Irradiation Assisted Stress Corrosion Cracking

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Outline

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- NRC Research Activities
- Effect of Hydrogen Water Chemistry on IASCC CGR
- Testing of Grain Boundary Engineered Specimens
- Radiation Induced Segregation
- HAZ Testing
- Summary
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- Future Work
IASCC Mechanisms

Theories suggested to explain detrimental effect of irradiation on SCC can be classified into 5 categories:

- Solutes depletion or enrichment at grain boundaries caused by radiation-induced segregation (RIS).
- Hardening and loss of ductility due to irradiation defects.
- Localized plastic flow due to cleared dislocation channels.
- Void swelling, Irradiation creep
- Corrosion potential change due to radiolysis (in-situ effect)
NRC IASCC research focused on the following areas

- Evaluate effectiveness of SCC mitigations
  - *Hydrogen Water Chemistry*
  - *Grain Boundary Engineering*
- Evaluate CGR models for BWRs and PWRs
- Evaluate the causes, mechanisms and effects of EAC on BWR internals
- Effect of welding, thermal processes, and cold work on crack growth rates
- Review and evaluation of EAC in vessel internal components and emerging aging degradation issues
- Radiation embrittlement at relevant to PWR conditions
- Radiation and thermal embrittlement of cast austenitic stainless steels
Hydrogen Water Chemistry

Experimental CGR (m/s) vs. Stress Intensity K (MPa m$^{1/2}$) for Irradiated Stainless Steels at 289°C.

- **8 x NUREG-0313 Curve**
- **NUREG-0313 Curve**
- **Open Symbols: NWC BWR Env.**
- **Closed Symbols: HWC BWR Env.**

**Material & Dose**
- 304L 1.35 dpa
- 316L 1.35 dpa
- 316 1.35 dpa
- 316NG 1.4-2.0 dpa
- 304 Sensitized 0.75 dpa
- 304 sensitized 2.16 dpa
- 304L SAW HAZ 0.75 dpa
- 304L SAW HAZ 2.16 dpa
- 304 SMAW HAZ 0.75 dpa
- 304 SMAW HAZ 2.16 dpa
- 304 SMAW HAZ TT 0.75 dpa
- CF-8M Aged 2.46 dpa
- 304L 1.35 dpa
- 316L 1.35 dpa
- 316 1.36 dpa
- 304 SMAW HAZ 0.75 dpa
- 304 SMAW HAZ TT 0.75 dpa
- 316NG 1.4-2.0 dpa
Hydrogen Water Chemistry

- K/size criterion:
  \[ B_{\text{eff}} \text{ and } (W - a) \geq 2.5 \left( \frac{K}{\sigma_{\text{eff}}} \right)^2 \]

- \( \sigma_{\text{eff}} \) was suggested to take:
  - \( \frac{(\sigma_{\text{irr}} + \sigma_{\text{nonirr}})}{2} \) - by P. Andresen
  - \( \frac{(\sigma_{\text{irr}} + \sigma_{\text{nonirr}})}{3} \) - by A. Jenssen

- Those high CGRs in HWC are questionable

- Need to resolve the apparent diminishing benefit of HWC at high values of \( K \)

NUREG/CR-6960
### Corrosion Potential

**Andresen et al., 1994**

- $\text{H}_2\text{O} \rightarrow \text{O}_2, \text{H}_2\text{O}_2, \text{HO}_2^-, \text{H}, \text{OH}, \text{H}_2$
- Concentration is proportional to $(\text{flux})^{1/2}$
- Elevated corrosion potential, especially for intermediate DO level
GBE Results Type 304 SS

- IG area fraction is slightly higher in GBE 304, but there is more IG areas in nonGBE 304.
- TG cracking is more severe in nonGBE 304.
GBE Results Type 304L SS

- IG area fraction is higher in GBE 304L, number of IG areas is the same in both materials.
- TG cracking is observed along the centerline of the nonGBE 304 specimen.
GBE Results Alloy 690

- GBE 690 shows a larger elongation and retains considerable amount of strain hardening.
- No IG crack in GBE 690, while a small IG area is found in the nonGBE 690.
With about ~65% CSL fraction ($\Sigma \leq 29$), GBE treatment does not appear to improve IG crack in Type 304 or 304L SSs.

With about ~70% CSL fraction ($\Sigma \leq 29$), GBE treatment seems to suppress IG cracking and increase elongation in Alloy 690 at $\approx 2$ dpa.

