EPRI Report MRP-211, Revision 1

Subsequent License Renewal

Kyle Amberge
EPRI

Heather Malikowski
Exelon

Steve Fyfitch, Sarah Davidsaver
Framatome, Inc.

NRC Public Meeting
February 13, 2018

Agenda

Time	Agenda Item	Presenter
8:45 am	Opening remarks	Kyle Amberge, *EPRI*
9:00 am	Purpose and objective of meeting	Kyle Amberge, *EPRI*
9:30 am	Background/history of MRP-211	Steve Fyfitch, *Framatome*
10:30 am	Break	All
10:45 am	MRP-211, Revision 1 irradiated materials database and models	Sarah Davidsaver, *Framatome*
12:00 pm	Lunch	All
1:30 pm	Discussion of MRP-211, Revision 1	NRC Staff
2:30 pm	Responses to Other Questions	Industry
3:00 pm	Break	All
3:15 pm	Stakeholder participation	All
3:30 pm	Summary for the day and actions	NRC Staff/EPRI
3:45 pm	Adjourn	All

Opening Remarks

Kyle Amberge, *EPRI*

Roadmap for MRP-227 Development

- MRP-134 – Fundamental approach and framework
- MRP-156/-157 – Materials issue management table
- **MRP-211** – Irradiated stainless steel properties
- **MRP-175** – Screening criteria for aging mechanisms
- MRP-135 – Constitutive models for ANSYS FE plug-in
- MRP-189/-191 – FMECA ranking by component
- MRP-229/-230 – Engineering analysis/finite element model
- MRP-231/-232 – AMP strategies assigned by component
- MRP-227 – Inspection recommendations

Purpose and Objective of Meeting

Kyle Amberge, *EPRI*

Purpose and Objective of 2/13-14 Meeting with NRC

- To interact with NRC Staff proactively for subsequent license renewal (SLR) program development
  - Previous similar Reactor Vessel Internals meetings for (first) LR pertaining to MRP-175 and MRP-211
    - 11/16/05 – MRP-175 discussion prior to publication (meeting summary ML053270247)
    - 5/3/06 – MRP-175 discussion after publication (meeting summary ML061290492)
    - 2/23/07 – MRP-211 discussion (meeting announcement ML070390233)
  - Per MRP-227-A SER, the industry considers that there are no open items for first LR
- To keep NRC Staff informed of the SLR program development and progress of the Joint EPRI MRP Reactor Internals Core Planning Team
- To obtain feedback from the NRC Staff regarding the SLR program development and progress of the Joint EPRI MRP Reactor Internals Core Planning Team

Purpose and Objective of NRC Meeting 2/13/18

- Present irradiated austenitic stainless steel material property database, including identification of updates from MRP-211, Revision 0 (2007)
- Present discussion of irradiated austenitic stainless steel material constitutive models, including identification of updates and changes from MRP-211, Revision 0 (2007)
- Foster technical discussion with NRC
  - Review and discuss other materials related questions from staff
- Identify future meetings/topics and interactions with NRC

Background/History of MRP-211 2005-2007

Steve Fyfitch, *Framatome*

Background/History of MRP-211, Revision 0

- Industry met with NRC 2/23/07 to discuss the development MRP-211 (ML070390233, Proprietary information presented at that meeting)
- MRP-211, Revision 0 published December 2007
  - Full citation:

    Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data—State of Knowledge (MRP-211). EPRI, Palo Alto, CA: 2007. 1015013.
  - Public access in 2009
- Prepared by AREVA NP, with contributions from Westinghouse, ANATECH, and EPRI
  - Steve Fyfitch, Peter Scott, Lionel Fournier, Robert (Bob) Gold, Joe Rashid, Robert Dunham, Mike Burke
- MRP-211, Revision 0 submitted to NRC (see ML093020614)

Background/History of MRP-211, Revision 0

- MRP-211 provides data to develop the technical bases for constitutive models to be used in engineering evaluations and assessments
  - Data compared to MRP-135 models and adjustments made
- These evaluations and assessments are used to refine the categorization and ranking of PWR RV internals component items and welds
  - MRP-211, Revision 0 supports (first) license renewal, i.e., 60 years
  - MRP-211, Revision 1 supports subsequent license renewal, i.e., beyond 60 years
- MRP-211 provides database and general trends/available models

Background/History of MRP-211, Revision 0

- Approach:
  - Expert panel elicitation
  - Relevant data gathered, reviewed, and captured in database
  - Existing models (i.e., MRP-135) discussed and evaluated against database
  - Recommendations identified for model changes (next MRP-135 revision)
- MRP-211, Revision 0 report sections:
  - Section 1 – Introduction
  - Report purpose (Section 1.1)
  - Background (Section 1.2)
  - Report structure (Section 1.3)
  - References (Section 1.4)

Background/History of MRP-211, Revision 0

- MRP-211, Revision 0 report sections (cont.):
  - Section 2 – Data (sources, trends, gaps, and references)
    - Tensile test data (Section 2.1)
    - Fracture toughness data (Section 2.2)
    - Thermal and irradiation creep/stress relaxation data (Section 2.3)
    - Void swelling data (Section 2.4)
    - IASCC initiation data (Section 2.5)
    - IASCC growth data (Section 2.6)

Background/History of MRP-211, Revision 0

- MRP-211, Revision 0 report sections (cont.):
  - Section 3 – Material constitutive equations
    - Tensile data models (Section 3.1)
    - Fracture toughness models (Section 3.2)
    - Irradiation-enhanced stress relaxation and creep model (Section 3.3)
    - Void swelling model (Section 3.4)
    - IASCC initiation model (Section 3.5)
    - References (Section 3.6)
  - Section 4 – Summary

Background/History of MRP-211, Revision 0

- MRP-211, Revision 0 report sections (cont.):
  - Appendices (contain the actual raw data)
    - Tensile property data (Appendix A)
    - Fracture toughness data (Appendix B)
    - Irradiation creep data (Appendix C)
    - Void swelling data (Appendix D)
    - IASCC data (Appendix E)
    - IASCC growth data (Appendix F)
    - Test specimen designs (Appendix G)
    - Composition (Appendix H)
    - References (Appendix I)

Background/History of MRP-211, Revision 0

- Process:
  - EPRI MRP reports (extensive data compilations) were published in 2001-2005 and other available literature data sources dating to 1950s were initially evaluated
  - Expert panel review of the data in these reports was performed
  - Remaining data gaps were identified after this 2007 review
    - (NRC Suppl. Question # 1.c)
  - A summary of the data applicable to PWR internals was prepared

Background/History of MRP-211, Revision 0

- Data collected were compared to constitutive models in MRP-135, Revision 0
- Updates and changes made as necessary to models to fit data
- Material properties were characterized as a function of neutron fluence and temperature to the extent possible
  - Fracture toughness, irradiation-enhanced stress relaxation/creep, and IASCC crack initiation data were characterized using a lower bound approach
  - Limitations were identified
  - Precision of models limited by breadth of database
- MRP-135 was updated to Revision 1 (2010) incorporating the MRP-211, Revision 0 model recommendations

