MRP RI-ITG Program
Results and Status

Jeff Gilreath  H. T. Tang
Duke Energy  EPRI
Rege Shogan
Westinghouse

NRC Meeting
October 23, 2003
NRC Headquarters
Rockville, MD
<table>
<thead>
<tr>
<th>Time</th>
<th>Item</th>
<th>Presenter</th>
</tr>
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<tbody>
<tr>
<td>9:00AM</td>
<td>Introductions</td>
<td>J.D. Gilreath, Duke Energy Corp.</td>
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<tr>
<td></td>
<td>Purpose of Meeting</td>
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<tr>
<td>9:05 AM</td>
<td>MRP RI-ITG Goals, Objectives and Scope</td>
<td>J.D. Gilreath, Duke Energy Corp.</td>
</tr>
<tr>
<td>9:20 AM</td>
<td>MRP RI-ITG Programs and Status</td>
<td>H. T. Tang, EPRI</td>
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<tr>
<td>10:30 AM</td>
<td>Recent Hot Cell Test Results</td>
<td>Rege Shogan, Westinghouse</td>
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<tr>
<td>11:15 AM</td>
<td>Open Discussion/NRC Comments</td>
<td>All</td>
</tr>
<tr>
<td>11:30 AM</td>
<td>Adjourn</td>
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Purpose of Meeting

Provide the status of the Material Reliability Program (MRP) Reactor Internals (RI) Issues Task Group (ITG), which also supports License Renewal Aging Management Programs referenced by some owners groups and utilities.
RV Internals Programs Background

- PWR Owners Groups
  - Have addressed rv internals for many years
    - A286 & X750 bolting, X750 split pin, thermal shields, upflow mod.
  - In early 1990s began evaluating significance of baffle bolt cracking
    - Plant categorization, operability evaluation, industry follow
  - In mid to late 1990s initiated more comprehensive programs addressing potential of baffle bolt cracking
    - Identified bolt design Information and susceptibility
    - Performed some baffle bolt inspections in U.S.
    - Removed and replaced some baffle bolts
    - Performed hot cell evaluations on failed baffle bolts
    - Prepared Safety Evaluation
  - Formed the JOBB
  - In late 1990s evaluated all susceptible rv internal components for potential aging effects - license renewal topicals
  - Industry-wide PWR Materials Reliability Program (MRP) developed (1998)
  - Created an Issue Task Group (ITG) to manage RV Internals aging issues
RI-ITG Program Mission/Vision

- To support the establishment of overall technically correct programs, which will **assure reactor vessel internals performing its design function through plant life** (60+ years of operation).

- This function is supported by:
  - serving as an industry focal point for resolution of issues related to reactor internals materials degradation,
  - **implementing needed programs to bring resolution of potential aging effects,**
  - **providing research on the effects of identified aging mechanisms on reactor internals,**
  - providing a focal point to support communication with the NRC when addressing aging of PWR reactor internals components.
RI-ITG Program Formation and Funding

• The RI-ITG program is formed to support EPRI/MRP member utilities to manage aging of reactor internals components
• The program cooperates with international partners to achieve cost effectiveness
• MRP/RI-ITG is funded by:
  – All US nuclear utilities
  – Foreign members of the EPRI Nuclear Power Sector, e.g., EDF, TEPCO
  – Foreign members of the MRP program, e.g., Kansai Electric Power Corp. and Japan Atomic Power Corp.
RI-ITG Program Coordination

- WOG Material Subcommittee
- International IASCC Program
- BWOG Material Subcommittee
- EDF RI Materials R&D Program (JOBB)
- BWRVIP
- EPRI Corrosion Research Program/CIR
RI-ITG Program Scope and Future Products

- Define and Quantify Material Degradation Mechanism
- Demonstrate Component Functionality
- Develop Flaw Tolerance Technical Basis and Acceptance Criteria
- Develop Screening Criteria and Inspection Guidelines
R-ITG Approach to Address Aging

• Identify potential aging effects to RV internals
• Screened RV Internal’s components for potential susceptibility
• Develop research programs to address identified needs
• Utilize experimental data and operating experience to further screen components where aging effects are negligible and determining lead components for inspections
• Perform functionality assessment on susceptible components
• Continue self assessments to assure end products are meeting need of industry and future products are highly effective addressing identified issues
# PWR Reactor Internals Materials (Examples)

