B.2 "IASCC of Stainless Steels and Other Irradiation Induced Phenomena," by Peter L. Andresen and Peter M. Scott

This background paper provides a foundation for understanding the proactive materials degradation concerns originating from irradiation effects. The emphasis is on irradiation assisted stress corrosion cracking (IASCC) of ***wrought, austenitic stainless steels*** in BWR and PWR environments. Other radiation induced phenomena that are discussed include radiation creep relaxation, swelling and microstructural evolution. There are separate papers for SCC in unirradiated stainless steels, in cast stainless steels, and for lower temperature, mostly chloride related pitting and SCC of stainless steels. There are also separate papers related to BWR and PWR water chemistry, evolving operational practice, startup and shutdown, and other factors that influence SCC.

The paper on SCC of unirradiated stainless steels provides an introduction to the mechanisms, processes and dependencies in high temperature water. The primary factors that control SCC of stainless steels in hot water [1-8]–many of which are affected by radiation – include:

- Degree of sensitization, i.e., Cr depletion along the grain boundaries.
- Oxidants and corrosion potential, which affect the crack chemistry as well as the nature of the oxide films on the free surfaces.
- Water purity and pH, which primarily affects the crack chemistry.
- Yield strength, which produces an increase in crack growth rate. There are many ways by which yield strength is increased, including bulk or surface cold work, weld residual strain, precipitation hardening, etc., but not usually to the same degree as is caused by irradiation.
- Temperature.
- Stress and Stress Intensity Factor.

It is widely accepted that irradiation assisted SCC (IASCC) is literally that: an *irradiation assisted* process [2,9-19]. When viewed in a given time frame in plant components (Figure B.2.1a) or in accelerated laboratory tests, there can appear to be a threshold fluence for IASCC, but in fact SCC is observed in unirradiated stainless steels [2-5,9,15,16,20-22]. Irradiation is known to affect primarily the grain boundary chemistry (i.e., degree of sensitization), the oxidants and corrosion potential, the yield strength and the stress (via irradiation creep relaxation) components in this list of factors. In sufficiently careful and sensitive laboratory tests (e.g., crack growth rate tests), all grades of austenitic stainless steel have been shown to have inherent susceptibility to SCC. However, the numerous factors that promote SCC give rise to orders of magnitude difference in susceptibility, i.e., the incidence of cracking. Importantly, the effects of most parameters, such as corrosion potential, water impurities, stress, stress intensity factor, temperature, etc. are known to have a similar effect on both irradiated and unirradiated stainless steels.

The effect of corrosion potential (Figure B.2.1b) and water purity (Figure B.2.2) is similar for unirradiated and irradiated stainless steels exposed in BWR environments, which supports the concept that the underlying mechanisms and dependencies are similar. While the term "threshold fluence" appears in the literature, it should be recognized that unirradiated (and unsensitized, un-cold-worked) materials have some small susceptibility to SCC, and an apparent "threshold fluence" depends strongly on the details of other controlling parameters, such as the environment, loading, cold work, temperature, etc. Thus, a "threshold fluence" has relevance primarily from an engineering perspective within a specific context of environment, loading, etc. [5,9,11]

Figure B.2.1 shows the increasing SCC incidence with increasing fast neutron fluence in BWR crevice control blade sheath and in laboratory slow strain rate tests. While small amounts of intergranular cracking have been observed in tests in inert environments on irradiated stainless steels, there is an incontestable and dominant aqueous environmental effect. Thus, the concerns for cracking in irradiated components are appropriately characterized as IASCC, not as a simple effect on mechanical properties [2,9-19].

The radiation dose achieved in various components and the onset of various radiation effects is shown in Figure B.2.3 [11]. Most aspects of IASCC are well understood qualitatively, and a good quantitative description seems to exist in BWR water chemistry and temperature regime, but it is not completely clear that all of the aggravating effects of radiation on SCC are identified or qualified for all light water reactor conditions, esp. at the higher temperatures and fluences in PWRs. For all systems, the following factors are known to be important (Figure B.2 .4):

I. **Radiation hardening** (RH), in which the radiation generated defects produce an increase in yield strength (and a localization of deformation to “channels” in the material). Figure B.2.5 shows the increase in yield strength of a variety of austenitic stainless steels vs. irradiation dose. An increase in the yield strength from 150 – 200 MPa up to ≈ 700 – 1000 MPa is commonly observed, with a saturation after several dpa. Cold worked materials have a higher initial yield strength, but follow a broadly similar trajectory vs. dose. Much of the microstructural evidence of the initial cold worked microstructure has vanished after about 5 dpa.

    The increase in yield strength results primarily from the formation of vacancy and interstitial loops (Figure B.2.6). Source hardening and dispersed barrier hardening models provide reasonable correlations between hardening and the dislocation loop microstructure, with the increase in yield strength (or hardness) proportional to $(N_{loop} \times d_{loop})^{0.5}$, where N_{loop} is the loop number density and d_{loop} is the loop diameter.

    The effect of yield strength on SCC growth rates is discussed in the topical paper on unirradiated austenitic stainless steels, and appears to be a common effect among many materials and many mechanisms of yield strength enhancement (cold work, martensite formation, irradiation, precipitation hardening, etc.). Growth rates are increased in both BWR and PWR chemistries.

    The homogeneous nature of deformation at low dose is replaced by heterogeneous deformation at higher doses as the defect microstructure impedes the motion of dislocations. Initial dislocations clear defects along narrow channels, and plasticity becomes highly localized. The channels are very narrow (< 10 nm) and closely spaced (<1 mm) and typically run the full length of a grain, terminating at the grain boundaries. Dislocation channeling results in intense shear bands that can cause localized necking and a sharp reduction in uniform elongation, but the reduction in area generally remains very high. Dislocation channeling may also be an important in IASCC [11,19].

II. **Radiation induced segregation** (RIS), in which the migration of radiation generated defects (vacancies and interstitials) to sinks (esp. grain boundaries), alters the local chemistry within the material. Figure B.2.7 shows two examples of the grain boundary composition of high purity and commercial purity heats of stainless steel. The enrichment or depletion of major alloying elements and impurity elements can be significant [2,9-19], with depletion of >5% Cr and enrichment of Si by >5-10X often observed [2,9-22].

RIS is driven by the flux of radiation-produced defects to sinks, and is therefore fundamentally different from thermal segregation or elemental depletion from grain boundary precipitation processes (e.g., sensitization from Cr carbide or boride formation and growth). In simple terms, radiation displaces an atom from its lattice site, and it comes to rest in a relatively distant location in an interstitial site. In fact, this primary displaced atom itself interacts with other atoms along its path, producing a cascade of damage as it loses energy and comes to rest. The resultant vacancies and interstitials can reach concentrations that are orders of magnitude greater than the thermal equilibrium concentrations. They migrate and are absorbed at sinks, creating profiles in concentrations of the constituent elements near grain boundaries. The species that diffuse more slowly by the vacancy diffusion mechanism are enriched at the grain boundary and the faster diffusers become depleted. Enrichment and depletion can also occur by association of the solute with the interstitial flux. In this case, the undersized species will enrich and the oversized species will deplete.

Even though the depletion and enrichment profiles are very narrow compared to those that form from, e.g., Cr carbide formation during welding or heat treatment, the effect on SCC remains very pronounced. For example, very narrow Cr depletion profiles can be generated during complex, multi-step heat treatments, and for a given level of Cr depletion, they have as strong an effect on SCC in high temperature water as much wider Cr depletion profiles.