MRP-211, Revision 1 Irradiated Materials Database and Models 2016-2017

Sarah Davidsaver, *Framatome*

MRP-211, Revision 1 Irradiated Materials Database and Models

- NRC Staff requested MRP-211, Revision 1 (and MRP-175, Revision 1) on November 29, 2017 [ML17307A156]
  - To support SLR implementation of MRP-227, Revision 2
- EPRI transmitted proprietary and non-proprietary versions of both documents on December 18, 2017 [ML17361A187]
  - MRP-227 roadmap, developed in 2010, was also included for reference
- This presentation summarizes the current state-of-the-technology of <u>neutron irradiation-induced</u> property changes in austenitic stainless steels and recommended degradation models provided in MRP-211, Revision 1
  - Comparisons are made using MRP-211 Rev.0 figures and changes
- More detailed discussions of age-related degradation mechanisms (ARDMs) are in MRP-175

MRP-211, Revision 1 Irradiated Materials Database and Models

- Approach:
  - Expert Panel elicitation
  - Relevant new data sources since 2007 publication of MRP-211, Revision 0 identified and gathered
    - Environmental-Degradation Conferences
    - Fontevraud Conferences
    - Other literature sources (e.g., Journal of Nuclear Materials)
    - EPRI MDM and IMT documents
    - NRC NUREG/CR and PMDM documents
    - EPRI Materials Handbook
    - Recent Framatome/Westinghouse evaluation reports
    - PWR Owners Group reports
    - ICG-EAC meetings
    - EPRI BWRVIP, MRP and PSCR reports and meetings

MRP-211, Revision 1 Irradiated Materials Database and Models

- Approach (cont.):
  - Critically review and analyze the most recently available irradiated material data
  - Critically assess data fit to available models
    - Identify alternative formulations recommended and reach consensus, as appropriate
  - Identification of remaining gaps in database for potential future actions

{NRC Suppl. Question # 1.c}

MRP-211, Revision 1 Irradiated Materials Database and Models

- MRP-211, Revision 1 published October 2017
  - Full citation:

    Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data—State of Knowledge (MRP-211, Revision 1). EPRI, Palo Alto, CA: 2017. 3002010270.
  - EPRI Proprietary document
- Prepared by AREVA NP, with contributions from Westinghouse, SIA (formerly ANATECH), Vattenfall, EDF, Dominion, Southern Nuclear, Exelon, and EPRI
  - Steve Fyfitch, Sarah Davidsaver, David Burak, Daniel Brimbal, Josh McKinley, Randy Lott, Michael Burke, Michael Ickes, Greg Troyer, Ryan Hosler, Joe Rashid, Nathan Capps, Pal Efsing, Faiza Sefta, Jean-Paul Massoud, Glenn Gardner, Tim Wells, Heather Malikowski, Kyle Amberge, Jean Smith, Cem Topbasi, Peter Chou

MRP-211, Revision 1 Irradiated Materials Database and Models

- Selection of age-related degradation mechanisms (ARDMs)
  - Based on EPRI's Materials Degradation Matrix (MDM)
  - Non-irradiated degradation modes addressed in MRP-175:
    - SCC (IG/TG)
    - Wear
    - Fatigue (HC)
    - Reduction in Fracture Properties (Th)
  - Others addressed in MRP-211
  - General corrosion mechanisms are not pertinent to austenitic stainless steels due to PWR water chemistry controls

Table 3-2: PWR Reactor Vessel Internals

MATERIAL	DEGRADATION MODE													
	Corrosion				Wear	SCC		Fatigue		Reduction in Fract Properties		Irradiation Effects		
	Wstg	Pitting	FAC	Foul	Wear	IG/TG	IA	HC	EAF	Th	Env	Emb	VS	IC / SR
STRUCTURAL COMPONENTS & WELDS														
SS: 300 Series Base Metal & HAZ	N	N	N	N	Y IMP p2-5a	Y p2-6a	Y LTO p2-7a	Y p2-8a	Y LTO p2-9a	N	Y p2-11a	Y p2-12a	Y LTO p2-13a	Y p2-14a
SS: 300 Series Welds & Clad	N	N	N	N	N	Y p2-6b	Y LTO p2-7b	Y p2-8b	Y LTO p2-9b	Y p2-10b	Y p2-11b	Y p2-12b	Y LTO p2-13b	Y p2-14b
Cast Austenitic Stainless Steel	N	N	N	N	N	Y p2-6c	? LTO p2-7c	Y p2-8c	Y LTO p2-9c	Y p2-10c	Y p2-11c	Y LTO p2-12c	N	N
Ni-Alloy: A600 Base Metal	N	N	N	N	Y IMP p2-5d	Y LTO p2-6d	N	Y p2-8d	Y LTO p2-9d	N	Y p2-11d	N	N	N
FASTENERS & HARDWARE														
SS: 300 Series (304, 347, 316CW)	N	N	N	N	N	Y p2-6e	Y LTO p2-7e	Y p2-8e	Y LTO p2-9e	N	Y p2-11e	Y p2-12e	Y LTO p2-13e	Y LTO p2-14e
SS: A-286 Precip. Hardened SS	N	N	N	N	Y p2-5f	Y p2-6f	Y LTO p2-7f	Y p2-8f	Y LTO p2-9f	N	Y p2-11f	Y LTO p2-12f	N	Y LTO p2-14f
SS: Martensitic (Tp. 403, 410, 431, 17-4PH, 15-5PH)	N	N	N	N	Y p2-5g	Y p2-6g	N	Y p2-8g	Y LTO p2-9g	Y p2-10g	Y p2-11g	N	N	N
Ni-Alloy: X-750	N	N	N	N	Y p2-5h	Y p2-6h	Y LTO p2-7h	Y p2-8h	Y LTO p2-9h	N	Y p2-11h	Y LTO p2-12h	N	N

EPRI Materials Degradation Matrix, Revision 3. EPRI, Palo Alto, CA: 2013. 3002000628.

MRP-211, Revision 1 Irradiated Materials Database and Models

- Comparison between MRP-175 ARDMs and MRP-211 database/models
  - MRP-211 contains data used for MRP-135 constitutive irradiation effects model development
  - ARDMs not requiring irradiated property constitutive models, such as SCC, wear, HC fatigue, and reduction in fracture properties (Th) are not included in MRP-211

ARDM	MRP-175	MRP-211
SCC	Appendix A	Not irradiated property
IASCC	Appendix B	Section 2.5 and Appendix E (Initiation) Section 2.6 and Appendix F (Growth)
Wear	Appendix C	Not irradiated property
Fatigue	Appendix D	Section 2.8 and Appendix H
Thermal Embrittlement	Appendix E	Section 2.7 and Appendix G
Irradiation Embrittlement	Appendix F	Section 2.1 and Appendix A (Mechanical Properties) Section 2.2 and Appendix B (Fracture Toughness)
Void Swelling	Appendix G	Section 2.4 and Appendix D
Stress Relaxation and Irradiation Creep	Appendix H	Section 2.3 and Appendix C

MRP-211, Revision 1 Irradiated Materials Database and Models

- MRP-211 summarizes the available data that describes the current state-of-knowledge of neutron irradiation-induced property changes in austenitic stainless steels, principally:
  - Solution-annealed Type 304 and 304L
  - Cold-worked Type 316 and 316L
  - Grades CF3/CF3M and CF8/CF8M cast austenitic stainless steels
  - Austenitic stainless steel weld metals (e.g., Type 308)
- Age-related degradation mechanisms include:
  - Irradiation embrittlement (IE)
  - Thermal and Irradiation-enhanced stress relaxation/creep (ISR/IC)
  - Void swelling (VS)
  - Irradiation-assisted stress corrosion cracking (IASCC)
  - Fatigue including EAF