<table>
<thead>
<tr>
<th>Material Type</th>
<th>Product Form</th>
<th>Components</th>
<th>Material Specifications</th>
</tr>
</thead>
<tbody>
<tr>
<td>304 / 304L</td>
<td>Bar</td>
<td>lugs, pads, pins, shims, plugs, retainers</td>
<td>A276 TP-304</td>
</tr>
<tr>
<td>316 / 316L</td>
<td></td>
<td></td>
<td>A479 TP-304</td>
</tr>
<tr>
<td>347</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Plate</td>
<td></td>
<td>core barrel, plenum cylinder, former and baffle plates, guide tubes</td>
<td>A240 TP-304A</td>
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<td></td>
<td></td>
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<td>A240 TP-304</td>
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<tr>
<td>Forgings</td>
<td></td>
<td>flanges, nozzles, radial keys, lugs</td>
<td>A473 TP-304</td>
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<td></td>
<td></td>
<td></td>
<td>A182 Gr F304</td>
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<tr>
<td>Pipe/Tubing</td>
<td></td>
<td>pipe, support posts, column sleeves, column extensions</td>
<td>A312 TP-304</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>A316 TP-304</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>A213 TP-304</td>
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<tr>
<td>Fastener</td>
<td></td>
<td>bolts, cap screws</td>
<td>A193 Gr B8</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>SA193 Gr B8M</td>
</tr>
<tr>
<td>308 / 308L</td>
<td>Welding Rod and Filler Metal</td>
<td>welds</td>
<td>A298 TE-308</td>
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<td></td>
<td></td>
<td></td>
<td>A371 TER-308</td>
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<tr>
<td>Alloy A286</td>
<td>Bar / Fastener</td>
<td>bolts, cap screws</td>
<td>SA-453 Gr 660</td>
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<tr>
<td>CF8 / CF3M</td>
<td>Casting</td>
<td>vent valve body, lower support columns</td>
<td>A351 Gr 660</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>A351 Gr CF8</td>
</tr>
</tbody>
</table>
Reactor Vessel Internals Description

- Core Support Assembly
- Core Support Shield Assembly
- Core Barrel Assembly
- Lower Internals Assembly
- Plenum Assembly
- Vent Valves
- Control Rod Guide Tube Assembly
- Upper Grid Assembly
- Thermal Shield
- Core Barrel
- Baffle Plates
- Former Plates
- Lower Grid
- Flow Distributor Head
- Incore Guide Tubes
- Plenum Cover Assembly
- Core (not in scope)
RI-ITG Program Elements to Address Reactor Internals Aging

State of Knowledge Assessment
- Radiation embrittlement
- IASCC
- Void swelling
- Creep/Stress relaxation
- License renewal SERs

Operating Conditions Analysis
- Temperature
- Fluence
- Stress

Hot Cell Testing
- Mechanical Properties
- Fracture toughness
- Crack initiation/growth
- Void swelling
- Creep/Stress Relaxation

Aging Management
- Screening/Inspection
- Operating conditions
- Improved materials
- Functionality
- Repair/replacement
Reactor Internal Inspections for License Renewal

1. **Scope**
2. **Preventive Actions**
3. **Parameters Monitored**
4. **Detection of Aging Effects**
5. **Monitoring and Trending**
6. **Acceptance Criteria**
7. **Corrective Action**
8. **Confirmation Process**
9. **Administrative Controls**
10. **Operation Experience**
Inspection Attributes

1. Scope
2. Preventive actions
3. Parameters Monitored
4. Detection of aging effects
5. Monitoring and Trending
6. Acceptance Criteria
7. Corrective Action
8. Confirmation Process
9. Administrative Controls
10. Operating Experience
Reactor Internals Aging Management Coordination and Time Line

1999- 2000

1999- 2001

1999- 2005

2005- 2008

2008 +

Susceptible Materials (GALL, topical, GTRs, SERs, etc.)

Limiting Environment (GALL, topical, GTRs, SERs, etc.)

Identify Potential Mechanism

Group Components/ Materials

Obtain Material Aging Data

Aging Management Guidelines and Analysis

Inspection Guidelines

Inspectors to support extended operations

Defined Attributes (Purpose, scope, method, sample size, acceptance criteria, corrective action, timing, administrative controls)

Material List

Temp. Profile

Fluence Profile

Material

Design Parameters

Stress

Inspect.

Inspect.

EPRI

EPRI

Owners Groups

EPRI

Owners Groups, EPRI

Utility, Industry, EPRI, NEI

 sparkle

EPR21
RI-ITG 1999 – 2000 Products Delivered

- Analysis of Baffle Former Bolt Cracking in EDF CPO Plants (MRP-03), TR-112209, 6/1999
- JOBB – CD Version 00.06, AP-114929-CD
- JOBB - CD Version 00.12, 1000777, 11/2000
- Inspection and Replacement of Baffle to Former Bolts at Point Beach-2 and Ginna, TR114779, 2/2000
RI-ITG 2001 Products Delivered

• Hot Cell Testing of Baffle/Former Bolts Removed From Two Lead Plants (MRP-51), 1003069, 11/2001

• Determination of Operating Parameters of Extracted Bolts (MRP-52), 1006075, 10/2001

• JOBB – CD Version 01.06, 1001360, June 2001

• In-Situ NDT Measurements of Irradiation-Induced Swelling in PWR Core Internal Components -- Phase 1: Testing of Unirradiated Surrogate Material, 100658, 10/2001

• Technical Basis Document Concerning Irradiation-Induced Stress Relaxation and Void Swelling in Pressurized Water Reactor Vessel Internals Components (MRP-50), 1000970, 11/2001
RI ITG 2002 Products Delivered