Si enrichment is potentially of great concern because many stainless steels containing 0.5 – 1% Si can enrich to >5% at the grain boundary. Indeed, since the measurements are generally made by analytical electron microscopy, which has a 1 – 2 nm beam size, the actual Si concentration at the grain boundary can approach 50 atomic percent. Crack growth rate measurements on stainless steels with elevated Si levels (e.g., 1.5 – 5% Si) show high growth rates and limited or no effect of stress intensity factor and corrosion potential (Figure B.2.8) [20-22]. This may help explain the loss of the benefit of lowering the corrosion potential at high fluence in some stainless steels, esp. since Si enrichment appears to continue after Cr depletion saturates.

III. **Radiation creep relaxation**, in which the migration of radiation generated defects under stress produces an accelerated creep rate (e.g., at constant load) and/or stress relaxation (e.g., at constant displacement, as for weld residual stresses, bolts and springs). Figures B.2.9 and 10 show two examples of radiation creep relaxation, which produces a large reduction in stress after a dose of several dpa. The radiation creep rate is proportional to the dose rate (flux) and stress.

Radiation creep relaxation is a mixed benefit. For welds, the weld residual stress is significantly relaxed in the same range of fluence where radiation hardening and segregation occur, and the net effect is generally beneficial. However, in many bolting application, the loss of stress over time can cause other problems related, e.g., to inadequate clamping forces that allow leakage that can produce erosion or fatigue. Because radiation creep inherently represents deformation, it can also promote SCC nucleation and help sustain crack growth. The radiation creep rates are very small compared to other sources (e.g., cyclic loading, slow strain rate testing, and strain redistribution at the tips of SCC cracks), so there is no evidence or expectation that growth rates will be elevated. However, the low rates of continuous deformation resulting from radiation creep may promote crack nucleation and help sustain crack advance.

Radiation creep provides a good example of a complicating factor in understanding the effects of radiation. Because radiation affects grain boundary chemistry and increases yield strength while it simultaneously reduces the stress near welds and in bolts, understanding and deconvoluting the effects of radiation on SCC is very difficult to do from field data. Add to this the effects of plant operating conditions (such as having high impurity levels early in BWR life), and it becomes very difficult to use plant data as a basis for understanding the real affects of various parameters on SCC, or to anticipate the response of one component (e.g., with constant displacement stresses) with others (e.g., with active pressurization stress). This is an example of the importance of developing a fundamental framework from which hypotheses can be formulated and tested. Such an approach has been undertaken, and even twenty years after the original hypotheses, this framework still represents the basis for current understanding of irradiation effects on SCC. While some improvements in quantifying some aspects of irradiation effects on yield strength, corrosion potential, radiation induced segregation and radiation creep relaxation could undoubtedly be made, there is a strong basis for both understanding and predicting radiation effects on SCC.

IV. **Radiolysis**, in which H_2O is broken into various constituent elements, including H_2O_2 and H_2 (the longer lived species) as well as radicals (e.g., e_{aq}^-, H, OH, HO_2). While stoichiometric quantities of oxidizing and reducing species are formed, the corrosion potential inevitably increases, sometimes dramatically. Radiolysis is suppressed at coolant H_2 levels above about 500 ppb (5.6 cc/kg), so there is little concern for radiolysis in PWRs (whose coolant H_2 level is typically 25 – 35 cc/kg).

In BWRs, the primary radiolytic species of interest are H_2 and H_2O_2. H_2 preferentially partitions to the steam phase, while H_2O_2 remains in the recirculating water, creating a net oxidizing environment. The effect of these (and other) species on SCC is accurately characterized by their effect on corrosion potential. The corrosion potential on most structural materials is similar in deaerated water, and drops by about 57 mV per 10X increase in H_2 and 114 mV per unit increase in pH at 300 °C. As soon as even very small amounts of oxidants are present (e.g., ppb levels), the corrosion potential can rise dramatically, generally increasing by 500 mV or more at >10 ppb of oxidant. Most importantly, when oxidants change the corrosion potential, a differential aeration cell forms, which produces an altered crack chemistry – this does not occur if only H_2 is present because it is not consumed in cracks (as is H_2O_2 and O_2).

Concerns have been expressed that radiolysis could produce oxidizing conditions within cracks, and thereby alter the corrosion potential, mass transport processes, and SCC. However, an evaluation of the corrosion potential in a tight crevice under highly irradiated conditions showed no consequential elevation in corrosion potential (e.g., < 25 mV).

V. **Radiation induced swelling**, in which voids form within the material that produce a change in material density and dimensions. This can produce distortion and warping, which can in turn produce elevation in stresses, e.g., in bolted structures. The occurrence of swelling in austenitic stainless steels is very rare and/or limited below 310 °C, even at high fluence (>30 dpa). Gamma heating of thick components can produce perhaps a 40 – 50 °C elevation in internal temperature, and at such temperatures swelling is more likely at moderate to high fluence.

One possible area of significance for void swelling is in PWR baffle plates and bolts. Because the bolts are fabricated from cold worked stainless steel, swelling is delayed compared to the adjacent annealed plates. Radiation creep relaxation will reduce the stress applied by the bolts, but differential swelling of the plates relative to the bolts will cause reloading of the bolts, which achieves some dynamic equilibrium (between reloading from differential swelling and ongoing radiation creep relaxation). This is difficult to quantify precisely at this time.

VI. **Gamma heating** has already been mentioned in relation to swelling. Another possible consequence of gamma heating is superheating of crevices in PWRs above the temperature of the pressurizer (for example crevices between the shanks of baffle bolts and baffle plates). Thus local boiling with consequent changes in environmental chemistry can occur. Although there is no hard evidence that this has caused any environmentally induced cracking, the phenomenon cannot be ignored when searching for contributing factors in service failures.

VII. **Fracture toughness** is reduced substantially in irradiated stainless steels. There is substantial scatter in the available data, but many stainless steels drop by a factor of five or more from 250 – 300 MPa√m to 50 MPa√m or even slightly lower (Figure B.2.11). These are also data obtained in air, and there may be further environmental degradation in fracture toughness in the environment, both at 288 – 323 °C and in the 75 – 140 °C regime [23].

Predictability of IASCC

A solid qualitative understanding and at least semi-quantitative predictive capability exists for IASCC, esp. for BWR water chemistries and temperatures (there may be additional aggravating effects of radiation that become important at the higher temperatures and fluences in PWRs). The crack growth predictive capability is built on the basis of irradiation *assisted* SCC – that is, the understanding and predictive framework used for unirradiated stainless steels can be extended to radiation effects by defining key characteristics of SCC and quantifying those effects. This has been done for the four primary radiation induced phenomena: segregation, hardening, radiolysis and relaxation (Figure B.2.4) [2,9,15,17,18]. Figure B.2.12 shows the crack growth response at high and high corrosion potential of irradiated stainless steel and of sensitized stainless steel exposed to high and flow neutron fluxes. These (and other) data are replotted in Figure B.2 13 to show crack velocity vs. corrosion potential. Other examples of predictive capability are shown in Figure B.2.14, which shows the response of neutron irradiated stainless steel in slow strain rate and constant load tests in 288 °C water. As in many SCC systems, obtaining high quality, reproducible, consistent SCC data experimentally is often a limiting factor in quantifying and validating predictive models.