MRP-211, Revision 1 Irradiated Materials Database and Models

- MRP-211, Revision 1 supports:
  - Evaluations and assessments required to refine/update categorization and ranking of PWR internals components and items
  - SLR I&E guideline (i.e., MRP-227, Revision 2) development
- Update to MRP-211 used same process that generated Rev.0
- Updated MRP-135 constitutive models will be based on recommendations in MRP-211, Revision 1
  - Revision scheduled for mid-2018

MRP-211, Revision 1 Irradiated Materials Database and Models

- Fundamental differences between MRP-211, Revision 1 and MRP-211, Revision 0:
  - Section 1 – Introduction
    - Updated each sub-section, as appropriate
  - Section 2 – Data (sources, trends, gaps, model, and references)
    - Tensile test data (Section 2.1)
    - Fracture toughness data (Section 2.2)
    - Thermal and irradiation creep/stress relaxation data (Section 2.3)
    - Void swelling data (Section 2.4)
    - IASCC initiation data (Section 2.5)
    - IASCC growth data (Section 2.6)
    - Combined TE and IE data (Section 2.7)
    - Fatigue (including EAF) life data (Section 2.8)

MRP-211, Revision 1 Irradiated Materials Database and Models

- Fundamental differences between MRP-211, Revision 0 and MRP-211, Revision 1 (cont.):
  - Section 3.0 – Summary
    - Previous section 3.0 was completely revised and model recommendations were incorporated into current Section 2.0
  - MRP-211, Revision 0 Section 4.0 – Summary
    - No longer required in MRP-211, Revision 1
  - Appendices - tables updated with new data
    - Tensile property data (Appendix A)
    - Fracture toughness data (Appendix B)
    - Irradiation creep data (Appendix C)
    - Void swelling data (Appendix D)
    - IASCC data (Appendix E)
    - IASCC growth data (Appendix F)
    - <u>TE and IE data (Appendix G)</u>
    - <u>Fatigue (including EAF) data (Appendix H)</u>
    - Test specimen designs (Appendix I)
    - Composition (Appendix J)
    - References (Appendix K)

Break

MRP-211, Revision 1 Irradiated Materials Database and Models

Irradiated Materials Database

MRP-211, Revision 1 Irradiated Materials Database and Models

Table 1-1
Data sources for materials irradiated in test reactors

Reactor	Spectrum	Temperature, °C (°F)	Maximum Dose, dpa (n/cm^2, E > 1.0 MeV)	Materials	Data Type
EBR-II	Fast	~375 (707)	~30 (~2.00 × 10^{22})	304SA, 316CW	Tensile, creep, fracture toughness
Phénix	Fast	380 (716) to 400 (752)	~30 (~2.00 × 10^{22})	304SA, 316CW	Tensile, creep, fracture toughness
BOR-60	Fast	~320 (608)	~120 (~8.00 × 10^{22})	304SA, 316CW, 308L weld, HAZ	Tensile, creep, fracture toughness, corrosion
SM-2	Mixed (fast + thermal)	~300 (572)	~15 (~1.00 × 10^{22})	304SA, 316CW	Tensile, corrosion
Osiris	Mixed (fast + thermal)	~320 (608)	~12 (~8.00 × 10^{21})	304SA, 316CW	Tensile, creep, corrosion

Note: SA = solution-annealed; CW = cold-worked.

MRP-211, Revision 1 Irradiated Materials Database and Models

Table 1-2
Data sources for materials irradiated in PWRs

Material Component/Source	Material	Laboratory	Temperature, °C (°F)	Dose, dpa (n/cm^2, E > 1.0 MeV)
Bolt (Bugey 2)	316L CW	EDF	330 (626) to 360 (680)	10–25 (~6.67×10^{21}–1.67×10^{22})
Bolt (Fessenheim 2)	316L CW	EDF	330 (626) to 360 (680)	12–25 (~8.00×10^{21}–1.67×10^{22})
Bolt (Tihange 1)	316L CW	PNNL EDF	300 (572) to 363685	7–24 (~4.67×10^{21}–1.60×10^{22})
Bolt (Farley)	316 CW	Westinghouse BNFL	307 (584) to 396 (745)	9–19 (~6.00×10^{21}–1.27×10^{22})
Bolt (Point Beach)	347 SA	Westinghouse BNFL	303 (577) to 396 (745)	4–15 (~2.67×10^{21}–1.00×10^{22})
Locking bar (Fessenheim 2)	304 SA	EDF	330 (626)	25 (~1.67×10^{22})
Locking bar (Farley)	304 SA	Westinghouse BNFL	304 (579)	20–22 (~1.33×10^{22}–1.47×10^{22})
Locking clip (Point Beach)	304 SA	Westinghouse BNFL	304 (579)	23–26 (~1.53×10^{22} to 1.73×10^{22})
Core barrel (U.S. PWR Plant)	304 SA	Westinghouse	280 (536) to 294 (561)	0.5 (~3.33×10^{20})
Baffle plate (U.S. PWR Plant)	304 SA	Westinghouse	285/545 to 307/584	4–17 (~2.67×10^{21}–1.13×10^{22})
Former plates (U.S. PWR Plant)	304 SA (4 heats)	Westinghouse	301 (574) to 321 (610)	9–13 (~6.00×10^{21}–8.67×10^{21})
Baffle corner (Chooz A)	304 SA	EDF	~300 (572)	~ 35 (~2.33×10^{22})
Baffle bolt (Chooz A)	304 CW	CEA	~300 (572)	2.5–21 (~1.67×10^{21}–1.40×10^{22})
Thimble tube (Beaver Valley 1)	316 CW	Westinghouse	340 (644)	12–51 (~8.00×10^{21}–3.40×10^{22})
Thimble tube (HB Robinson 2)	316 CW	Westinghouse	340 (644)	20–30 (~1.33×10^{22}–2.00×10^{22})

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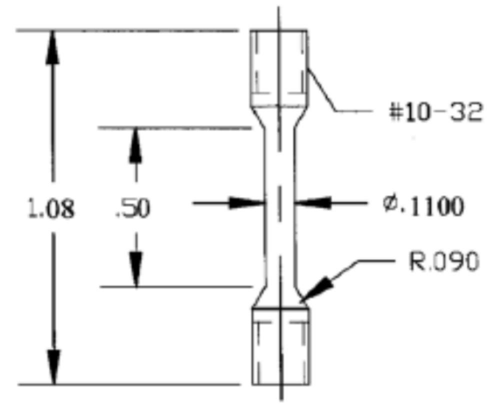
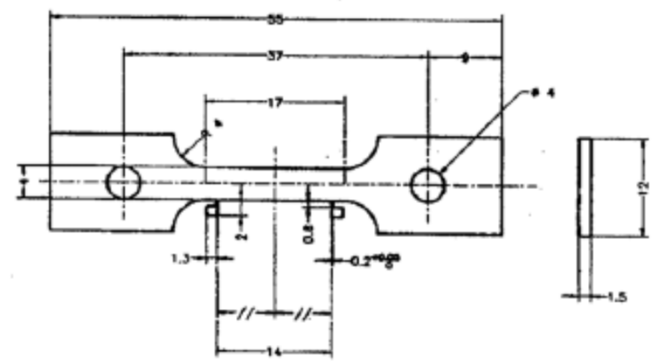
MRP-211, Revision 1 Irradiated Materials Database and Models (Tensile)