• Strategies for Management of Aging Effects in PWR Reactor Vessel Internals (MRP-62), 1006582, 2/2002
• JOBB CD 01.12, 1002858, 3/2002
• Characterization of Type 316 Cold-Worked Stainless Steel Highly Irradiated under PWR Operating Conditions (MRP-73), 1003525, 8/2002
• A Review of Radiation Embrittlement of Stainless Steels for PWRs (MRP-79), 1003524, 11/2002
• JOBB CD 02.12, 1002810, 12/2002
RI-ITG 2003 Studies

- Hot cell testing of decommissioned PWR baffle plate, former plate and core barrel samples (cofund with NEPO 051577)
  - Tensile, crack initiation, crack growth, fracture toughness, microstructure (void swelling, ...)
- In-pile PWR crack growth testing at Halden (Cofund with CIR and ROBUST Fuel Program)
- Hot Cell testing of Boris 5 and 6 materials
  - Tensile, crack initiation
- Boris 6 and 7 irradiation to achieve 40, 60 and 80 dpa
  - US materials
- Integration of CIR (Cooperative IASCC Research) Program data and understanding
- Inspection and flaw evaluation strategy
- Integrated component functionality evaluation strategy
  - Embrittlement, void swelling, stress relaxation/creep, fracture toughness, crack growth
Research Program and Results
RI-ITG Program Results and Planned Studies

• Characterize Irradiated Materials Properties
  – Programs
    • JOBB program
    • Baffle/former bolts and high strength bolts test program
    • Decommissioned PWR internals materials test program
    • International IASCC program
    • Halden crack growth program
  – Areas
    • Tensile tests – stress, strain, fracture toughness
    • Creep-stress relaxation tests
    • Corrosion tests – crack initiation, crack propagation
    • TEM investigation – microstructure, void swelling

• Inspection and flaw evaluation strategy (future reporting)
• Integrated component functionality evaluation strategy (future reporting)
JOBB Program

- EDF experience on service induced cracking of baffle/former bolts
- EDF R&D programs on effects of irradiation on current baffle bolting and vessel internal and possible replacement materials
- Performance of current and possible replacement materials
- Irradiation in BOR 60 fast reactor
  - EDF materials
  - US materials
JOBB Irradiation

Materials

– Representative of Core Internals of PWRs
  • SA 304L Baffle plates, Formers, Core barrel
  • CW 316, 347 Baffle bolts
  • 308 Welds, 304 HAZ
  • CASS

– Possible “replacement” materials
  • Density,
  • Irradiation creep,
  • Tensile Properties,
    - High doses,
    - Neutron spectrum effect,
  • Fracture toughness,
  • Microstructural investigations
Neutron Spectrum Profiles

Irradiation: Reactor thermal/fast mixed spectrum $\equiv$ PWR

Production of gas atoms and point defects

OSIRIS: Opencore pool-type research thermal reactor
70 MW

1 dpa $\sim 7 \times 10^{20} \text{n/cm}^2 (E > 1.0 \text{ MeV})$
Database of JOBB Materials Irradiated in the Boris Experiments

Materials representative of core internals:

- **Type 316 CW**: 4 bolt materials (F) + 3 bolt materials (US)
- **Type 304L SA**: 2 materials (F) + 2 materials (US) + 2 HAZ (F+US)
- **Type 308**: 2 materials (F) + 2 material (US)
- **Type 347 SA**: 1 material (D) + 1 material (US)
- **Type 321 SA**: 1 material (D) + 1 material (R) + 1 material (F)
- **Type CF8 CASS**: 1 material (US), 3 different TT
- **Two 316 SA and one 304 CW**

Other industrial materials:

- **N9 (12Cr-25Ni-Si-Ti)** Solution Annealed, Cold Worked and Thermal Aged
- **Inconel 690, Incolloy 800**
- **Nitronic 50 et 60**
- **Uranus, NMF18 SA and CW**

“Tailored” materials type 316:

- **Type 316 small grains, large grains, monocristal**
- **Type 316 High Purity, High Purity + Si, with adding of Zr, Hf or Fe**
- **316 Titanium (CEA)**: (0.25 Ti-0.35 Si), (0.25 Ti-0.8 Si), (1.1 Ti-0.35 Si), SA and CW
- **316 SPh SA and CW 20%**
- **316Nb**

“Tailored” materials type 304:

- **304 with adding of Zr, Hf et Fe**
## JOBB Phase 1 US Materials Irradiated in BOR 60 (tensile specimens only)

<table>
<thead>
<tr>
<th>Material No. of heats (Code) (Supplier)</th>
<th>Fluence 20dpa (Boris 4)</th>
<th>Fluence 40dpa (Boris 5)</th>
<th>Fluence* 60dpa (Boris 6)</th>
<th>Fluence* 80dpa (Boris 7)</th>
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<tbody>
<tr>
<td>Type 347 1 heat (EC) (W)</td>
<td>4</td>
<td>2</td>
<td></td>
<td>3</td>
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<tr>
<td>Type 316CW 2 heats (EA &amp; EB) (W)</td>
<td>5</td>
<td>2</td>
<td></td>
<td>5</td>
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<tr>
<td>Type 316SA 1 heat (ED) (CE)</td>
<td></td>
<td>3</td>
<td>4</td>
<td></td>
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<tr>
<td>Type 304SA 2 heats (FD &amp; EH) (FTI &amp; CE)</td>
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<td>4</td>
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<td>3</td>
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<tr>
<td>Type 308 1 heat (FE) (FTI)</td>
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* Irradiations planned or in progress
## JOBB Phase 2 US Materials in Bor 60 - Boris 6 & 7