The effect of individual changes (such as flux, fluence, temperature, radiolysis, segregation, hardening, relaxation, etc.) cannot be viewed in isolation in most experiments, and rarely if ever in plant components. For example, the temperature of irradiation, the presence of stress, the radiation dose rate (flux), etc. can all affect the result at a given dose / fluence. It must also be recognized that there is a time-based evolution in radiation damage (Figure B.2.15), and these produce complex changes in predicted and observed response. In most components that undergo IASCC, there are damaging elements of radiation exposure (e.g., hardening and segregation) and beneficial elements (radiation creep relaxation of constant displacement stresses). There are then further complications when considering plant operation and the evolution of cracks. For example, if the water chemistry in BWRs is good, so that cracks don't nucleate (or remain vanishingly small) by a dose of 1 – 3 x 10^{21} n/cm^2, then the weld residual stresses will have markedly decreased and the likelihood that SCC will occur also decreases markedly. Figure B.2.15 shows an example of this interaction in terms of the predicted difference in crack growth trajectory vs. time for different water purities in a BWR core shroud. Figure B.2.16 shows an example of crack length vs. time predictions for a type 304 stainless steel BWR core shroud with multiple inspections and multiple cracks, and a comparison of observed and predicted crack depth for a number of BWR core shrouds.

IASCC in baffle bolts has also been evaluated and some controlling factors identified [24], although the state of knowledge does not yet permit prediction.

IASCC Mitigation

There are a variety of approaches for mitigating SCC in light water reactors, and they fall into categories of water chemistry, operating guidelines, new alloys, stress mitigation and design issues. Since most components in light water reactors are not intended to be replaceable (and are therefore very expensive to replace), water chemistry is the most attractive mitigation strategy, with operating guidelines and perhaps stress mitigation providing more limited opportunities. While the focus of this paper is on IASCC, most mitigation approaches (esp. water chemistry) are applicable to both irradiated and unirradiated components.

Water chemistry mitigation approaches are the easiest to implement and can often provide mitigation to many areas and components in the plant. In BWRs, the focus is primarily on lowering the corrosion potential, which can be done with H_2 injection, but is more effectively achieved using NobleChem™ [25-27]. In both BWRs and PWRs, the addition of Zn appears to provide some crack growth rate benefit for stainless steels, although more work is needed. Similarly, improvements in surface finish, stresses, etc. are effective in both reactor types.

Alloying with oversized elements reduces the extent of radiation-induced segregation (esp. Cr depletion) [19], but it's not clear that it will reduce Si enrichment. Cr depletion is less important in BWRs at low potential and in PWR primary water, but low potentials cannot be achieved in all locations in a BWR (it requires stoichiometric excess H_2 in the water, which doesn't exist in areas where boiling occurs). Radiation hardening differs somewhat among stainless steel types and heats, but it's not clear that it can be changed sufficiently to make an adequate difference in SCC response. Slip localization may aggravate SCC, and there are alloying approaches for altering stacking fault energy which influences slip localization [19]. Operationally, it is always wise to avoid higher stresses, vibration, start up and shutdown, fatigue (e.g., from mixing of cold and hot water, which has increased in low leakage core configurations in PWRs), etc. The timing of H_2O_2 injection during PWR cooling and deaeration and H_2 injection during PWR heat up may be important. In BWRs, the early injection of H_2 during start up, and maintaining H_2 injection close to 100% of the time during operation should reduce SCC.

IASCC – Concerns and Emerging Issues

There remains a number of uncertainties and emerging concerns in the area of IASCC. The uncertainties arise in part from the huge scatter in data that has been obtained on irradiated stainless steel, much of which is caused by weaknesses in the experimental techniques. While factors such as good control and monitoring of water chemistry, transitioning from transgranular fatigue to intergranular SCC morphology, and similar concerns exist, perhaps the biggest issue is associated with K-size validity for crack growth specimens of irradiated materials [16,28]. There remain some concerns for the prospect of additional radiation related degradation (such as precipitation of new phases, high He and void swelling) at higher temperatures and fluences associated with PWR components. As in all materials / systems, the understanding and prediction of crack nucleation is much weaker than for crack growth.

Among the emerging concerns is the role of Si, which appears to continue to segregate under irradiation at fluences where Cr depletion has effectively saturated. This may occur because Si undergoes radiation-induced segregation by a different (or an additional) mechanism than does Cr. Evidence of highly elevated (> 3%) Si levels in irradiated stainless steels and very pronounced effects of Si on crack growth rate are a significant concern, esp. since many stainless steels have a nominal Si level of 0.6 – 1%. Crack growth rate studies show elevated growth rates and a limited effect of stress intensity factor or corrosion potential (Figure B.2.8) [20-22]. Si readily oxidizes and is quite soluble in high temperature water – indeed, it is typically present in BWR (and probably PWR) water at levels about 100X higher than other impurities (typically 100 – 1000 ppb). It does not affect conductivity because is dissolves primarily in non-ionic form.

Another concern is that role of increasing stress intensity factor (K) as the crack grows (dK/da) [29]. Because $K \propto \sigma\sqrt{a}$ (stress times the square root of crack depth), and because the weld residual stress profile changes vs. crack depth, there is usually a large positive dK/da early in the crack growth process. K also changes when the crack is longer, but the magnitude of the +dK/da or –dK/da is smaller. Unfortunately, most studies have been performed using a fixed change in load or displacement vs. time (similar to dK/dt), but this yields non-conservative response since it does not produce the accelerating effect of *positive feedback* as the crack begins to grow faster, causing K to increase faster, causing the crack to grow faster… Conversely, with decreasing dK/da, as the crack slows, the rate of change of K slows, causing further slowing in the crack growth rate… dK/dt fails to provide the important feedback between the rate of change of K and the rate of crack growth, and tends to produce crack arrest. Examples of this are shown in the topical paper on SCC unirradiated stainless steel and in reference [29].

Finally, fracture toughness data obtained in situ (after prolonged exposure to high temperature water) [21] might be substantially lower than the vast majority of available data, all obtained in air. The reduction in toughness from irradiation might be broadly representative of cold worked stainless steel (Figure B.2.11), and both may show significant effect of the environment, both in 288 C and ~100 °C tests, and in tearing resistance (e.g., J-R tests) and impact loading (e.g., Charpy or K_{IC}).

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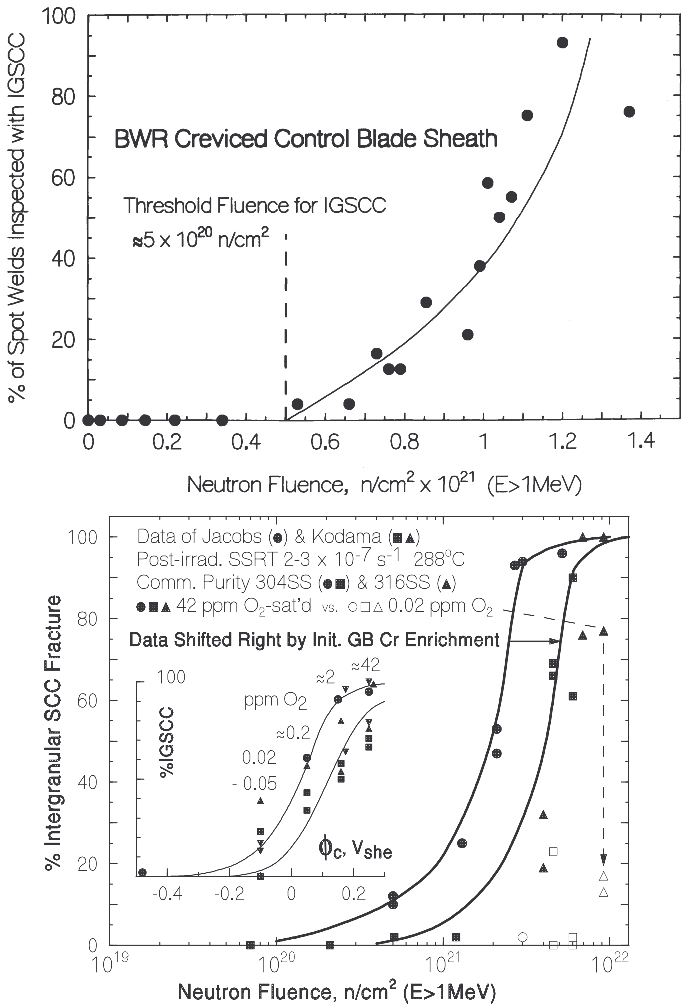


