared materials in air at 25°C [25].

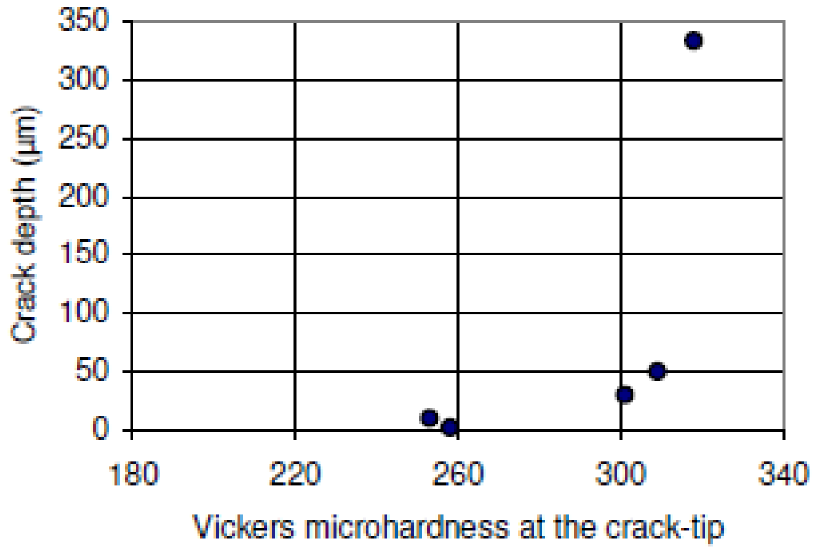

Figure 18. The micro-hardness threshold for initiation and propagation of SCC during SSRT in PWR environment (360°C, strain rate = 5×10^{-8}/s) [25].

Besides the level of cold work, a number of laboratory studies also pointed out that the "strain path" due to cold work has a profound impact on the SCC initiation susceptibility of austenitic SS in PWR primary water. In the same work mentioned above, as shown in Figure 17, plane tensile specimens have been tested monotonically, i.e. in the same direction as the pre-straining, in the reverse direction to the prestraining, and at right angles to the pre-straining direction. IGSCC is clearly favored by the more complex strain paths such that the completely reversed strain path is the most severe. It is believed that oxidation is enhanced along pathways where strain is localized. Together with enhanced stress concentration ahead of crack tips due to strain hardening, it enabled cracks to grow by sequential brittle fracture of oxide penetrations ahead of the crack tip. A continuation of this investigation used cruciform specimens, allowing the prestrain and test-strain to be applied orthogonally [26]. A series of tests were conducted using specimens strained under tension, which address the experimental correlation between SCC initiation and strain localization in Types 304L and 316L austenitic stainless steels. Results showed that complex strains promoted IGSCC initiation in stainless steels. High straining levels were not required in the initial (prior) or subsequent straining phase when applied as a complex strain sequence to cause SCC initiation in stainless steels.

Strain incompatibilities were found to be significant factors in promoting the localized initiation of SCC. Figure 19 illustrates that the initiation of an intergranular crack during SSRT testing occurred on the boundary separating regions of high and low strain, as determined from digital image correlation during the test [27]. Complex strain path changes are believed to result in the activation of different deformation systems, potentially causing a rapid latent hardening effect in certain grains. Such an effect may increase the stresses between grains and contribute to the observed tendency for IG cracking following a strain path change. There was also a correlation between local strain concentrations and the ease of oxidation of the slip lines generated by the strains, which increased the material's susceptibility to SCC.

In addition, it has been reported that cold work could affect SCC morphology in both BWR and PWR water chemistries. TGSCC occurs more often with increasing cold work, whereas IGSCC dominates in lower cold work conditions [28].

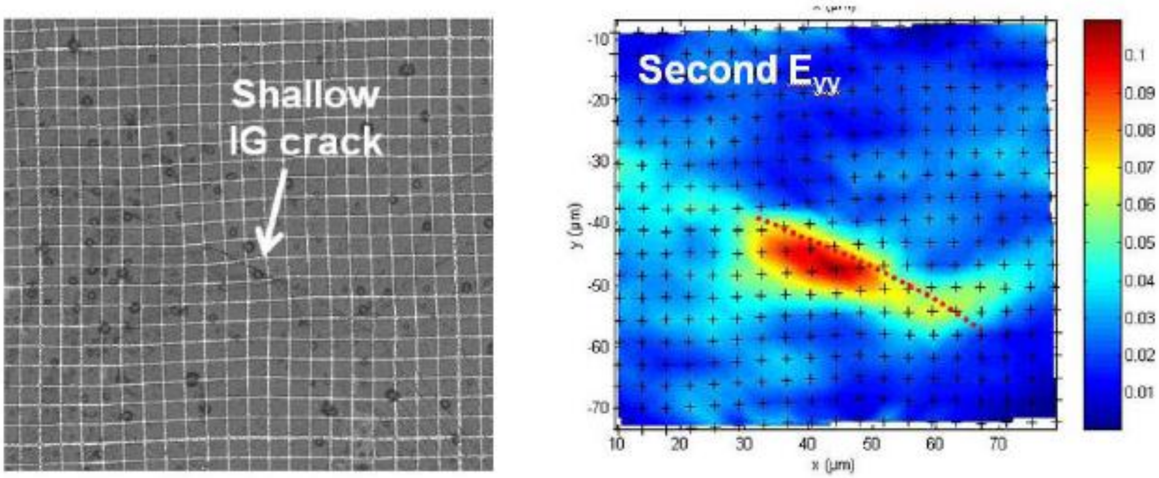

Figure 19. IGSCC initiation during SSRT relative to the surface strain as measured by digital image correlation [27].