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<tr>
<th>Materials</th>
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<th>5 dpa</th>
<th>10 dpa</th>
<th>20 dpa</th>
<th>40 dpa</th>
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<td>308 TIG/MIG weld and 304 HAZ (CE)</td>
<td>Tensile O-ring CT 3mm disc</td>
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<td>304 SA and CW (CE)</td>
<td>Tensile O-ring CT 3mm disc</td>
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<td>316 two heats (W, EDF)</td>
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<tr>
<td>Cast austenitic – as-received &amp; two levels of thermal aging* (W)</td>
<td>Tensile O-ring CT 3mm disc</td>
<td>19</td>
<td>13</td>
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<td>0</td>
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</table>

*400 °C for 100 and 1000 hours
JOBB Tensile Tests
Synthesis of the Tensile Tests Results: JOBB Boris Irradiation Representative Materials

- Saturation of the tensile characteristics between 5 and 10 dpa; higher for 316CW than for 304SA
- No significant change noticed between 10 and 125 dpa (at 330°C)
- Residual ductility at saturation is significant at ~10% total elongation while uniform elongation is often <1%
- No heat to heat variations of tensile properties after irradiation for 316CW nor for 304SA
- 308 welds and CASS have roughly the same behaviour as SA 304
Synthesis of the Tensile Tests Results (up to 125 dpa): Tensile Characteristics (YS and UTS) at 330°C

- Saturation of the hardening between 5 and 10 dpa; earlier for SA304 than for CW316
- Saturation hardening higher for 316CW (1000 MPa) than for 304SA (800 MPa)
- No significant change between 10 and 125 dpa (at 330°C)
Synthesis of the Tensile Tests Results (up to 125 dpa): Tensile Characteristics (UE and TE) at 330°C

- UE saturation level higher for CW316 than for SA304, TE similar (8-10%)
- No significant change between 10 and 125 dpa (at 330°C)
Synthesis of the Tensile Tests Results
Heat to Heat Variations: CW 316

- 7 different CW 316 materials; good homogeneity of mechanical characteristics
- No heat to heat variations after irradiation in terms of tensile properties
Synthesis of the Tensile Tests Results
Heat to Heat Variations: SA 304

- 4 different SA 304 materials (also SA 347); good homogeneity of mechanical characteristics
- No heat to heat variations after irradiation in terms of tensile properties
Synthesis of the Tensile Tests Results: Cast Stainless Steels

- Before irradiation, minimum thermal aging effect.
- Thermally aged hardening is higher than non-aged.
- CASS shows higher uniform elongation than the reference SA-304L after 5 dpa. After 10 dpa, the total elongation of the thermally aged CASS is lower than the non-aged CASS and the reference SA-304L.
JOBB Creep Tests
Synthesis of the Creep-Irradiation Results (up to 120 dpa)

Diameter 5.65 mm

Argon

Length mm

Tube Length

Unirradiated

Diameter mm

70 dpa
52 dpa
41 dpa
28 dpa

6.5
6.0
0 10 20 30 40 50

Length mm

55 mm
A creep law of the type: \( \varepsilon = B_0 \cdot \text{stress} \cdot \text{dose} - B_1 \), where \( B_0 \) is the creep rate, \( B_1 \) relates to incubation dose and stress is the uniaxial equivalent stress.
Stress Relaxation Based on Irradiation Induced Creep (Qualitative)
JOBB Constant Load test
Constant Load Tests on CHOOZ A Material in VTT LAB

Material:
- Chooz A baffle plate
- SA 304 stainless steel
- dose: 26-32 dpa,
- temperature: ~ 310°C,
- YS: 895 MPa; UTS: 900 MPa
Constant Load Tests on CHOOZ A Material - Stress Versus Time-to-Failure Curve

- Chooz A Corner, 30 dpa, SA 304 failed
- Bugey-2 bolt, 3P11R2, CW 316, 10 dpa, failed
JOBB Microstructure Investigation
Intergranular segregation

Slight increase in Ni at Grain Boundary
11 → 15 %

Slight decrease in Fe, Cr
Radiation Induced Segregation in CW 316 Steel Irradiated in BORIS

Intergranular segregation

Slight increase in Ni at Grain Boundary
11 $\Rightarrow$ 15%

Slight decrease in Cr
19 $\Rightarrow$ 14%
Radiation Induced Segregation in CW 316 Steel Irradiated in BORIS