Figure B.2.1 Dependence of IASCC on fast neutron fluence for (a) creviced control blade sheath in high conductivity BWRs and (b) as measured in slow strain tests at 3.7 x 10^7 s^{-1} on pre-irradiated type 304 stainless steel in 288C water. The effect of corrosion potential via changes in dissolved oxygen is shown at a fluence of ≈ 2 x 10^{21} n/cm^2. The effect of corrosion potential on unirradiated and irradiated materials is similar under BWR conditions [2,9,15]. *(© NACE International 1995)*

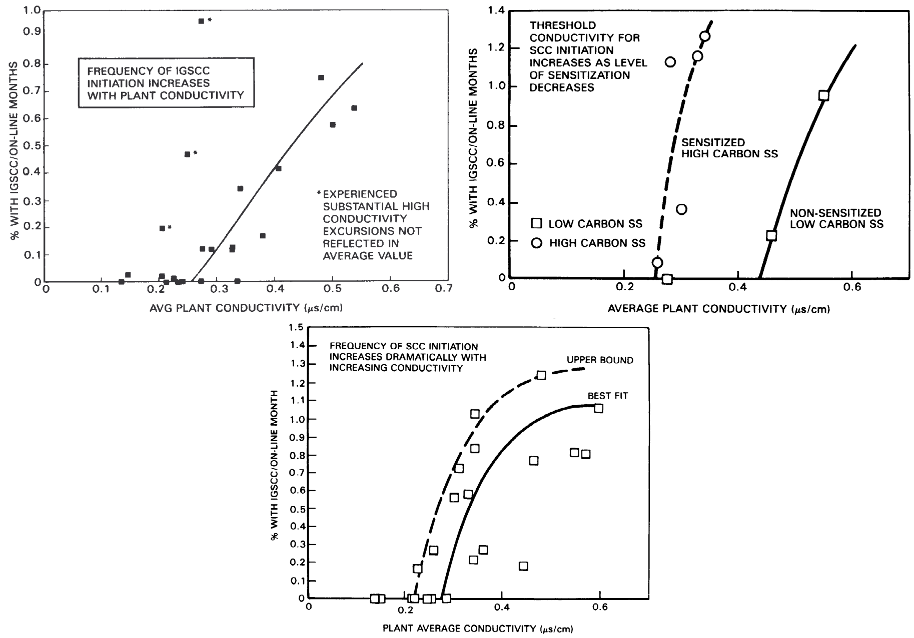


Figure B.2.2 The effects of average plant water purity are shown in field correlations of the core component cracking behavior for (a) stainless steel IRM/SRM instrumentation dry tubes, (b) creviced stainless steel safe ends, and (c) creviced Inconel 600 shroud head bolts, which also shows the predicted response vs. conductivity. The effect of conductivity on unirradiated and irradiated materials is similar under BWR conditions [2,9,15]. *(© NACE International 1990)*

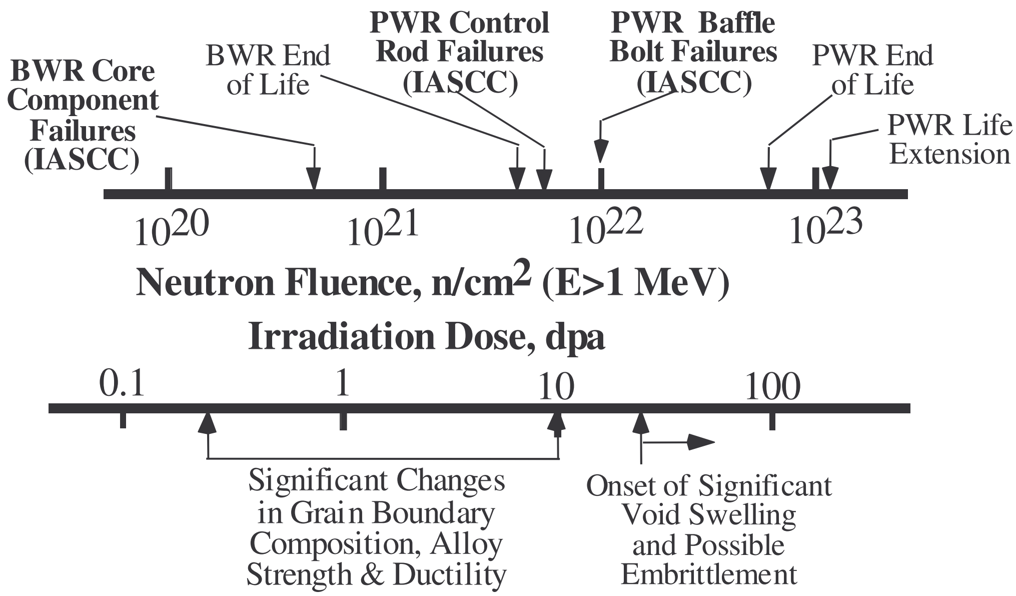
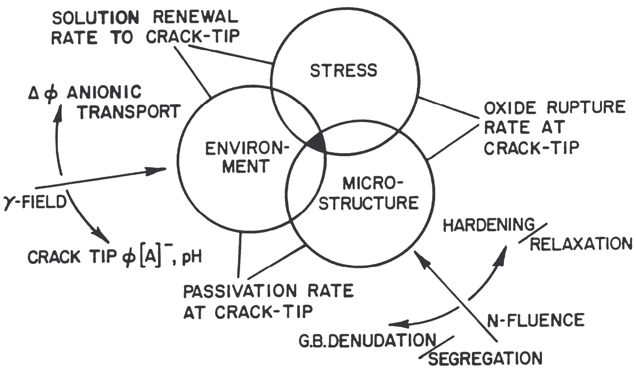


Figure B.2.3 Neutron fluence effects on irradiation-assisted stress corrosion cracking susceptibility of type 304SS in BWR environments [11]. *(Reprinted with Permission from Elsevier)*



Crack depth vs. time = f (Material, Stress, Environment)

Chromium	Primary	Corrosion Potential
% Cold Work	Static	Oxidizing Species
Neutron Fluence	Dynamic	Anionic Species
Silicon	Residual	Type
Nickel	Crack Depth	Activity
Sulfur	Neutron Fluence	Temperature
Phosphorus	Relaxation	Flow Rate
Carbon	Thermal	Neutron flux
Weld heat input	Radiation	Crack Depth

Figure B.2.4 Schematic diagram of the engineering parameters (stress, environment and microstructure), underlying scientific processes (mass transport, oxide rupture, and repassivation rates) and effects of radiation. The complexity of SCC is reflected in the large number of influential variables and the associated requirement that all 20 to 40 in a given system be adequately controlled, all of which are inter-dependently affect SCC [2,9,15]. *(© NACE International 2002)*