2.2.2 Sensitization

Material sensitization is a multifaceted effect involving a broad range of material, thermal and mechanical parameters that predominantly lead to nucleation and growth of Cr carbides along GBs, resulting in Cr depletion at GBs that are not covered by Cr carbides. Formation of Cr carbides occurs when the material is exposed to temperatures over the range ~500-850°C during either isothermal treatment or cooling following exposure to higher temperatures [29]. In the latter case, cooling is usually followed by high heat input during fabrication or weld repair operations. One weld cycle alone can sensitize a material.

Although material sensitization is a key contributor to SCC initiation and propagation of austenitic SS in oxygenated environments such as BWR normal water chemistry, SCC initiation studies comparing solution annealed and sensitized cold worked Type 304 and 316SS both consistently show that sensitization does not affect SCC initiation in hydrogenated PWR primary water, probably due to the low corrosion potential in such environments. Interestingly, sensitization showed a complex effect in SCC propagation behavior of austenitic SS in PWR primary water, resulting in a non-arrhenius dependency of CGR on temperature [30]. It appears that sensitization promotes SCCGRs at temperatures less than 250°C but enhances and extends the propensity for SCCGR retardation at higher temperatures. Such a tendency was observed both in sensitized and then cold worked and cold worked and then sensitized 304 SS, suggesting that Cr depletion, not the enhancement of deformation at GBs due to carbides, controls this dependency.

2.2.3 Surface finish

Operating experience showed that several IDSCC field incidents in free-flowing PWR primary water are associated with heavily deformed surface layers with excessive hardness values measured as high as >470 HV [31]. Consequently, laboratory tests have been performed at several laboratories to study the effects of machined surface finish on SCC initiation of austenitic SS, revealing mixed results.

Zhong et al. [32] modified the inner surface finish of hollowed 316SS cylindrical tensile specimens by either drilling or honing, and performed SSRT tests on them in 325°C PWR primary water. Both machining techniques produced a hardened layer on the inner surface of the specimens, with higher hardness values measured within the first ~100 µm from the drilled surface. A mixed mode of IGSCC +

TGSCC initiation was observed on both surfaces. The drilled specimens showed a higher density of cracks, consistent with the higher surface hardness measured immediately below the drilled surface.

Chang et al. [33] performed a few machining procedures to produce a surface-hardened layer of varying depth (50-90 μm) and cold work level on 316L SS tensile specimens. SSRT tests were performed in 300°C PWR primary water. The result revealed a dependency between the susceptibility to SCC initiation and machining method and orientation. A higher initiation susceptibility was found in specimens with machining grooves orientated perpendicular to the loading direction, where they likely acted as stress amplifiers. Nevertheless, all initiated cracks are transgranular and confined within the deformation layer produced by surface machining. Interestingly, in another study performed by the same authors, warm-forged 316L exhibited higher susceptibility to IGSCC and deeper cracks in specimens with a highly polished surface than those with a machined surface [34]. The primary cause was attributed to the ultrafine-grained layer induced by surface machining, which promoted faster uniform oxidation than the highly polished surface, thereby avoiding localized oxidation that served as precursor to IGSCC initiation on the surface. It should be noted that the SSRT method is not adequate for studying the effect of machining-induced residual stress on SCC initiation of the testing material, as the residual stresses can quickly level off to similar values upon plastic deformation in the SSRT tests.

2.2.4 Second phases

2.2.4.1 Delta ferrite

Austenitic SS, such as 304 and 316 SS, typically contain a small amount of delta ferrite in the matrix. Delta ferrite is known to promote material strength and corrosion resistance and is generally considered beneficial to SCC resistance in hydrogenated PWR primary water. However, recent SCC initiation testing performed on 304/304L SS in PWR primary water using SSRT seem to suggest the δ-ferrite/austenite interface was very susceptible to SCC initiation and promoted IGSCC in materials with a high level of delta ferrite. It is suspected that such interfaces underwent localized deformation upon dynamic straining imposed during SSRT [35, 36].

2.2.4.2 Strain-induced α'-martensite

Austenitic SS are thermodynamically metastable and prone to martensitic formation due to cold work [37]. This is particularly evident for Type 304L, which is less stable and has a lower stacking fault energy than other austenitic stainless steels such as Type 316 [38]. Chang et al. evaluated the effect of strain-induced martensite on SCC initiation of austenitic SS by performing SSRT tests on SS304L and 316L rolled at different temperatures (room temperature vs. 200°C). Results showed a lower SCC initiation susceptibility in cold-rolled 304L that exhibited martensitic transformation, which was primarily contributed to the blunting effect caused by the preferential oxidation of α'-martensite.

2.2.4.3 Others

Some intermetallic intragranular inclusions, such as MnS, tend to "dissolve" when in contact with water and can act as initiation sites for TGSCC [36].