### GBE Results Summary

<table>
<thead>
<tr>
<th>Material</th>
<th>nonGBE</th>
<th>GBE *</th>
</tr>
</thead>
<tbody>
<tr>
<td>Type 304</td>
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<td>67</td>
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<tr>
<td>Type 304L</td>
<td>64</td>
<td>66</td>
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<tr>
<td>Alloy 690</td>
<td></td>
<td>71</td>
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</table>

* Provided by vendor
Cr depletion is similar in both sensitized and irradiated SSs and, by analogy, RIS is suggested to play an important role in IASCC.

Some experimental results seem suggest a less important role of Cr depletion.
Radiation Induced Segregation - Carbon and Sulfur

- Carbon and Sulfur

GB at high fluence

Ni, Si, P, C, S segregated

Cr depleted

O diffuses fast along GB under stress and irradiation. Cr-Fe spinel oxide grows along GB.

S ions released into water

Ni is inert, remains unoxidized.

Some S ions diffuse to metal/spinel boundary.

Ni- and S-rich thin film and islands form.

At high S concentration, Ni- and S-rich film and islands melt or is amorphized under the influence of irradiation and stress. Voids and cavities form, preferentially near metal/spinel boundary and spinel tip.

When voids, cavities, and molten film are in significant amounts, the metal/spinel boundary loses strength, and crack tip advances along the boundary.

When S-rich Ni film and islands are amorphized or melt, Ni-S polyhedral cage is destroyed, S is freed from the cage and diffuses back into metal matrix. Altered distribution of S is observed when the tip is examined after discharge or test.
Small concentrations of Sulfur (within compositional specification) shown to be detrimental

Beneficial effect of Carbon
HAZ Crack Growth Rate Testing

Type 304 SS SMA Weld HAZ
Test CGRI JR–32 (Spec. 85-XA)
Fluence $1.44 \times 10^{21}$ n/cm$^2$

289°C
High–Purity Water

CGR = 2.21 x $10^{-10}$ m/s
$K_{\text{max}}$ = 13.9 MPa m$^{0.5}$

R = 0.74, Rise Time = 26 s

1.86 x $10^{-08}$ m/s

13.3 MPa m$^{0.5}$

0.42, 0.16s

CGR = 7.07 x $10^{-10}$ m/s
$K_{\text{max}}$ = 13.0 MPa m$^{0.5}$

R = 0.72, Rise Time = 433 s

1.98 x $10^{-10}$ m/s

13.9 MPa m$^{0.5}$

Constant Load

CGR = 2.61 x $10^{-10}$ m/s
$K_{\text{max}}$ = 14.0 MPa m$^{0.5}$

Constant Load

Crack advance direction

304 SMA HAZ, 2 dpa
Non-Irradiated HAZ Specimens

- NUREG-0313 curve bounds the nonirradiated HAZ CGR data.
- No difference between as-welded and thermally-treated SA HAZ
- Constant-load CGRs are considerably higher (8 - 10 times) than the NUREG-0313 curve at 0.75 dpa
- Thermal-treatment (500°C for 24 h) does not elevate IASCC susceptibility
- CGRs at 2 dpa are similar to that at 0.75 dpa
- The beneficial effect of HWC is evident
Irradiation effects that affect IASCC:
- RIS (Cr, Ni, Mo, Si, S, P …)
- Irradiation hardening and embrittlement
- Deformation mode
- Void swelling and irradiation creep (may be not at LWR temp.)
- Radiolysis

Work in progress to address regulatory needs:
- New irradiation to address synergistic effect of irradiation and thermal aging
- Verify dose limit for the onset of IASCC
- Effects of fabrication processes
- Measure fluence effects and determine saturation dose
- Effects of material chemistry, Effects of irradiation condition
Gaps in Understanding of IASCC

- Dose dependence of IASCC
- IASCC behavior under annealing
- Flux and temperature effects
- In-situ vs. ex-situ behavior,
- Irradiation effect on crack initiation and early stage of growth
- The relation between the deformation mode and IASCC
  - SFE, twining and channeling deformation mode in highly irradiated materials
Future Work

- Evaluate IASCC in PWR
  - Evaluate IASCC susceptibility in PWR environment as a function of fluence, material chemistry, and processing condition (e.g., SA or CW)
  - Void swelling behavior in austenitic SSs,
  - Synergistic effects of thermal and radiation embrittlement of cast SSs
  - Effectiveness of mitigation measures, e.g., GBE & low-S content

- Technical Approach
  - SSRT tests (5, 10, 40 dpa BOR-60 specimens)
    - To evaluate IASCC susceptibility (IG fraction) in PWR environment
  - TEM disks (<40 dpa BOR-60 specimens)
    - To assess void swelling at doses and temperatures relevant to PWRs
  - CGR/JR tests (new Halden irr., 4 CIR specimens, GE specimens, Zorita or others)
    - To evaluate IASCC CGRs at PWR relevant condition
    - To evaluate synergistic effects of thermal and radiation embrittlement