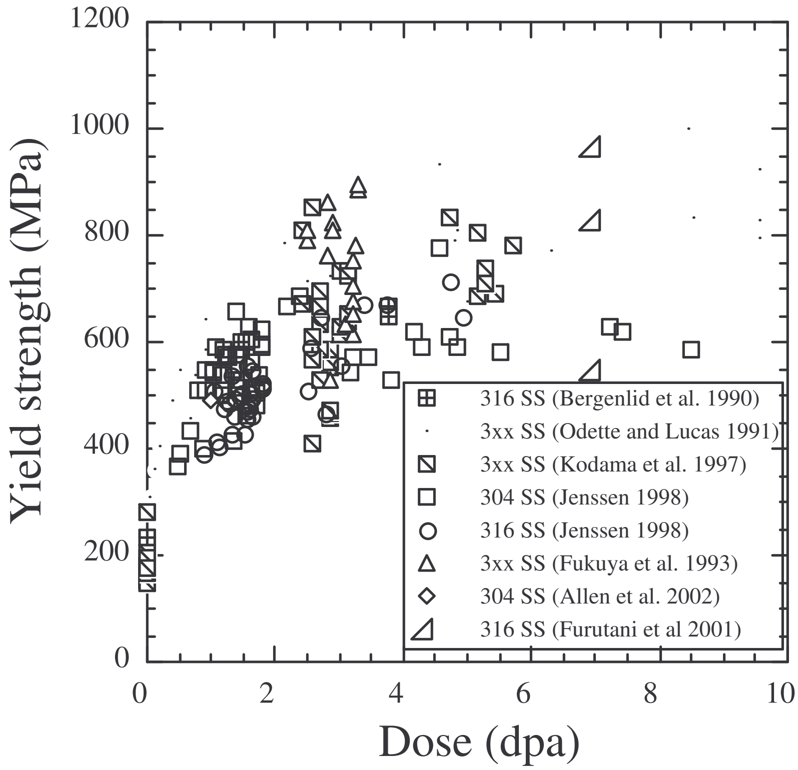


Figure B.2.5 Irradiation dose effects on measured tensile yield strength for several 300-series stainless steels, irradiated and tested at a temperature of about 300℃ [11,19]. *(Reprinted with Permission from Elsevier)*

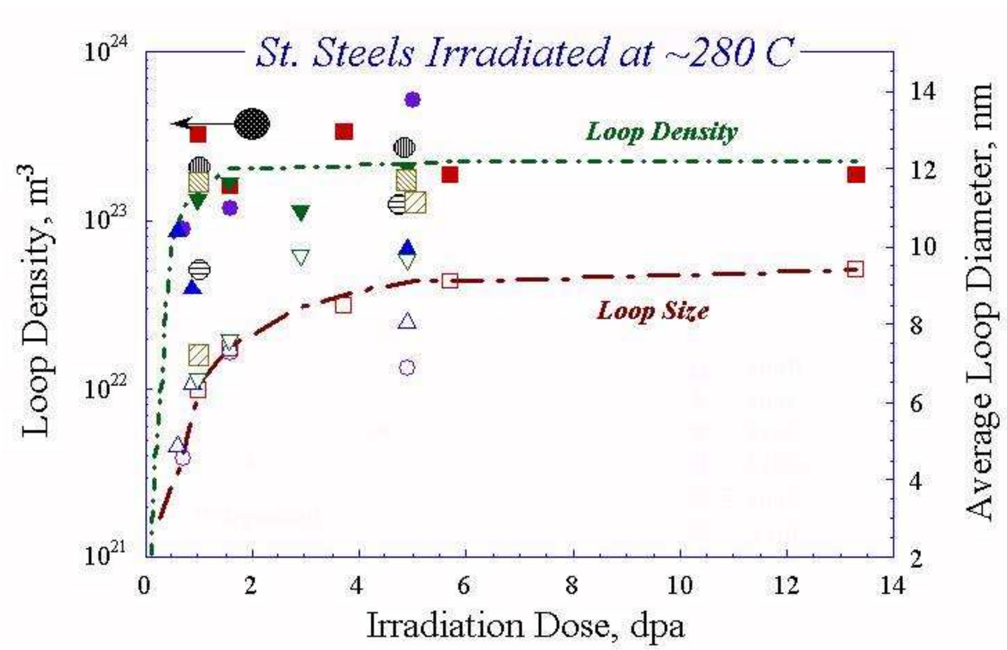


Figure B.2.6 Irradiation dose effects on the measured loop diameter and density for austenitic stainless steels at 280℃ [11,19]. *(Reprinted with Permission from Elsevier)*

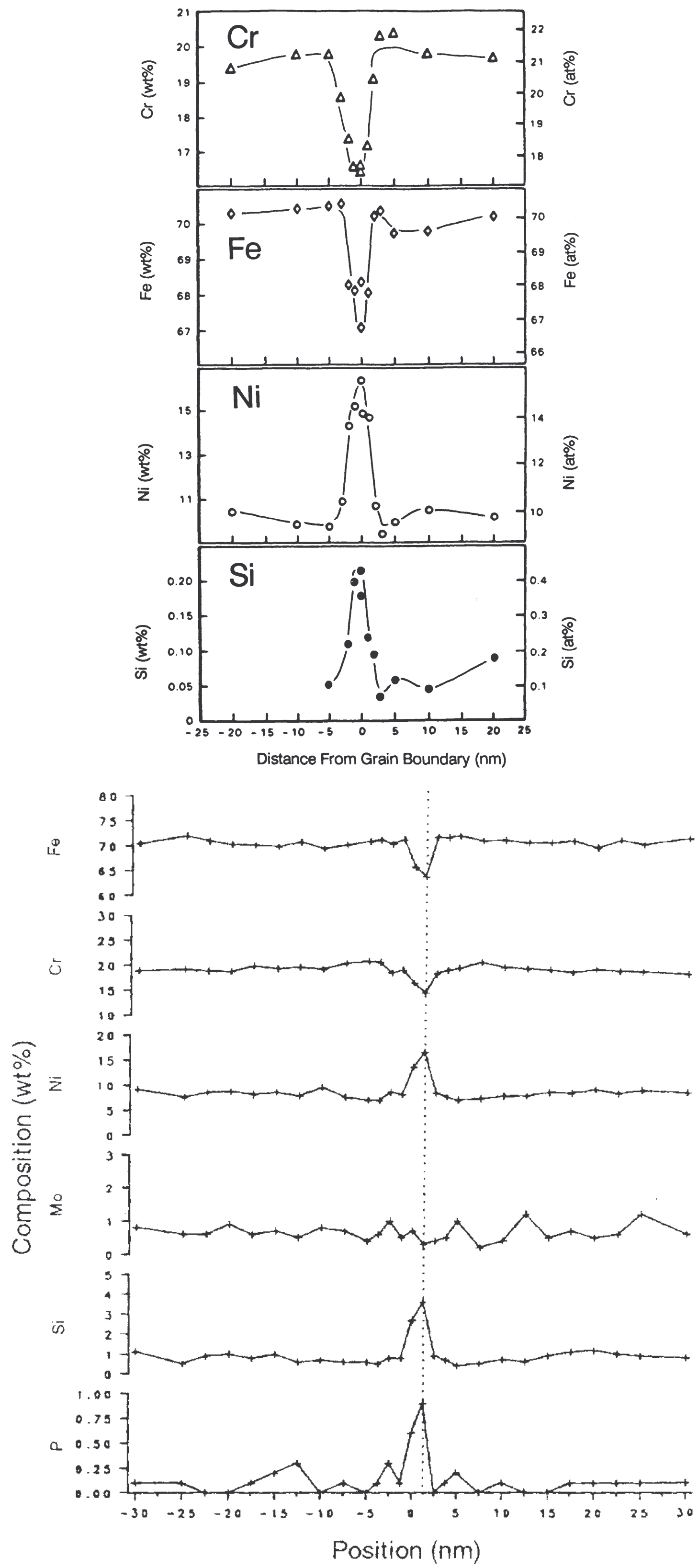
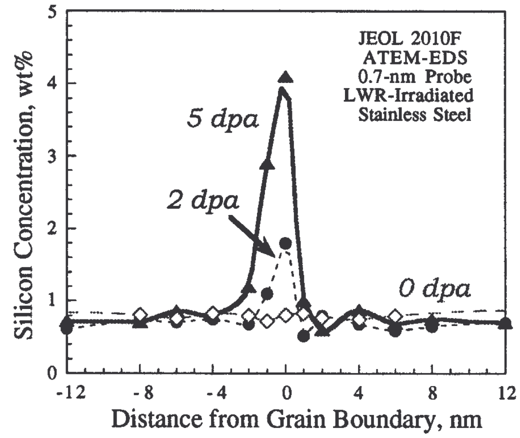