- Tensile properties
  - Irradiation produces defects and precipitates that form obstacles to dislocation movement, which causes:
    - an increase in yield strength (YS) and ultimate tensile strength (UTS)
    - a decrease in uniform elongation (%UE) and total elongation (%TE)
  - Tensile properties of irradiated stainless steels have been widely investigated
    - Fast reactor irradiations
    - Thermal reactor irradiations
    - Data trends are consistent
  - Majority of available data from fast reactors; however, significant LWR data are now available for comparison and evaluation





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MRP-211, Revision 1 Irradiated Materials Database and Models (Tensile)

- Tensile properties
  - Data compiled in Section 2.1 and Appendix A
  - Figures 2-2 through 2-30 show data
    - Room and high temperature, all properties
  - 28 additional references added
    - EPRI MRP and BWRVIP reports, conference proceedings, journals, PhD thesis, Halden reactor project reports, NUREG/CR report
    - Data has been extended from 124 dpa to 231 dpa
- **Final conclusion is unchanged from MRP-211, Rev. 0**
  - Data demonstrate that tensile properties saturate by a neutron exposure of ~20 dpa (~1.33E22 n/cm^2, E > 1 MeV) *(answers NRC suppl. question #1.b)*

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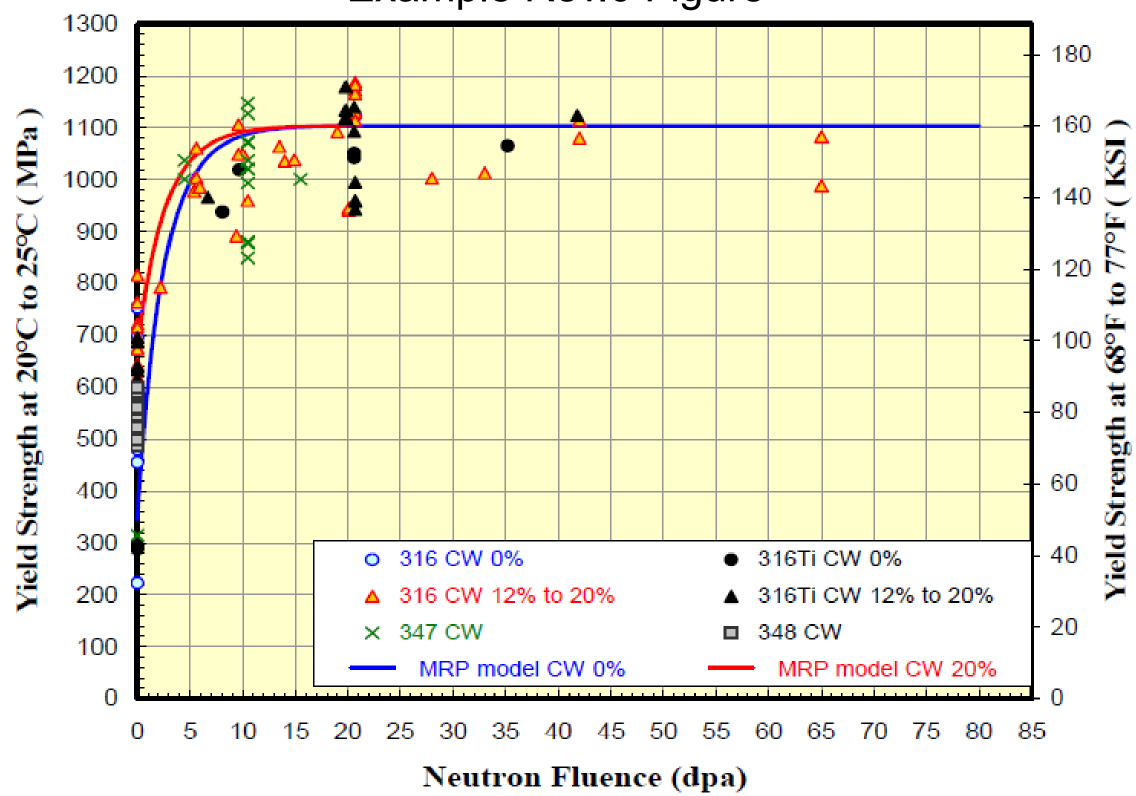
MRP-211, Revision 1 Irradiated Materials Database and Models (Tensile)

- MRP-211, Revision 1 contains:
  - Significantly more data in the 0-20 dpa and 140-231 dpa ranges
  - Additional data to fill in gaps in 30-50 dpa range
  - Data obtained from:
    - Fast reactors and thermal reactors
    - Low and high test temperatures
  - These data will be used in the current MRP-135 models

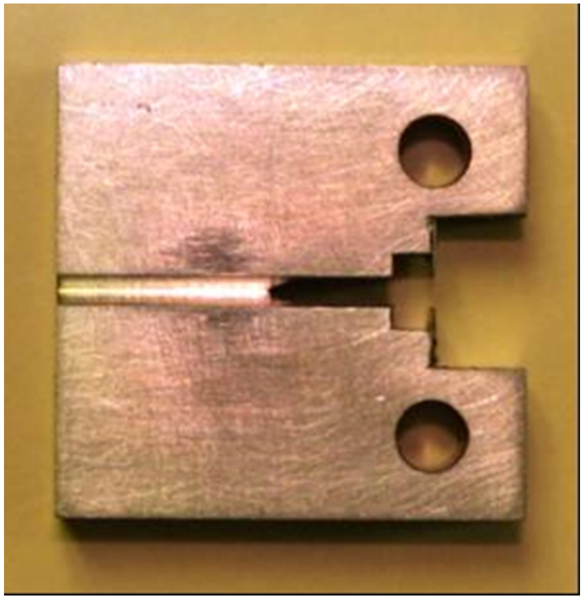
Example Rev.0 Figure

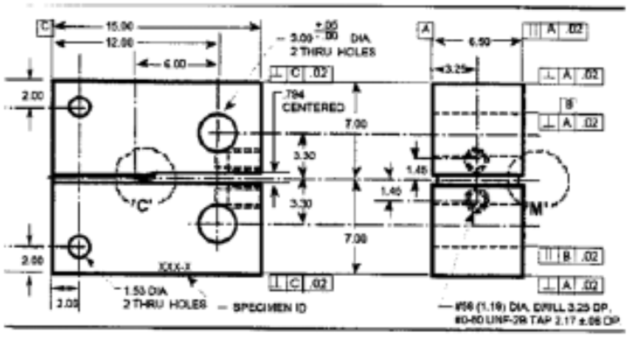


The Effect of Neutron Fluence on Room Temperature Yield Strength for Solution-Annealed and Cold-Worked Type 316, Type 347, and Type 348 Stainless Steels (MRP-211 Rev. 0 Figure 2-2)