2.2.5 Alloying elements

Among all the alloying elements, sulfur appears to have the strongest impact on the SCC susceptibility of austenitic SS in PWR primary water. Laboratory CGR testing has demonstrated that even within the content specification range (≤0.03 wt%), increasing sulfur can greatly improve the SCC resistance of austenitic SS in hydrogenated water. West et al. [39] performed SCCGR tests on model heats of SS with sulfur contents of <0.001, 0.006, and 0.012 wt% in 20% cold worked SS in 338°C deaerated, hydrogenated PWR primary water. They found a consistent reduction in SCCGR by 3X and 20X in the SS containing 0.006 and 0.012 wt% sulfur, respectively, compared to the base condition with <0.001 wt% sulfur. This reduction was associated with thicker crack tip oxide films and blunter crack tips, suggesting sulfur may have led to crack retardation in deaerated high-temperature water. The recent SCC initiation research performed using the Figure 16 setup at NNL also demonstrated a beneficial effect of higher sulfur on SCC initiation of 304/304L SS. Several heats with sulfur content varying between 0.001 and 0.006 wt% were tested as elbow and pipe specimens. SCC initiation was only found in low-sulfur elbow specimens after all three rounds of tests reported in [22]. These studies suggest that older plants may be more resistant to deaerated water SCC than equivalent fittings in modern plants because, in general, older fabrication practices tend to result in high sulfur content in SS while modern processing yields low sulfur SS. However, other environmental conditions still need to be evaluated further to clarify the role of sulfur in austenitic SS SCC.

2.2.6 Loading pattern

As partially covered in Section 3.2.1, SCC initiation at constant load or strain has proved to be very difficult. The longest-running PWR primary water constant load testing of SS is being conducted at Mistubishi Heavy Industry (MHI), where a total of ~66 specimens prepared from 304 and 316 SS with different bulk cold work or surface cold work are being evaluated at 290, 320, and 345°C PWR primary water. The test has reached 57,000 hours recently, with only one SCC initiation found in a cold bent 316 pipe specimen tested at 290°C [40].

Similar fact led to the conclusion that dynamic straining is necessary for SCC initiation to occur in austenitic SS in hydrogenated PWR primary water. Slow straining specimens up to ~5% deformation at a strain rate on the order of 10^{-8}/s appeared to be sufficient to initiate SCC cracks in the cold worked 304/316 SS [35, 36], but difficulty persists in producing SCC initiation at milder loading conditions, such as those based on periodic partial unloading. Nouraei, et al. [41] studied SCC initiation of 20% cold worked SS using blunt notch CT specimens in which notches were manufactured by grinding and oriented parallel to the cold work direction. The specimens were exposed to good-quality hydrogenated lithiated water and were tested for 3,210 hours under trapezoidal loading with an unloading period every 12 hours. The results showed that initiation processes in simulated good-quality PWR primary coolant under moderate loading is extremely slow. From the batch of 18 specimens, only two showed any evidence of SCC initiation.

It should be noted, however, that increasing the strain path complexity, for example, by introducing biaxial loading, could greatly enhance the ability to initiate cracks even under constant load conditions. SCC initiation was achieved in 304/316 SS in several cases without excessively high cold working or loading [22, 42].

2.2.7 Creep

Creep experiments in the air (at >450°C) on 20% cold worked (non-sensitized) Type 316 stainless steel has shown that cold work accelerated creep with a similar activation energy to that observed for crack propagation in PWR primary water. Post-test analytical transmission electron microscopy observations revealed Ni-enriched zones at crack tips, suggesting that diffusion of metal atoms (principally Cr) and vacancies occurred before crack opening during IGSCC and creep crack growth [43]. Nevertheless, the effect of creep on SCC initiation has yet to be explored. There is debate on whether primary or secondary (steady state) creep plays a more important role in SCC initiation. In addition, SCC initiation research performed on Ni-base alloys at PNNL demonstrated that cavity formation at the interface between GB carbides and the matrix due to creep served as an alternative crack initiation mechanism in the highly SCC-resistant Alloy 690 in highly cold worked, thermally treated microstructures featuring a semi-continuous distribution of nanometer-sized GB carbides [44] [45]. It is unknown whether a similar crack initiation mechanism exists in austenitic SS, especially in the highly strained region featuring a high density of GB carbides, which may be possible in HAZ.

2.2.8 Environment

2.2.8.1 Dissolved Oxygen (DO)

While nominal PWR primary water is hydrogenated, elevated oxygen level could be introduced into the main primary circuit during transients such as plant startup and shutdown, typically at temperatures below 120°C, or from air-saturated make-up water ingress due to a growing practice not to deaerate the make-up water tanks as the original vendors intended, which now affects about half of the operating PWR fleet [16]. The effect of elevated oxygen in SCC initiation and propagation of austenitic SS in PWR primary water was investigated using fracture mechanics CT specimen testing by Andresen & Morra [46]. They found that IGSCC easily initiated from pre-fatigue cracks in cold worked 304SS when oxygen is added to the lithium hydroxide/boric acid mixture typical of PWR primary water. The crack then continued to grow without difficulty, albeit at a significantly lower rate, after the corrosion potential is restored to low values similar to that of the hydrogen redox reaction by the normal hydrogen partial pressure of PWR primary water (Figure 20). If oxygen is added to PWR primary water so that the potential rises to values more typical of BWR NWC, the SCC growth rate in sensitized material begins to accelerate by up to nearly two orders of magnitude to a rate equivalent to ~0.3 mm per day after ~20-50 hours exposure to the oxygen transient [46].