Intergranular segregation

10 dpa
Study in progress

Slight increase in Ni 14%

MRP
Radiation Induced Segregation in CW 316 Steel Irradiated in BORIS

• Intergranular segregation
  – Slight increase in Ni content
  – Slight decrease in Cr and Fe
  – Slight difference between 10 and 20 dpa
  – Mo shows no change

  To be confirmed with future data at 10 and 40 dpa

• Same tendency observed in CW 316 Bugey Bolt and SA 304 Chooz A Corner
• Baffle/former bolts test program
• Decommissioned PWR internals materials test program
• International IASCC program
• JOBB corrosion test program
# Types of Materials Being Studied

<table>
<thead>
<tr>
<th>Program</th>
<th>Alloy</th>
<th>Component</th>
<th>Irradiation Source</th>
<th>Irradiation environment</th>
</tr>
</thead>
<tbody>
<tr>
<td>US Baffle-Former bolts</td>
<td>CW316 SS</td>
<td>Baffle-former bolts</td>
<td>PWR</td>
<td>Water 8-15 dpa</td>
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<tr>
<td></td>
<td>347SA SS</td>
<td>Baffle-former bolts</td>
<td>PWR</td>
<td>Water 2-21 dpa</td>
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<tr>
<td></td>
<td>CW304 SS</td>
<td>Lock bars &amp; washers</td>
<td>PWR</td>
<td>Water 20 dpa</td>
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<tr>
<td>International IASCC Adv. Com.</td>
<td>CW316 SS</td>
<td>BMI thimble</td>
<td>PWR</td>
<td>Water 01-65 dpa</td>
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<tr>
<td>Decommissioned 304 SS</td>
<td>CW 316 SS</td>
<td>Bar</td>
<td>BOR60</td>
<td>Sodium 0, 20, 40 dpa</td>
</tr>
<tr>
<td>304SA SS</td>
<td>Baffle</td>
<td>PWR</td>
<td>Water 0-23 dpa</td>
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</tr>
<tr>
<td>304SA SS</td>
<td>Former</td>
<td>PWR</td>
<td>Water 0-18 dpa</td>
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<tr>
<td>304SA SS</td>
<td>Barrel</td>
<td>PWR</td>
<td>Water 0-0.07 dpa</td>
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<tr>
<td>304CW SS</td>
<td>Baffle-former bolts</td>
<td>PWR</td>
<td>Water 0-23 dpa</td>
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<tr>
<td>JOBB/MRP</td>
<td>CW316 SS-W</td>
<td>PWR bolting</td>
<td>BOR60</td>
<td>Sodium 0, 20 dpa</td>
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<tr>
<td>304SA SS- FTI</td>
<td>Baffle-former bolts</td>
<td>BOR60</td>
<td>Sodium 0, 20 dpa</td>
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<tr>
<td>308 SS Weld-FTI</td>
<td>Core barrel weld</td>
<td>BOR60</td>
<td>Sodium 0, 20 dpa</td>
<td></td>
</tr>
<tr>
<td>347SA SS-W</td>
<td>PWR bolting</td>
<td>BOR60</td>
<td>Sodium 0, 20 dpa</td>
<td></td>
</tr>
</tbody>
</table>

1 dpa~7\times10^{20} \text{n/cm}^2, \text{E}>1\text{MeV}  
BOR60 is a fast neutron spectrum reactor
Baffle-Former Bolt Configuration

CE reactor design has a welded rather than a bolted assembly.
304SS Removed from Decommissioned Plant
Locations of Baffle and Former Plate Samples

- Top of the upper baffle
- Upper baffle
- Lower region of the upper baffle
- Core former location
Specimen Cutting

BAFFLE FERMER

\[ \frac{1}{2}T-CT \]
<table>
<thead>
<tr>
<th>Program</th>
<th>Alloy</th>
<th>Component</th>
<th>Tensile</th>
<th>Fracture Toughness</th>
<th>SSRT</th>
<th>Crack initiation</th>
<th>Crack growth rate</th>
<th>Microstructure/swelling</th>
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<tbody>
<tr>
<td>US B-F bolts</td>
<td>CW316 SS</td>
<td>Baffle-former bolts</td>
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<td>x</td>
<td>x</td>
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<td>347SA SS</td>
<td>Baffle-former bolts</td>
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<td>x</td>
<td>x</td>
<td>x</td>
<td>x</td>
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<tr>
<td></td>
<td>CW304 SS</td>
<td>Lock bars</td>
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<tr>
<td>International IASCC Adv. Com.</td>
<td>CW316 SS</td>
<td>BMI thimble</td>
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<tr>
<td>CW 316 SS</td>
<td>Bar</td>
<td></td>
<td>x</td>
<td></td>
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<tr>
<td>Decommissioned 304 SS</td>
<td>304SA SS</td>
<td>Baffle</td>
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<tr>
<td></td>
<td>304SA SS</td>
<td>Former</td>
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<td>x</td>
<td>x</td>
<td>x</td>
<td>x</td>
<td>x</td>
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<tr>
<td></td>
<td>304SA SS</td>
<td>Barrel</td>
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<td>x</td>
<td>x</td>
<td>x</td>
<td>x</td>
<td>x</td>
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<tr>
<td></td>
<td>304SA SS</td>
<td>Baffle-former bolts</td>
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<td>x</td>
<td></td>
<td></td>
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<td>JOBB/MRP</td>
<td>CW316 SS-W</td>
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<td>308 SS Weld-FTI</td>
<td>Core barrel weld</td>
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<tr>
<td></td>
<td>347SA SS-W</td>
<td>Baffle-former bolts</td>
<td>x</td>
<td></td>
<td>x</td>
<td></td>
<td></td>
<td>x</td>
</tr>
</tbody>
</table>
Tensile Testing