MRP-211, Revision 1 Irradiated Materials Database and Models (Fracture Toughness)

- Fracture toughness
  - Irradiation produces defects and precipitates, which causes a reduction of fracture toughness in austenitic stainless steels
  - The fracture toughness of stainless steel saturates when exposed to high-energy neutrons
  - The reduction of fracture toughness with increasing neutron dose in both BWRs and PWRs is consistent with that observed in fast reactors
  - Majority of available data from fast reactors; however, significant LWR data are now available for comparison and evaluation





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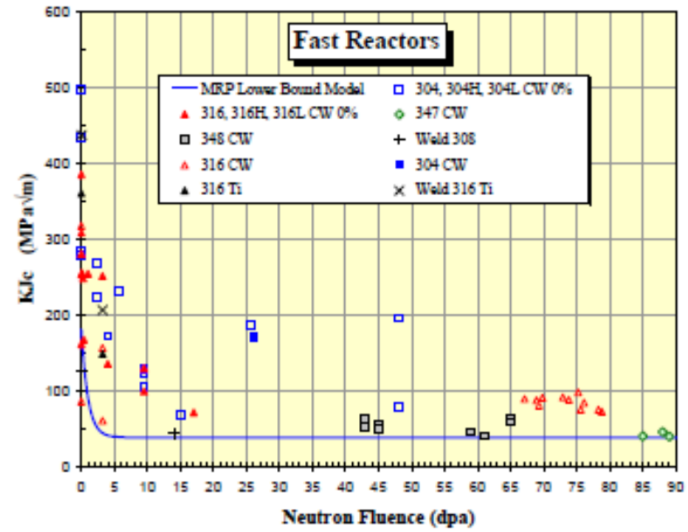
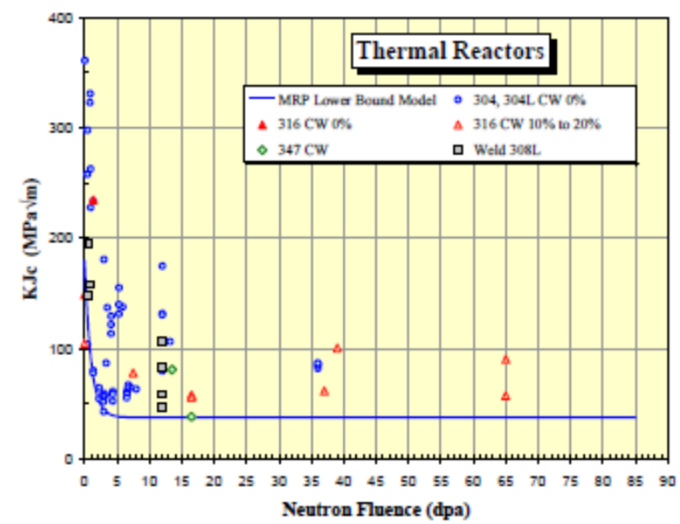
MRP-211, Revision 1 Irradiated Materials Database and Models (Fracture Toughness)

- Fracture toughness properties
  - Data compiled in Section 2.2 and Appendix B
  - Figures 2-31 through 2-34 show data
  - 16 additional references added
    - Conference proceedings, journals, NUREG/CR, EPRI BWRVIP and MRP letter and report, NRC final safety evaluation
    - New data mainly in the 0 to 15 dpa range
- Fracture toughness data with VS up to ~8% is similar to other data at lower doses without significant swelling *(NRC suppl. question # 5.b)*
- **The conclusion is unchanged from MRP-211, Rev. 0**
  - All data remain bounded by a saturated value of 38 MPa√m (34.6 ksi√in) for fluence greater than about 10 dpa (~6.67E21 n/cm^2, E>1 MeV)

Example Rev.0 Figures





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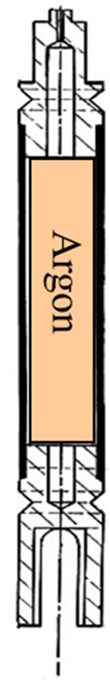
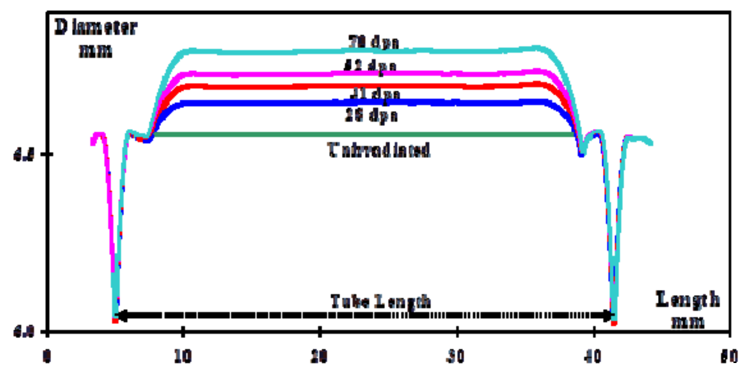
MRP-211, Revision 1 Irradiated Materials Database and Models (Thermal and Irradiation Creep/Stress Relaxation)

- Thermal and irradiation stress relaxation/creep
  - Thermal stress relaxation (creep) occur as a result of mobile vacancies, dislocation climb, and a material's defect structure
    - Conventionally, these occur above ~$0.5T_m$
  - However, under irradiation conditions such processes can occur at lower temperatures
  - ISR/IC depends mainly on the neutron fluence and the applied stress
  - Manifests itself in relaxation of preloaded components (potentially leading to excessive wear or fatigue)
    - Evident when BFBs are removed (based on removal torque)





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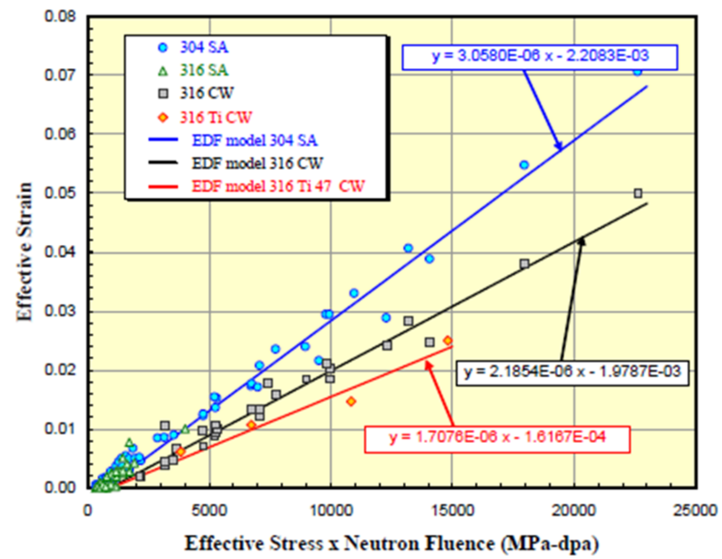
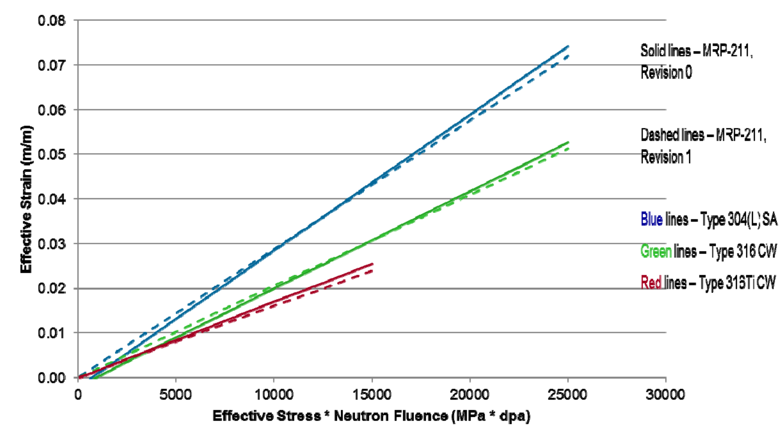
MRP-211, Revision 1 Irradiated Materials Database and Models (Thermal and Irradiation Creep/Stress Relaxation)