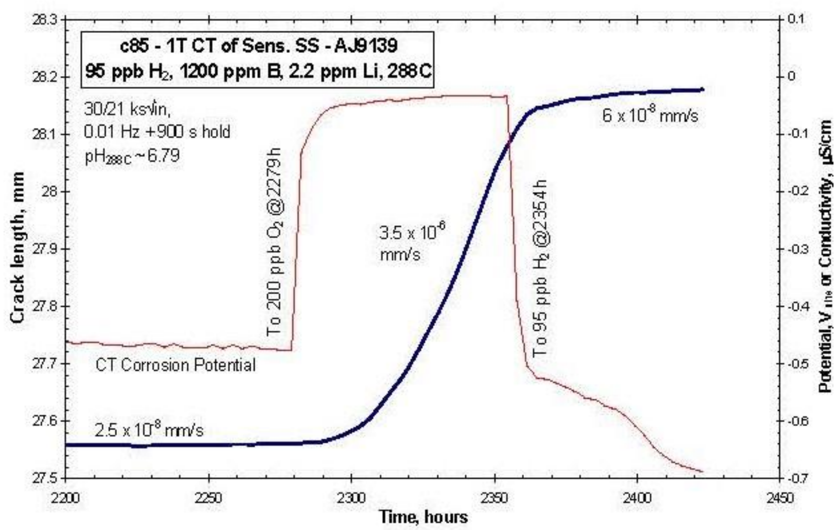

Figure 20. SCC crack length vs. time for sensitized SS in 288 °C showing that the presence of 1200 ppm B as H_3BO_3 and 2.2 ppm Li as LiOH results in a low growth rate until the corrosion potential becomes elevated at 2279 hours by the addition of 200 ppb O_2 [46].

As summarized in Table 2, the Mihama-2 Type 316SS SG safe end HAZ cracking revealed low CGR equivalent to 0.13 mm per year even at a hardness of 250 HV. In support of the failure analysis, 304 and 316L SS were prepared to hardness levels measured between 200-300 HV by cold working to 10–65% reduction in thickness and tested for SCCGR in both deaerated and oxygenated PWR primary water at various temperatures between 200 – 345 °C [47]. As shown in Figure 21, ~2 orders higher SCCGR were observed at all hardness levels tested in oxygenated PWR primary water compared to deaerated water.

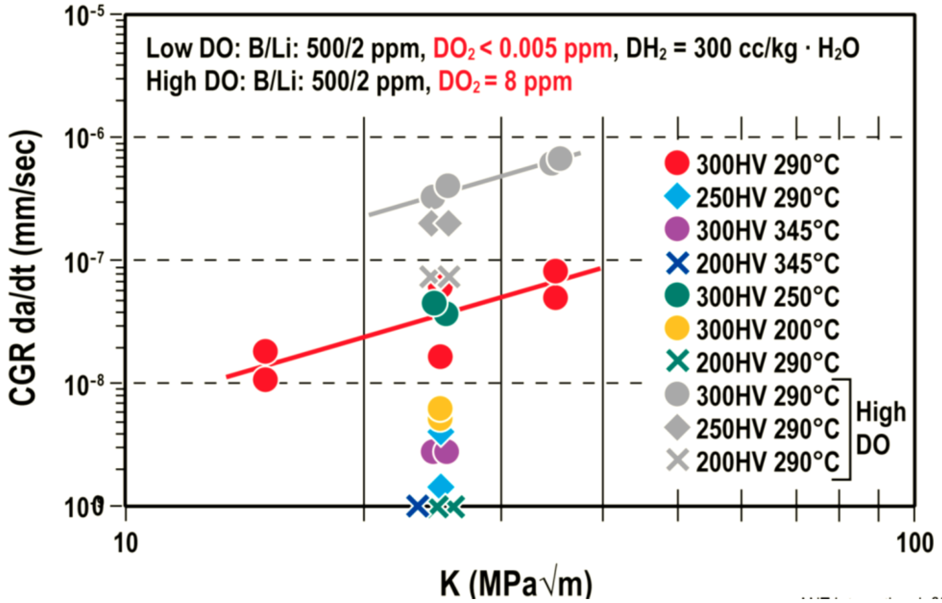

Figure 21. Comparison of SCCGR observed in cold worked Type 304 and 316L SS in normal and aerated PWR primary water [47].

2.2.8.2 pH

In support of the PWR pressurizer heater sheath IGSCC failure as documented in Table 2, Couvant et al. [48] evaluated whether potentially elevated pH in the boiling zone in PWR pressurizer heaters could be a contributing factor to the observed failure in service. They performed a series of SSRT tests on annealed Type 304L SS in 320°C hydrogenated PWR primary water at a pH range between 7 and 9 by adjusting the Li/B ratios in the water. As shown Figure 22, the specimens showed a decreased susceptibility to TGSCC with increasing pH until reaching a weakly alkaline condition ($pH_{320°C} < 7.7$). No SCC susceptibility was observed in the pH range of 7.7–8.7. A substantially increased susceptibility to SCC was revealed at pH >8.7 with a mixed TGSCC + IGSCC morphology until ~50 µm in depth, after which the crack was fully IGSCC, although the crack was confined within the superficial highly cold worked layer of ~50 µm thick. High-resolution analytical microscopy characterizations were performed on the surface oxides formed on the specimens to aid in understanding these observations. Results suggest that oxides formed at different pH shared similar compositions, but the inner oxide layer was substantially thickened at highly alkaline conditions than at a pH between 7–7.7 (100–200 nm vs. 20–50 nm). The thicker oxide was considered less protective, thereby promoting local brittle fracture, leading to crack initiation if the substrate metal is elastically deformed.