Figure B.2.7 Radiation induced segregation (RIS) of (a) high purity (including low Si) and (b) commercial purity stainless steel [9]. *(© NACE International 1990)*



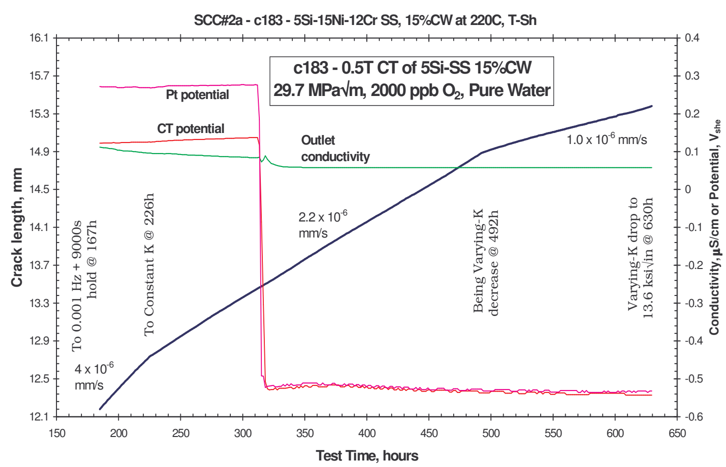


Figure B.2.8 (a) Grain boundary Si concentration in irradiated stainless steel. (b) Crack length vs. time for a 5% Si "stainless steel" whose composition simulates that in an irradiated grain boundary. No effect of corrosion potential and or stress intensity was observed. [20-22] *(© 2003 by The American Nuclear Society, La Grange Park, Illinois)*

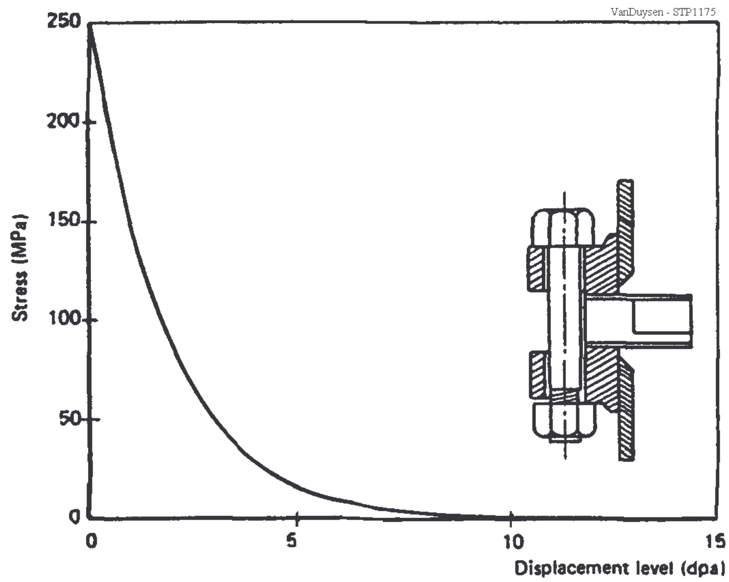


Figure B.2.9 The effects of radiation-induced creep on load relaxation of stainless steel in a constant displacement (bolt) condition [23]. *(Reprinted with Permission from ASTM)*

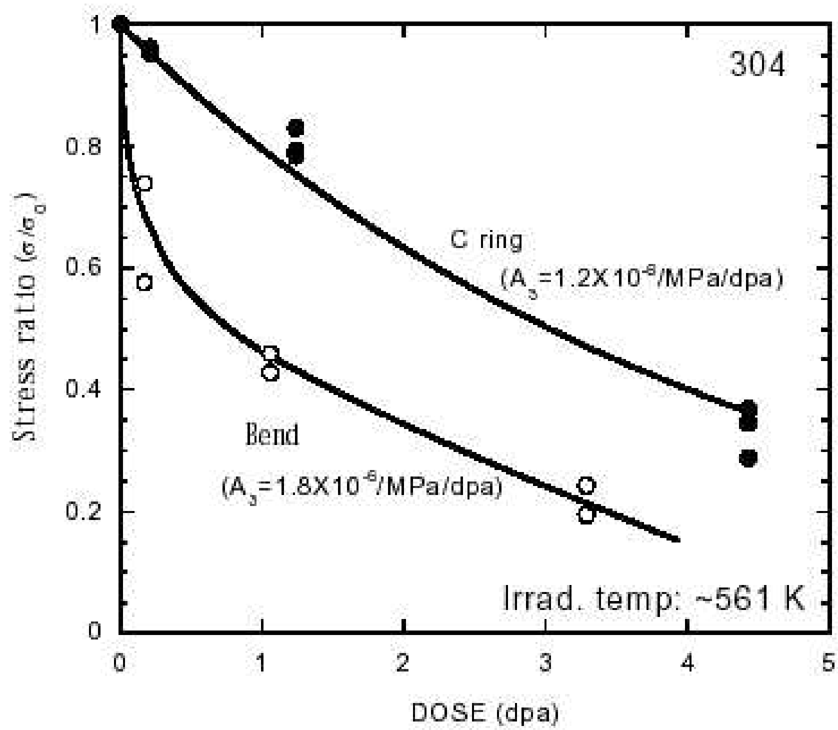


Figure B.2.10 Stress relaxation of bent beam and C-ring specimens of 304 SS in JMTR during irradiation at 288°C [23]. *(Reprinted with Permission from ASTM)*

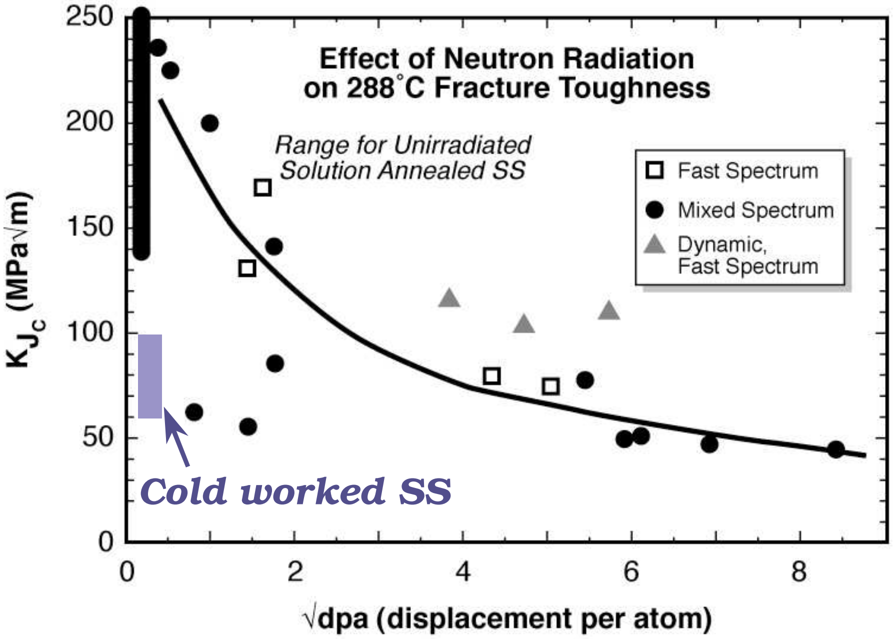


Figure B.2.11 The effect of fast neutron fluence under LWR conditions on fracture toughness of types 304 and 304L stainless steel at 288 °C [9,30,31]. A preliminary band based on the fracture toughness response of a few tests on unirradiated, cold worked stainless steel tested in-situ in 288 °C pure water is also shown [21]. *(Reprinted with Permission from Elsevier)*