- Thermal and irradiation creep/stress relaxation (ISR/IC) properties
  - Data compiled in Section 2.3 and Appendix C
  - Figures 2-39 and 2-40 show data
  - 14 additional references added
    - Conference proceedings, journals, EPRI MRP report, EdF report
  - Creep compliance values [linear correlation of (effective creep strain) with (effective stress * irradiation dose)] slightly changed by new data
    - Dashed line versus solid line change is insignificant
- **The conclusion is unchanged from MRP-211, Rev. 0**
  - Correlations still indicate that a greater creep rate occurs for Type 304 SA material than for Type 316 CW material

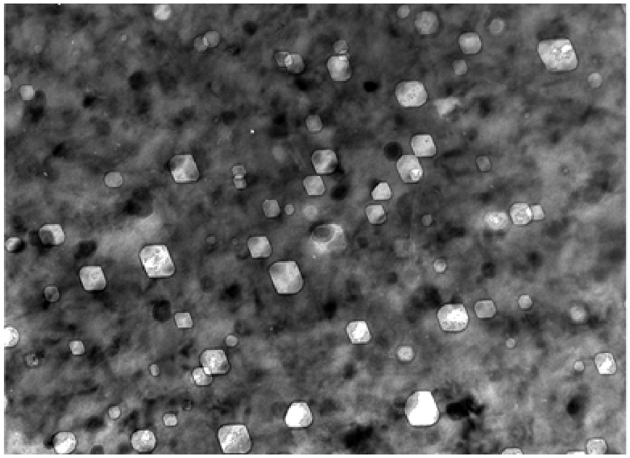
Example Rev.0 Figures





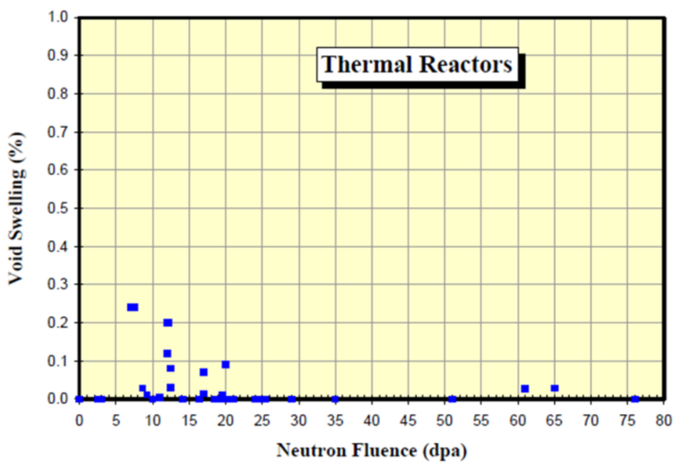
MRP-211, Revision 1 Irradiated Materials Database and Models (Void Swelling)

- Void swelling
  - Irradiation causes displacement of atoms from their lattice sites leading to formation of cavities (or, voids), which causes:
    - Volume and dimensional changes
    - Potential distortions of structural components
    - Potential for a reduced tearing modulus at PWR operating temperatures that may fall to zero at room temperature
  - VS is strongly sensitive to dose, dose rate, irradiation temperature, and material composition
  - Fast reactor data are insufficient to estimate VS trends in LWRs because different temperatures, damage rates, and helium production rates
  - PWR irradiation spectrum data show very little void swelling although data are limited

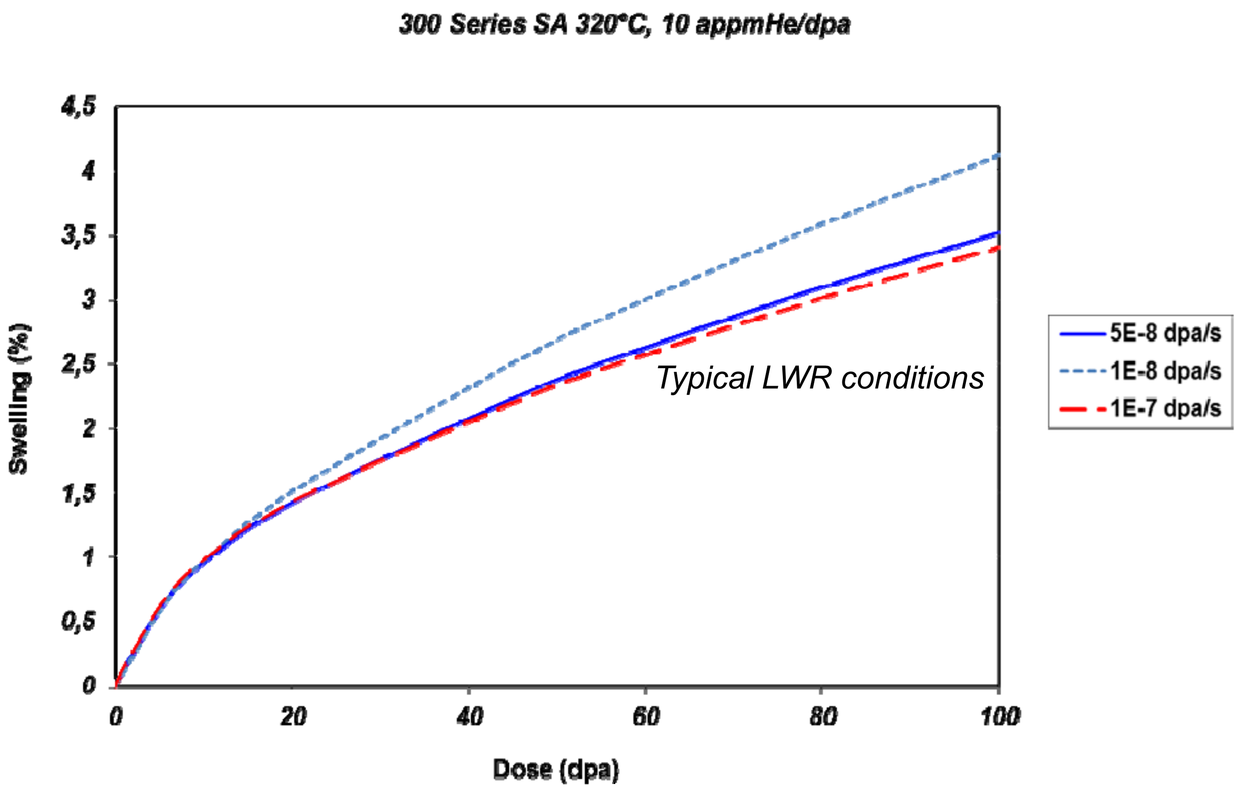


MRP-211, Revision 1 Irradiated Materials Database and Models (Void Swelling)