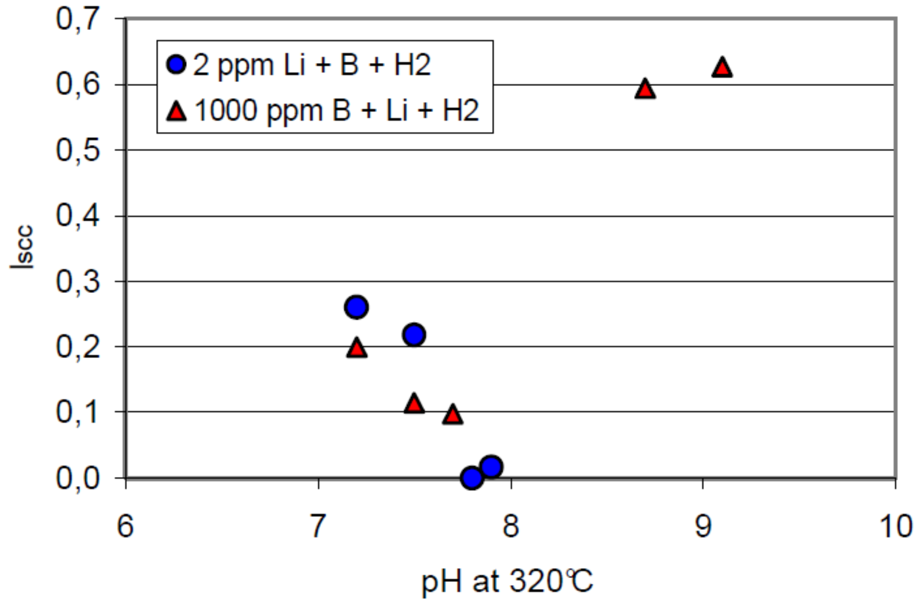

Figure 22. SCC susceptibility versus calculated $pH_{320°C}$ [48].

2.2.8.3 Temperature

No systematic investigation has been conducted on the temperature dependence of SCC initiation for austenitic SS in hydrogenated PWR primary water due to the obvious challenge in initiating SCC discussed above. In comparison, extensive research has been performed on the effect of temperature on the SCCGR of cold worked austenitic SS in the same environments. A broad temperature range of 150–360 °C was evaluated, revealing rather variable apparent activation energies between 60 and 110 KJ/mole, comparable with SSRT measurements. However, there is evidence that the effect of temperature may stabilize or even decrease somewhat between 325 and 360°C, suggesting a non-Arrhenius relationship with SCCGR. In a recent study, Morton et al. [30] demonstrated that heavily cold worked, non-sensitized 304 SS SCCGR follows an Arrhenius temperature functionality with an activation energy of 75 kJ/mol. A deviation from Arrhenius functionality occurs in 304 SS at low levels of cold work and other conditions that reduce the SCCGR propensity due to high-temperature SCCGR retardation (HTR). Sensitization added another complexity to the temperature vs. SCCGR dependence, which has already been discussed in Section 2.2.2.

2.3 Key Knowledge and Technical Gaps

As summarized above, due to the obvious challenge in producing SCC initiation in austenitic SS in nominally good-quality, free-flowing PWR primary water, high-quality SCC initiation data remains scarce and difficult to use for proactive material degradation management. Key knowledge/technical gaps exist in the following areas:

- There is a lack of fundamental understanding of the factors responsible for SCC initiation and the development of very shallow surface cracks into deep propagating cracks. There is also a lack of quantifiable relationships between the influencing factors and SCC initiation time/susceptibility. For example, although field experience and literature data suggest cold work could accelerate SCC initiation, there is little information about threshold yield strength and hardness values required for SCC initiation in PWR primary water under free-flowing conditions. The same issues exist for dissolved oxygen content. The dependency of SCC initiation on temperature is largely unexplored. The contribution of creep to SCC initiation kinetics and mechanism is poorly understood. Heat-to-heat variability is another topic that may have been overlooked. In addition, the synergism between the influencing factors remains to be studied.

- There is a lack of suitable SCC initiation testing techniques that could effectively produce high-quality, reproducible SCC initiation data to be directly used to inform industry practice. At the same time, susceptibility map already exists for IDSCC propagation in PWR primary water [17] and for SCC initiation in BWR normal water chemistry [49]. Most of the existing body of SCC initiation data in PWR primary water is produced by SSRT using severe loading and excessive cold working that are difficult to relate to plant applications. While the NNL's test loop setup of SS elbow and piping specimens offered a closer approximation to plant conditions, the high cost associated with specimen preparation and test operation makes it difficult to repeat by other laboratories. In addition, although the testing is equipped with in-situ monitoring techniques, unsatisfactory resolution due to specimen geometry and severe deformation made them unlikely to fit for detecting precursor events and small crack growth.
- There is a lack of statistically relevant SCC initiation data of austenitic SS in PWR primary water. Such data would be highly desired for industry to develop meaningful predictions on SCC initiation time/likelihood to guide inspection and maintenance strategy development.

3. Planning for SCC Initiation Research of Austenitic SS in PWR Primary Water

The review of operating experience and laboratory testing on SCC initiation of austenitic SS in PWR primary water clearly points out a strong need for a better understanding of influencing factors on SCC initiation in normal and off-normal PWR primary water environments, as well as improved testing methods capable of producing statistically relevant SCC initiation data that can be correlated to plant operating conditions. PNNL's expertise in performing state-of-the-art SCC initiation testing on Ni-base alloy positions us well for addressing such needs. Nevertheless, given the anticipation that producing SCC initiation in austenitic SS under normal PWR primary water at constant load can be extremely difficult, modification to the testing method and/or careful determination of accelerating factors are essential, and trials and errors may be necessary to find optimized solutions. This likely requires a large-scale, multi-year effort beyond the current bandwidth of this task, which is heavily committed to long-term SCC initiation testing of high-Cr, Ni-base alloys. Nevertheless, with the conclusion of the SCC testing on Alloy 82 for evaluating the effect of KOH vs. LiOH-containing PWR primary water in fiscal year (FY) 2023, a mid-sized autoclave will become available for new research in FY 2024. We propose to use this test system to address a rather unexplored concern raised by recent IDSCC incidents reported in French NPPs as a starting point of our SCC initiation research into SS in PWR primary water, with the possibility of expanding to larger scale testing in the next few years. The initial planning will be discussed below. It should be kept in mind that this is only a preliminary plan that is subject to change; we will keep our active engagement with the industry, regulatory, and research community to make sure the scope and research plan are tailored to address the most important concerns that may evolve with time and accumulated knowledge.