2 Tensile (HEAD)

2 Tensile (THREAD)

Baffle bolt specimens

1-0.25 dia 304SS from former

BMI Tubing

BOR reactor irradiated specimens
Tensile Test Parameters

• Generally subsized specimens sized relative to the available component size
• ASTM E8 & 21 as applicable
• Air environment
• Primarily 608°F (320°C), + room temperature
Effects of Irradiation on the Stress-Strain Curve

![Graph showing effects of irradiation on stress-strain curve.](image-url)
Tensile Test Results – Yield Stress at RT

Yield Strength vs dpa at Room Temperature

Yield Strength, ksi

Fluence, dpa

- 304SS
- CW316 tube
- 347SS-bolt
- CW 316 bolt
- CW 304 bar
Tensile Test Results – Uniform Elongation at RT

Uniform Elongation vs dpa at Room Temperature

- 304SS
- CW316 tube
- 347 bolt
- CW316 bolt
- CW304
Tensile Test Results – Total Elongation at RT

Total Elongation vs dpa at Room Temperature

- 304SS
- CW316 tube
- 347SS bolt
- CW316 bolt
- CW304
Tensile Test Results – Yield Strength at 608F

Yield Strength vs dpa at 608F

Yield strength, ksi

Fluence, dpa

- 304SS
- 347SS bolt
- CW304 bar
- CW316 bolt
- CW316 tube
Tensile Test Results – Uniform Elongation at 608F

Uniform Elongation vs dpa at 608F

- 304SS
- 347SS bolt
- CW304 bar
- CW316 bolt
- CW316 tube

Fluence, dpa

Uniform Elongation, %
Tensile Test Results – Total Elongation at 608F

Total Elongation vs dpa at 608F

- 304SS
- 347SS bolt
- CW304 bar
- CW316 bolt
- CW316 tube

Fluence, dpa

Total Elongation, %
Decommissioned 304 Material Program – Standard 1/2T and 1T-CT Specimens

Baffle bolt Program used actual bolts with ½ through crack under head
IASCC Susceptibility - SSRT Test Parameters

- Same specimen designs as for tensile testing
- Tensile test at ~$10^{-7}$ s$^{-1}$
- Simulated PWR water
  
  \[
  \begin{align*}
  \text{H}_3\text{BO}_3 & \quad 1000 \text{ ppm as B} \\
  \text{LiOH} & \quad 2 \text{ ppm as Li} \\
  \text{Dissolved oxygen} & \quad < 5 \text{ ppb} \\
  \text{Dissolved hydrogen} & \quad 30 \text{ cc/kg} \\
  \text{Chloride} & \quad < 30 \text{ ppb} \\
  \text{Fluoride} & \quad < 30 \text{ ppb} \\
  340^\circ\text{C}
  \end{align*}
  \]
SSRT Fractography
IASCC Test Results

Decommissioned 304 vs Commercial PWR alloys
320-340C

% intergranular in SSRT test

dpa

International Phase 2, CW316
 Decommissioned 304
 Bugey 316CW B-F bolt
 Point Beach 347 BF bolts
 Farley 316CW BF bolts
SSRT Test Results
Effects of Reactor Source

Test reactor vs PWR
All data

% intergranularity in SSRT test

PWR
TEST

dpa

0.0 20.0 40.0 60.0 80.0
Crack Initiation Testing
Crack Initiation Test Parameters

- Stressed “O” rings at constant load
- 316 CW thimble tube specimens
- Time to failure (~time to crack initiation)
- Simulated PWR water
  - $\text{H}_3\text{BO}_3$ 1000 ppm as B
  - LiOH 2 ppm as Li
  - Dissolved oxygen $< 5 \text{ ppb}$
  - Dissolved hydrogen 30 cc/kg
  - Chloride $< 30 \text{ ppb}$
  - Fluoride $< 30 \text{ ppb}$
- 340ºC
Crack Initiation Results

*No failures in BOR60 irradiated specimens at up to 40 dpa and loads equal to 120% of the yield strength
Crack Growth Rate Measurements

- Constant K
- Simulated PWR water

- $\text{H}_3\text{BO}_3$ 1000 ppm as B
- LiOH 2 ppm as Li
- Dissolved oxygen $<$ 5 ppb
- Dissolved hydrogen 30 cc/kg
- Chloride $<$ 30 ppb
- Fluoride $<$ 30 ppb

- 320°C
Crack Growth Rate Measurement Results

![Graph showing crack growth rate vs. K (MPa)]