- Void swelling properties
  - Data compiled in Section 2.4 and Appendix D
  - Figure 2-41 shows data
  - 32 additional references added
    - Conference proceedings, journals, EPRI MRP reports, ORNL report, PhD thesis
  - A cluster-dynamics-based VS model was developed through another EPRI-sponsored project and is recommended for calculation of VS in PWR environments (MRP-391)
- **This model is added to MRP-211, Rev. 1**
  - The VS model indicates that steady-state swelling rates of less than 0.1 %/dpa are reasonable for the fluence levels and temperatures expected in PWR internals during SLR
  - Data of extracted PWR internals components and/or materials are in agreement with the cluster-dynamic model



MRP-211, Revision 1 Irradiated Materials Database and Models (Void Swelling)

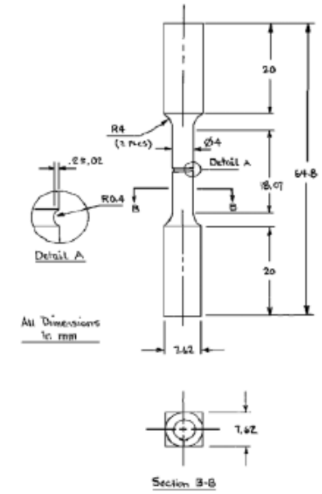


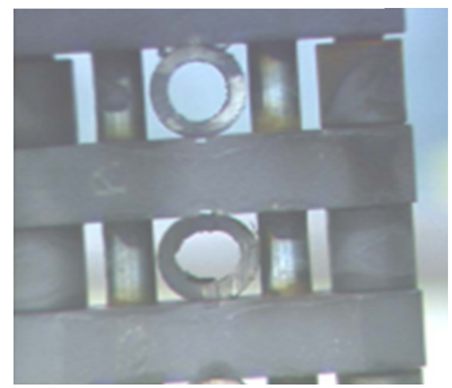
Swelling Predictions vs. dpa for Solution Annealed 300 Series Stainless Steels at 320°C (608°F) at a Helium-to-dpa ratio of 10 appm He/dpa, Various Displacement Rates Relevant to a PWR Environment (Davidsaver et al., 2017 Environmental Degradation Conference)

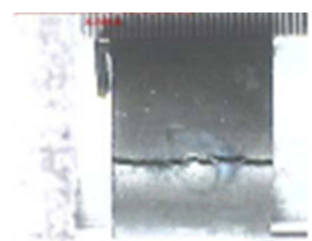
(NRC Suppl. Question 5.a and 5.b)

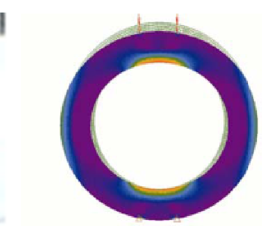
MRP-211, Revision 1 Irradiated Materials Database and Models (IASCC Initiation)

- IASCC initiation properties
  - IASCC results from a combined effect of irradiation damage to the material, stress state and water environment
    - No single mechanism that controls IASCC initiation has been identified
  - IASCC initiation data show both initiation and no-initiation domains









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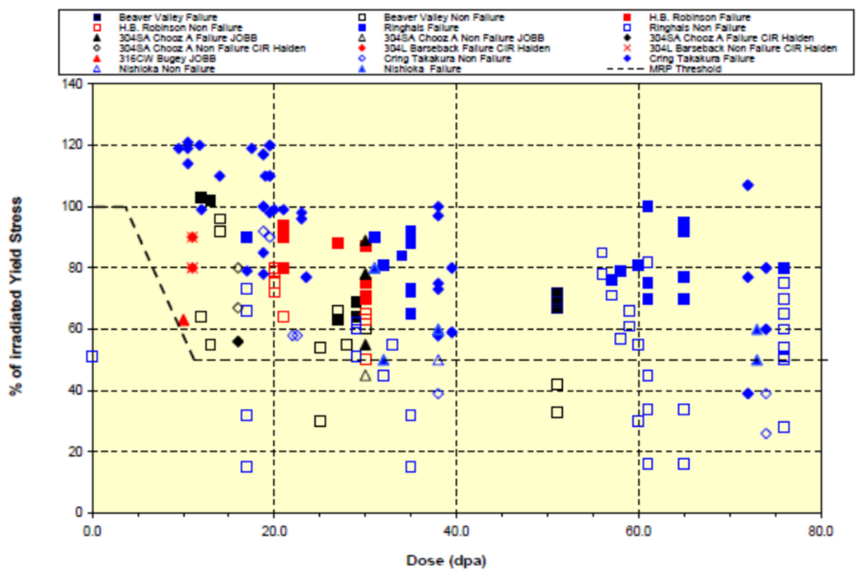
MRP-211, Revision 1 Irradiated Materials Database and Models (IASCC Initiation)

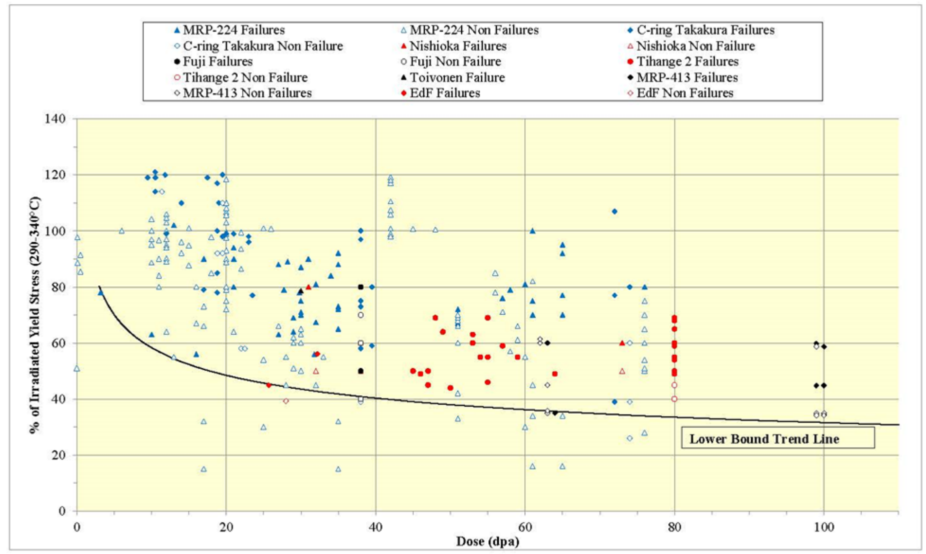
- IASCC initiation properties
  - Data compiled in Section 2.5 and Appendix E
  - Figures 2-45 through 2-47 show data
  - 19 additional references added
    - Conference proceedings, journals, Halden reports, NUREG/CR
  - Laboratory test data indicate that IASCC initiation susceptibility appears to continue to increase with irradiation damage, even though the tensile properties appear to saturate by 20 dpa (~1.33E22 n/cm^2 , E > 1.0 MeV),
- **Change made from MRP-211, Rev. 0**
  - IASCC crack initiation may not occur in materials irradiated to about 80 dpa (~5.33E22 n/cm^2 , E > 1.0 MeV) when component loaded to below approximately 35% of irradiated yield strength
    - Lower bound trending model adjusted