3.1 Near-Term Tasks

3.1.1 Task 1: Research in Support of French IDSCC Cause Analysis

As summarized in Section 1.2.3, elevated oxygen is considered a potential contributor to the IDSCC cases found in the SI lines elbow weld HAZ in multiple French NPPs, but its effect on SCC initiation has yet to be systematically studied. Since DO is known to enhance SCCGR of austenitic SS in PWR primary water, it is postulated that it can also greatly increase SCC initiation susceptibility of austenitic SS in similar environments, allowing DCPD-detectable SCC initiation to occur within a realistic time frame (e.g., between a few hundred to a few thousand hours) using current constant load SCC initiation test configuration at PNNL. The first round of testing will start with DO level simulating the worst-case scenario, i.e., air-saturated make-up water with 8 ppm O_2. If meaningful SCC initiation time data can be collected within an acceptable timeframe, the DO level will be reduced in step for two more rounds of testing to evaluate whether a threshold DO exists for continued SCC initiation. The ultimate goal is to establish an SCC initiation susceptibility map between SCC initiation time, DO level, and cold work/applied stress. However, if no SCC initiation data can be collected within a reasonable timeframe during the first round of test, the constant load test method will be revisited and modified in order to initiate cracks more effectively. This will be discussed in more detail in 3.1.3.

3.1.2 Task 2: Material Selection, Acquisition, and Preparation

Because the mid-sized SCC initiation test system can only accommodate up to six cylindrical tensile specimens simultaneously, the initial material selection will limit to one 316L SS only. It will be cold tensile strained to achieve a hardness around ~300 HV with the possibility to include a wider hardness range for expanded investigation on cold work effect.

It will be advantageous to test a 316L SS that has already been extensively tested for CGR, so correlations between SCC initiation and propagation could be made when needed. Currently, communications are underway with a few laboratories and industry partners to see if we could get sufficient material from one such heat.

3.1.3 Task 3: Re-evaluation of SCC Initiation Test Method and Specimen Design

Even in PWR primary water with high DO, there is a risk that SCC initiation may not take place within an acceptable time frame under constant load. In such a case, our current SCC initiation test method may need to be revisited to allow for easier SCC initiation without exerting excessive accelerants that are difficult to gauge with plant conditions. As discussed in Sections 2.2.1 and 2.2.6, multi-axial loading can promote complex strain paths and greatly ease the difficulty in initiating cracks under constant load. It is also believed that multi-axial loading can better simulate the complex operating conditions present in plants. As a result, modification of the test loading method will be centered on enabling multi-axial loading at a controllable load. Currently, multi-axial test setup information is being collected from the literature and several designs are under consideration. Further activities associated with this task will be reported in more detail in future.

3.2 Phase Planning

The tasks described above are planned to be started in the last quarter of FY 2023 with a tentative schedule presented in Table 3 below.

Table 3. Tentative schedule of near-term SCC initiation research of austenitic SS at PNNL.

	FY23 Q4	FY24 Q1	FY24 Q2	FY24 Q3	FY24 Q4
Task 1		Test round #1	Test round #2		Test round #3
Task 2	Mat & sample prep				
Task 3	Info collection	Prototype design & fab		Proof-of-concept test	

REFERENCES

[1] P. Ford, P.M. Scott, P. Combrade, C. Amzallag, Environmentally-Assisted Degradation of Structural Materials in Water Cooled Nuclear Reactors - An Introduction, August 2015.

[2] G.O. Ilevbare, F. Cattant, N.K. Peat, SCC of stainless steels under PWR service conditions, in: Fontevraud 7 - 7th International Symposium on Contribution of Materials Investigations and Operating Experience to LWRs' Safety, Performance and Reliability, 2011.

[3] C. Mayor, EDF Stress Corrosion Cracking Operating Experience Discussion, in Advisory Committeee on Reactor Safeguards (ACRS) Meeting 11/16/2022 on French PWR Safety Injection System Cracking in, 2022. https://www.nrc.gov/docs/ML2214/ML22143A834.pdf

[4] B.M. Gordon, Corrosion and Corrosion Control in Light Water Reactors, JOM, 65 (2013) 1043-1056.

[5] A.J. Sedriks, Corrosion of Stainless Steels, John Wiley and Sons, 1979.

[6] Materials Reliability Program: Stress Corrosion Cracking of Stainless Steel Components in Primary Water Circuit Environments of Pressurized Water Reactors (MRP-236, Rev. 1), 3002009967, 2017.

[7] Materials Reliability Program: Assessment of the Current Status and Completeness of Work on Inner and Outer Diameter Stress Corrosion Cracking of Austenitic Stainless Steels in PWR Plants (MRP-352), 3002000135, 2013.