- Crack Growth Rate (m/s)
- K (MPa)

1 inch/year growth rate

- 347 SS, 17 DPA
- 316 SS, 13 DPA
Halden Crack Growth Test Program
Halden Crack Growth Test in PWR Environment

- In-core, long term, crack growth data under PWR conditions of 2 ppm Li, 1200 ppm B and 2-4 ppm hydrogen at 340°C
- Four 304 SS CT specimens with fluence of $2.5 \times 10^{22}$, $2.5 \times 10^{22}$, $1.2 \times 10^{22}$ and $9.0 \times 10^{21}$ n/cm$^2$ (>1 MeV)
Halden 304 SS Crack Growth Test Results – An Example
Halden PWR Crack Growth Test

IFC 657: PWR conditions (temp. 335 °C, 2-3 ppm H₂)
Microstructure Investigation
Swelling (290 to 350°C)

Data from CW316SS & 347SS B-F bolts & CW316SS tube
# PWR Void Swelling Results to Date

<table>
<thead>
<tr>
<th>Samples</th>
<th>Swelling (%)</th>
<th>Temperature Range (°C)</th>
<th>Exposure (dpa)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Point Beach Bolt SA 347</td>
<td>0.029</td>
<td>359 to 401</td>
<td>7 to 8</td>
</tr>
<tr>
<td>Farley Bolt CW 316</td>
<td>0.011</td>
<td>359 to 401</td>
<td>9 to 11</td>
</tr>
<tr>
<td>Tihange Bolt A CW 316L (EDF)</td>
<td>0.12</td>
<td>363 to 373</td>
<td>12.05</td>
</tr>
<tr>
<td>Tihange Bolt B CW 316L S1 (PNNL)</td>
<td>&lt;0.010</td>
<td>320</td>
<td>19.5</td>
</tr>
<tr>
<td>Tihange Bolt B CW 316L S2 (PNNL)</td>
<td>0.2</td>
<td>342</td>
<td>11.9</td>
</tr>
<tr>
<td>Tihange Bolt B CW 316L S3 (PNNL)</td>
<td>0.24</td>
<td>330</td>
<td>7</td>
</tr>
<tr>
<td>High Flux Thimble CW 316 S1</td>
<td>0.0294</td>
<td>320</td>
<td>65</td>
</tr>
<tr>
<td>High Flux Thimble CW 316 S2</td>
<td>0.0274</td>
<td>295</td>
<td>61</td>
</tr>
<tr>
<td>High Flux Thimble CW 316 S3</td>
<td>0.014</td>
<td>325</td>
<td>17</td>
</tr>
</tbody>
</table>
Radiation Induced Segregation

![Graph showing radiation induced segregation](image)
Gas Content

Gas Content

Fluence, dpa

ppm by weight

Hydrogen

Helium

0 10 20 30 40 50 60 70

0 10 20 30 40 50 60 70

50 50
Radiation and Temperature Analysis
Macroscopic Sample Data from Decommissioned PWR Core Midplane XYZ TORT
Fast (E > 1.0 MeV) Neutron Fluence and Stainless Steel 304 dpa

<table>
<thead>
<tr>
<th>Core Side Surface</th>
<th>Middle of Plate</th>
<th>Back Side Surface</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Point</td>
<td>Phi</td>
</tr>
<tr>
<td>1</td>
<td>2</td>
<td>2.9E+21</td>
</tr>
<tr>
<td>2</td>
<td>1.6E+22</td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>2.9E+21</td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>1.6E+22</td>
<td></td>
</tr>
</tbody>
</table>

Looking at Sample from Core Side of Sample
### Former Plate Results - Midplane

Macroscopic Sample Data from Decommissioned PWR Core Midplane XYZ TORT
Fast (E > 1.0 MeV) Neutron Fluence and Stainless Steel 304 dpa

**Former Plate 3 - Inner Corner Sample**

<table>
<thead>
<tr>
<th>Bottom Surface</th>
<th>Middle of Plate</th>
<th>Top Surface</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Point</strong></td>
<td><strong>Phi</strong></td>
<td><strong>dpa</strong></td>
</tr>
<tr>
<td>1</td>
<td>1.2E+22</td>
<td>18.1</td>
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<tr>
<td>2</td>
<td>3.5E+21</td>
<td>5.3</td>
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<tr>
<td>3</td>
<td>4.2E+21</td>
<td>6.3</td>
</tr>
<tr>
<td>4</td>
<td>7.5E+21</td>
<td>11.1</td>
</tr>
</tbody>
</table>

Looking Down at Sample from Above

```
Pnt 1
     \ |
     \|
  Pnt 2
     /|
     / \
Pnt 3  Pnt 4
```
Temperature Calculations