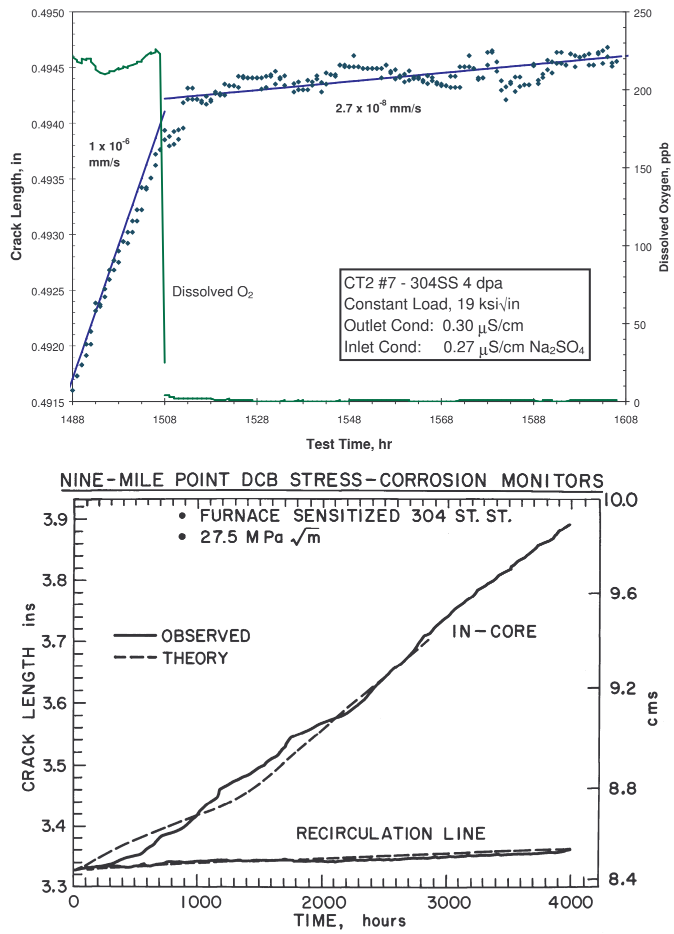
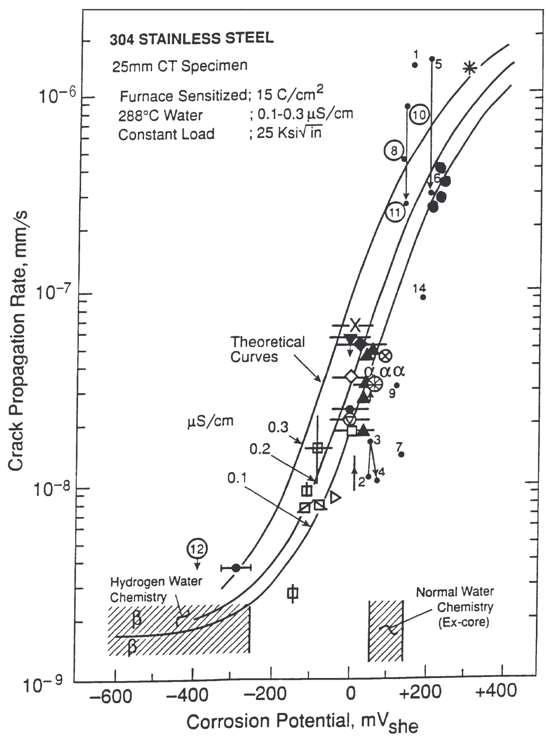


Figure B.2.12 Crack length vs. time for: (a) a CT specimen of irradiated type 304 stainless steel tested at constant load in 288 °C water at both high and low corrosion potential at 19 ksi√in. (b) DCB specimens of sensitized type 304 stainless steel exposed in core (high corrosion potential from radiolysis) and in the recirculation system [2,9,15-19]. *(© 2003 NACE International 1995)*



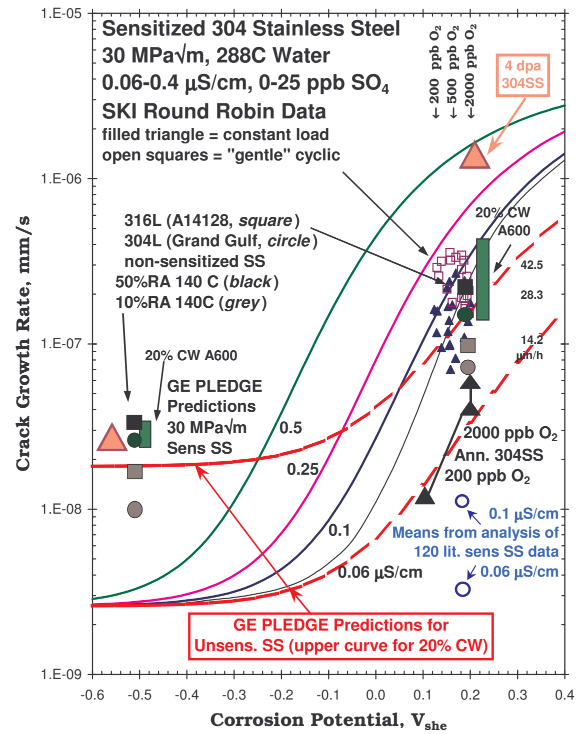


Figure B.2.13 SCC growth rate vs. corrosion potential for sensitized (left graph) and for annealed (black triangles), cold worked (large symbols), sensitized (small symbols) and irradiated (pink triangles) (right graph) SS in 288 °C water. Unirradiated and irradiated materials of similar yield strength show similar SCC response at low corrosion potential. At high potential, the combined effect of radiation hardening and radiation segregation produces a higher growth rate than either factor alone (i.e., in the unirradiated data that is either cold worked or sensitized). [2,9,15-19] *(© 2003 by The American Nuclear Society, La Grange Park, Illinois)*