[8] Kansai Electric Power Company, Ohi-3 Pressurizer Spray Line Pipe Weld Investigation Restul Dataset (大飯発電所 3号機加圧器スプレイライン配管溶接部の調査結果(デ ータ集)), 12/24/2020. https://www.nra.go.jp/data/000338756.pdf

[9] Kansai Electric Power Company, Regarding the cause and countermeasures for the occurrence of the event at the pressurizer spray line pipe weld (加圧器スプレイライン配管溶接部における事象の発生原因における及び対策について), 12/24/2020. https://www.nra.go.jp/data/000338755.pdf

[10] Nouvelles détections de fissures sur des tuyauteries du système d'injection de sécurité des réacteurs n° 1 et n° 2 de la centrale nucléaire de Penly et n° 3 de Cattenom (New detection of cracks on pipes of the safety injection system of reactors n° 1 and n° 2 of the Penly nuclear power plant and n° 3 of Cattenom), March 16th, 2023. https://www.irsn.fr/sites/default/files/2023-03/IRSN_note_info_fissures_16mars2023.pdf

[11] Safety-relevant damage in the safety injection systems of French nuclear power plants, in, 05/19/2022. https://www.grs.de/en/news/safety-relevant-damage-safety-injection-systems-french-nuclear-power-plants

[12] R. Hosler, Auxilliary Piping SCC OE Focus Group Update (Industry Perspectives), in Advisory Committeee on Reactor Safeguards (ACRS) Meeting 11/16/2022 on French PWR Safety Injection System Cracking in, 2022.

[13] T. Couvant, C. Varé, J.M. Frund, Y. Thébault, B. Audebert, E. Lemaire, Susceptibility to IGSCC of cold work austenitic stainless steels in non-polluted primary PWR environment, in: Fontevraud 10 - 10th International Symposium on Contribution of Materials Investigations and Operating Experience to LWRs' Safety, Performance and Reliability, 2022.

[14] Avis IRSN N° 2022-00189, February 14th, 2022. https://www.irsn.fr/sites/default/files/documents/expertise/avis/2022/Avis-IRSN-2022-00189.pdf

[15] Décryptage du phénomène de corrosion sous contrainte identifié sur cinq réacteurs nucléaires (Deciphering the phenomenon of stress corrosion identified on five nuclear reactors), Societe Francaise d'Energie Nucleaire - SFEN, January 18th, 2022. https://www.sfen.org/rgn/decryptage-du-phenomene-de-corrosion-sous-contrainte-identifie-sur-cinq-reacteurs-nucleaires/

[16] P. Ford, P.M. Scott, P. Combrade, Environmentally-Assisted Degradation of Stainless Steels in LWRs, February 2010.

[17] Materials Reliability Program: Stress Corrosion Crack Growth Rates in Stainless Steels in PWR Environments (MRP–458), 3002020451, 2022.

[18] T. Moss, J. Hesterberg, D. Morton, D. Paraventi, Non-Arrhenius Stress Corrosion Crack Growth Behavior of 304 Stainless Steel in Deaerated Water, in: 20th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Association for mAterials Protection and Performance (AMPP), 2022.
[19] Y. Watanabe, T. Kondo, Current and Potential Fluctuation Characteristics in Intergranular Stress Corrosion Cracking Processes of Stainless Steels, Corrosion, 56 (2000).
[20] W. Zhang, L. Dunbar, D. Tice, Monitoring of stress corrosion cracking of sensitised 304H stainless steel in nuclear applications by electrochemical methods and acoustic emission, Energy Materials, 3 (2008) 59-71.
[21] T. Moss, J. Brockenbrough, Development of residual stress driven stress corrosion crack initiation and growth specimens, in: 19th International Conference of Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, American Nuclear Society, 2019.
[22] D. Paraventi, J.R. Hetersberg, T.E. Moss, S.C. Healey, Stress corrosion cracking testing of laboratory processed 304/304L stainless steel elbows and pipe, in: 20th International Conference of Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Association for Materials Protection and Performance (AMPP), 2022.
[23] O. Raquet, E. Herms, F. Vaillant, T. Couvant, J.-M. Boursier, SCC of cold-worked austenitic stainless steels in PWR conditions, Adv. Mater. Sci., 7 (2007).
[24] T. Yonezawa, K. Fujimoto, H. Kanasaki, T. Iwamura, S. Nakada, K. Ajiki, K. Sakai, Stress Corrosion Cracking Susceptibility of 316 Stainless Steel in PWR Primary Water, in: Fontevraud 8 - 8th International Symposium on Contribution of Materials Investigations and Operating Experience to LWRs' Safety, Performance and Reliability, 2014.
[25] T. Couvant, L. Legras, F. Vaillant, J.M. Boursier, Y. Rouillon, Effect of strain-hardening on stress corrosion cracking of aisi 304L stainless steel in PWR primary environment at 360°C, in: 12th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, 2005, pp. 1069-1081.
[26] G.O. Ilevbare, T. Couvant, Effect of localized strain deformation on SCC initiation of stainless steels in simulated PWR primary water, in: NACE - International Corrosion Conference Series, 2014.
[27] Program on Technology Innovation: Interactions between Strain Localization and Environmentally Assisted Cracking in Austenitic Alloys, 1022816, 2011.
[28] N. Ishiyama, M. Mayuzumi, Y. Mizutani, J.I. Tani, Stress corrosion cracking of type 316 and 316L stainless steels in high temperature water, in: 12th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, 2005, pp. 57-64.
[29] C.D. Lundin, C.H. Lee, R. Menon, E.E. Stansbury, Sensitization of Austenitic Stainless Steels; Effects of Welding variables on HAZ Sensitization of AISI 304 and HAZ Behavior of BWR Alternative Alloys 316NG and 347, No. 319, November 1986.
[30] D. Morton, E. West, T. Moss, G. Newsome, The Temperature Functionality of Sensitized Stainless Steel SCC Growth in Deaerated Water, in: 20th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Association for Materials Protection and Performance, 2022.
[31] N. Matsubara, T. Kobayashi, K. Fujimoto, Y. Nomura, N. Chigusa, S. Hirano, Research program on SCC of cold-worked stainless steel in Japanese PWR N.P.P., in: Fontevraud 7 - 7th International Symposium on Contribution of Materials Investigations and Operating Experience to LWRs' Safety, Performance and Reliability, 2010.
[32] X. Zhong, S.C. Bali, T. Shoji, Effects of dissolved hydrogen and surface condition on the intergranular stress corrosion cracking initiation and short crack growth behavior of non-sensitized 316 stainless steel in simulated PWR primary water, Corrosion Science, 118 (2017) 143-157.
[33] L. Chang, M.G. Burke, F. Scenini, Stress corrosion crack initiation in machined type 316L austenitic stainless steel in simulated pressurized water reactor primary water, Corrosion Science, 138 (2018) 54-65.