Finite element Mesh

Temperature Contours at EOC, Cycle 11
### Bolt Centerline Temperature

<table>
<thead>
<tr>
<th>Removed Bolt ID</th>
<th>Former Level</th>
<th>Centerline Bolt Max. Temperature (Deg.F)</th>
<th>HGR State Point</th>
<th>Fast State (E &gt; 1.0 MeV) Neutron Fluence</th>
<th>Stainless Steel dpa</th>
<th>Stress Ranking (1=Highest)</th>
</tr>
</thead>
<tbody>
<tr>
<td>3321</td>
<td>1</td>
<td>596</td>
<td>EOC-05</td>
<td>4.51E+21</td>
<td>7.0</td>
<td>6</td>
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<tr>
<td>4821</td>
<td>1</td>
<td>577</td>
<td>EOC-06</td>
<td>2.37E+21</td>
<td>3.6</td>
<td>1</td>
</tr>
<tr>
<td>1312</td>
<td>2</td>
<td>657</td>
<td>EOC-05</td>
<td>6.80E+21</td>
<td>10.5</td>
<td>2</td>
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<tr>
<td>4122</td>
<td>2</td>
<td>657</td>
<td>EOC-05</td>
<td>6.80E+21</td>
<td>10.5</td>
<td>2</td>
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<td>2922</td>
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<td>EOC-06</td>
<td>8.26E+21</td>
<td>12.4</td>
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<td>2</td>
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<td>EOC-02</td>
<td>1.06E+22</td>
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<td>5</td>
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<td>3812</td>
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<td>621</td>
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<td>4522</td>
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<td>10.3</td>
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<td>4326</td>
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<td>634</td>
<td>EOC-02</td>
<td>9.05E+21</td>
<td>14.1</td>
<td>11</td>
</tr>
</tbody>
</table>

Broken bolts are in bold
Temperature Map of the Decommissioned PWR Plant Baffle/Former Assembly

Model of middle level formers and baffle
Overall Summary
IASCC Observations

• No evidence for significant electrochemical polarization due to radiolysis or of chemical concentration in super-heated crevices in PWRs

• IASCC can occur in highly irradiated (generally >5dpa) 300 series stainless steels in hydrogenated PWR water (consistent with results in hydrogen water chemistry for BWRs)

• No evidence that grain boundary segregations, helium or hydrogen embrittlement play a significant role in IASCC

• High strength from irradiation hardening does seem to be an important factor and may explain heat to heat variability

• Quantitative correlation of PWR and Fast Reactor Data will be performed
SSRT Test Results – Stress/dpa Effect

Susceptibility to IASCC (Qualitative)

Stress vs. dpa
O-ring Tests - Crack Initiation

Need quantification

- % of yield stress vs Time to failure, hrs
- 0 dpa
- 20 dpa
- 40 dpa
- 75 & 70 dpa
Fracture Toughness

• Reductions in the toughness of the austenitic stainless steel internals components are expected during PWR operation
• The materials are expected to retain sufficient ductility and toughness
• PWR irradiated data currently available is ~ 15 dpa using non-standard bolt specimens
• Faster reactor irradiated data available is ~ 10 to 20 dpa
• More data are being and will be generated, decommissioned PWR materials, Bor irradiated materials
• Bor materials will also include thermally aged CASS materials
Crack Growth

- Non-standard bolt specimens of ~15 dpa gave some qualitative data of crack growth, however not satisfying validity rule
- CT specimen data are gradually coming out for fluence >10 dpa – Halden in-pile testing, decommissioned PWR materials, CIR program
- Future data will include both fast and PWR irradiated materials
Void Swelling

- Limited PWR data show very small swelling (low dose CW316 and SA347)
- Fast reactor data, many from non-PWR type of materials, fitted equations such as the Foster-Flynn equation cannot and should not be applied to PWRs
- 304 appears to swell more than 316
- Swelling is coupled with state of creep and stress relaxation
- Need PWR high dose and high temperature swelling data to develop swelling prediction applicable to PWRs
SA 304 and CW 316 Swelling

Irradiation 375°C – 450°C

- SA 304 and CW 316 Swelling
- Incubation dose
  - SA 304-304L: 10-15 dpa
  - CW 316-316L: 30-40 dpa
- Swelling rate
  - 0.2% dpa⁻¹
  - 0.06% dpa⁻¹

Parameters fonction of Temperature
Flux Effect on Swelling

\[
\frac{\Delta V}{V_0} = 3 \frac{\Delta D}{D_0}
\]

**Conditions**
- EBR II
- 316
- 420°C
- 20 dpa

**Graphs**
- PWR fluxes: \( \phi_3 < \phi_2 < \phi_1 \)
- FBR fluxes: \( \phi_1 \)
- Phénix

**Equation**
- \( \Delta V/V \) vs. Flux

**Figure**
- MRP

**Chart**
- Plot of Swelling (S) vs. Dose Rate (10^-7 dpa/s)
Summary

- Reactor internals material degradation mechanisms have been extensively studied.
- Data obtained and to be obtained will support the development of degradation threshold (behavior model) as a function of material, fluence, temperature and stress.
- The degradation threshold values (behavior models) will support:
  - the development of screening criteria and flaw tolerance technical basis.
  - the evaluation of component functionality.
  - the disposition of inspection findings.
  - the development of inspection guidelines for aging management.