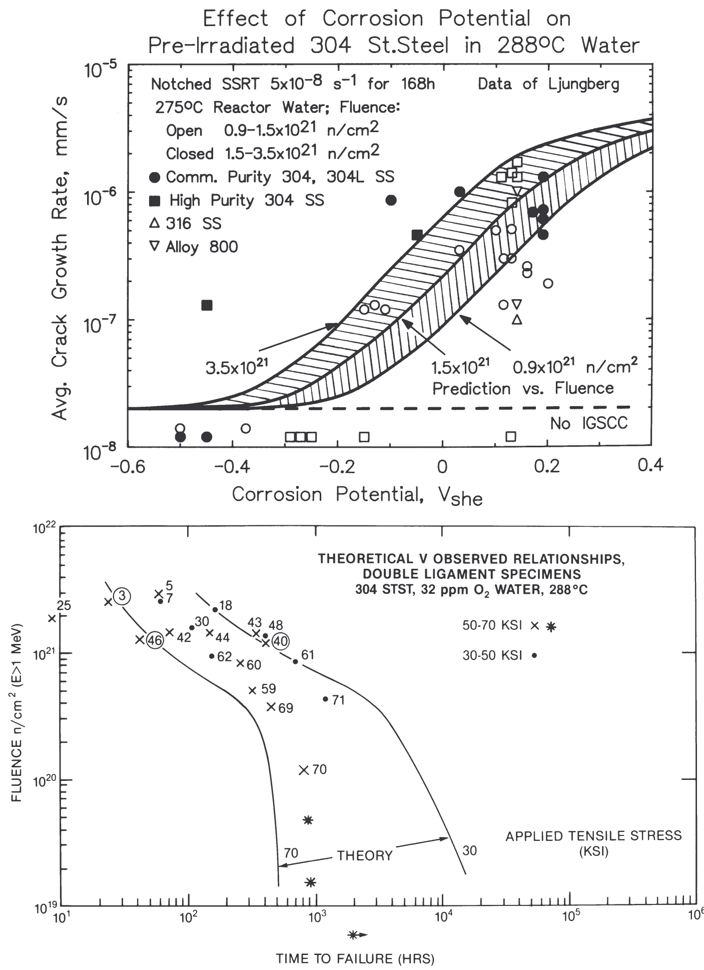


Figure B.2.14 Comparison of predicted and observed crack growth rates for stainless steels irradiated in a BWR at 288C to various fluences [2,9,15-19]. (a) Notched tensile specimens were tested by Ljungberg [32] at a slow strain rate in 288C pure water and interrupted after a given strain / time. (b) time-to-failure for the effect of fast neutron fluence on pre-irradiated type 304 stainless steel tested at constant load in the laboratory in oxygen saturated, 288 °C water [31]. *(© NACE International 1995)*

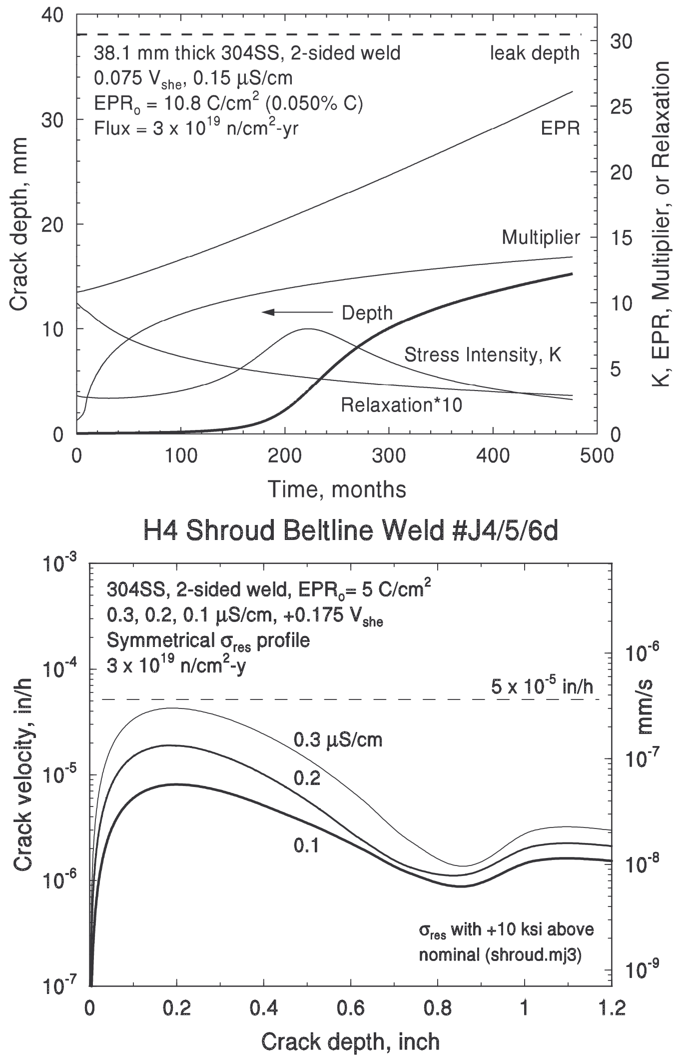
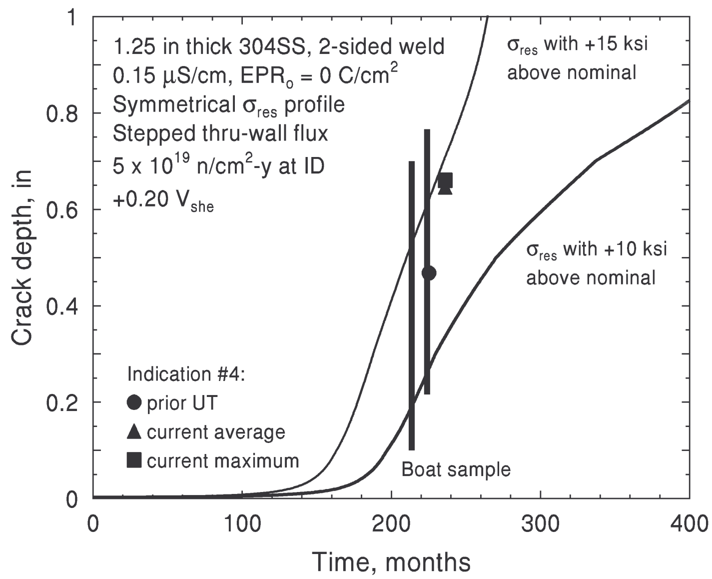


Figure B.2.15 Examples of IASCC predictions illustrating the interactions among radiation "damage" (segregation and hardening) and radiation creep relaxation (reduction in weld residual stress) for a BWR core shroud. (a) crack depth vs. time with individual curves for the increase in EPR (Cr depletion), stress relaxation, and "multiplier" (radiation hardening). The stress intensity factor is also shown, which goes through a peak due to the nature of the residual stress profile as well as radiation relaxation. (b) crack velocity vs. depth illustrating that at high coolant conductivity (0.3 µS/cm), cracks nucleate and grow earlier in life when the weld residual stresses are higher, resulting in higher growth rates and a shorter time to achieve a given crack depth [9,15].



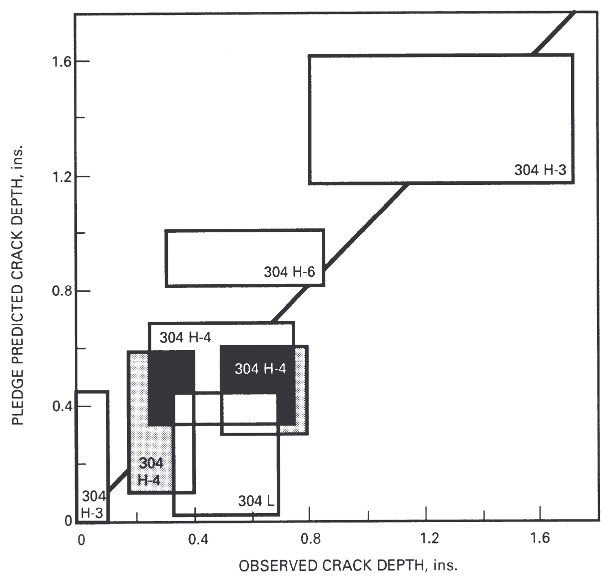


Figure B.2.16 (a) Crack length vs. time predictions and observations for a type 304 stainless steel BWR core shroud with multiple inspections and multiple cracks. (b) Comparison of observed and predicted crack depth for a number of BWR core shrouds. [9,15] *(© NACE International 1995)*