[34] L. Chang, L. Volpe, Y.L. Wang, M.G. Burke, A. Maurotto, D. Tice, S. Lozano-Perez, F. Scenini, Effect of machining on stress corrosion crack initiation in warm-forged type 304L stainless steel in high temperature water, Acta Materialia, 165 (2019) 203-214.
[35] D. Tice, Contribution of research to the understanding and mitigation of environmentally assisted cracking in structural components of light water reactors, Corrosion Engineering, Science and Technology, 53 (2018) 11-25.
[36] F. Scenini, J. Lindsay, L. Chang, Y.L. Wang, M.G. Burke, S. Lozano-Perez, G. Pimentel, D. Tice, K. Mottershead, V. Addepalli, Oxidation and SCC Initiation Studies of Type 304L SS in PWR Primary Water, in: 18th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, Springer International Publishing, 2019, pp. 793-810.
[37] J. Talonen, H. Hänninen, Formation of shear bands and strain-induced martensite during plastic deformation of metastable austenitic stainless steels, Acta Materialia, 55 (2007) 6108-6118.
[38] L. Chang, M.G. Burke, K. Mukahiwa, J. Duff, Y. Wang, F. Scenini, The effect of martensite on stress corrosion crack initiation of austenitic stainless steels in high-temperature hydrogenated water, Corrosion Science, 189 (2021) 109600.
[39] E.A. West, C. Tackes, G. Newsome, N. Lewis, Influence of sulfur and ferrite on SCC and corrosion fatigue behavior of model heats of stainless steel, in: 17th International Conference of Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Canadian Nuclear Society, 2015.
[40] Personal communication with K. Sakima and T. Maeguchi.
[41] S. Nouraei, D.R. Tice, K.J. Mottershead, D.M. Wright, Effects of thermo-mechanical treatments on deformation behavior and IGSCC susceptibility of stainless steels in PWR primary water chemistry, in: 15th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, 2011, pp. 2250-2270.
[42] W. Zhang, X. Wang, S. Wang, H. Wu, C. Yang, Y. Hu, K. Fang, H. Jiang, Combined effects of machining-induced residual stress and external load on SCC initiation and early propagation of 316 stainless steel in high temperature high pressure water, Corrosion Science, 190 (2021) 109644.
[43] K. Arioka, T. Yamada, T. Terachi, G. Chiba, Cold work and temperature dependence of stress corrosion crack growth of austenitic stainless steels in hydrogenated and oxygenated high-temperature water, Corrosion, 63 (2007) 1114-1123.
[44] Z. Zhai, M. Toloczko, K. Kruska, S. Bruemmer, Precursor Evolution and Stress Corrosion Cracking Initiation of Cold-Worked Alloy 690 in Simulated Pressurized Water Reactor Primary Water, Corrosion, 73 (2017) 1224-1236.
[45] Z. Zhai, M. Olszta, M. Toloczko, S. Bruemmer, Crack initiation behavior of cold-worked alloy 690 in simulated PWR primary water – Role of starting microstructure, applied stress and cold work on precursor damage evolution, in: 19th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, 2019, pp. 373-385.
[46] P.L. Andresen, M.M. Morra, Emerging issues in environmental cracking in hot water, in: 13th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Canadian Nuclear Society, 2007, pp. 1414-1438.
[47] K. Fujimoto, T. Iwamura, S. Suzuki, T. Kobayashi, Y. Kikuchi, SCC Growth rate of cold worked austenitic stainless steel in PWR environment, in: AECL/COG/EPRI Workshop: Effects of Cold Work on Stress Corrosion Cracking of Materials in Water Cooled Nuclear Plants, 2007.
[48] T. Couvant, L. Legras, C. Pokor, F. Vaillant, Y. Brechet, J.M. Boursier, P. Moulart, Investigations on the mechanisms of PWSCC of strain hardened austenitic stainless steels, in: 13th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Canadian Nuclear Society, 2007, pp. 499-514.
[49] B.M. Gordon, D.E. Delwiche, G.M. Gordon, Service experience of BWR pressure vessels, in: Vessels and Piping Division (Publication) PVP, American Society of Mechanical Engineers, 1987, pp. 9-17.