MRP-211, Revision 1 Irradiated Materials Database and Models (IASCC Initiation)

IASCC initiation and no-initiation domains



IASCC Flaw Initiation Stress Versus Dose – Constant Load Tests (MRP-211 Rev. 0 Figure 2-38)

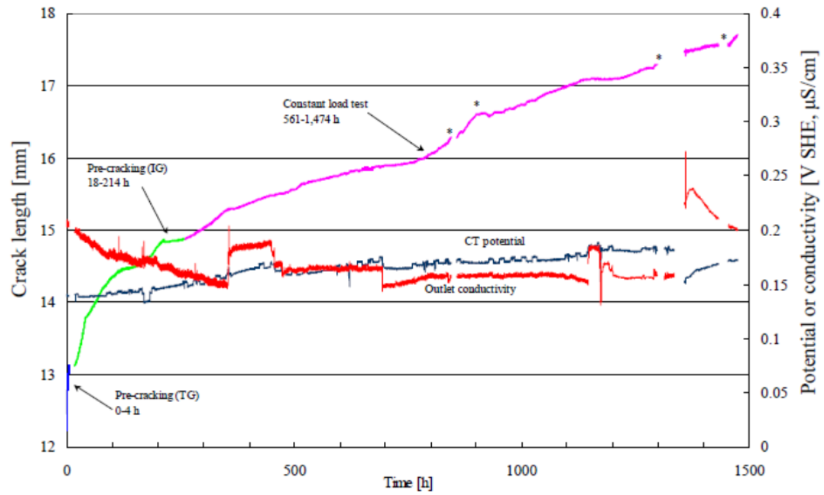


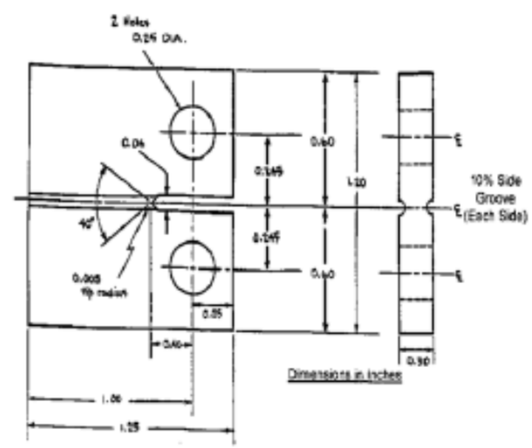
Source: Fyfitch et al., 2017 Environmental Degradation Conference

NRC Suppl. Question # 3
Initiation = above trend line
No-Initiation = below trend line

MRP-211, Revision 1 Irradiated Materials Database and Models (IASCC Growth)

- IASCC growth properties
  - Subcritical crack growth by IASCC occurs as a result of three factors:
    - Stress intensity
    - Neutron fluence level
    - Temperature
  - IASCC Expert Panel to gather, review, and evaluate crack growth data for both PWRs and BWRs
  - Data then used to create recommended CGR disposition curves
  - CGR curves were submitted to the ASME Code in a Code Case for use by the industry
  - Core barrel welds expected to be well below maximum fluence of the CGR dataset *{NRC Suppl. Question #4}*





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MRP-211, Revision 1 Irradiated Materials Database and Models (IASCC Growth)

- IASCC growth properties
  - Data compiled in Section 2.6 and Appendix F
  - Figures 2-48 through 2-49 show example disposition lines
  - All references removed, only one reference used
    - *Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments: Volume 1: Disposition Curves Development and Volume 2: Disposition Curves Application*. EPRI, Palo Alto, CA: 2014. 3002003103.
  - Disposition curves based on stress intensity factor, K
    - Applicable to flaw evaluations in PWR and BWR plants
- **This is an addition to MRP-211, Rev. 1**

MRP-211, Revision 1 Irradiated Materials Database and Models (IASCC Growth)

- Example IASCC crack growth rates (PVP2015-45323)

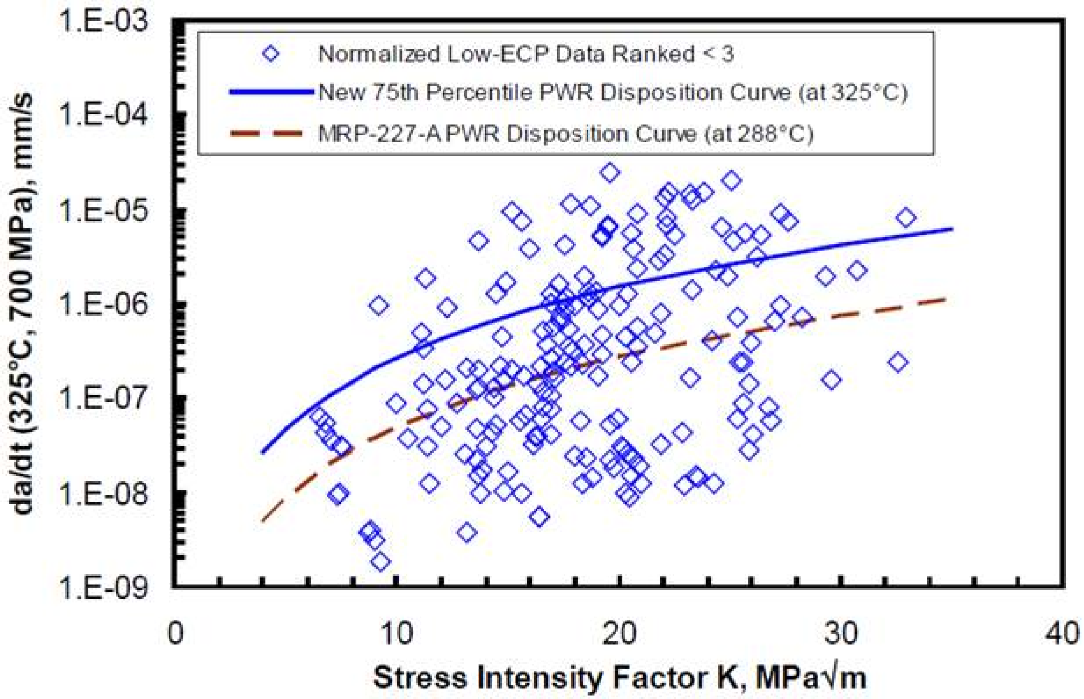
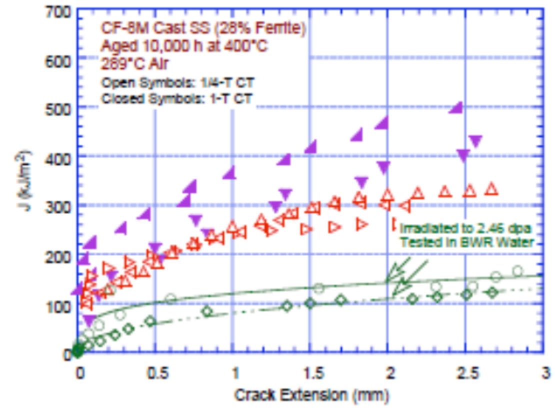
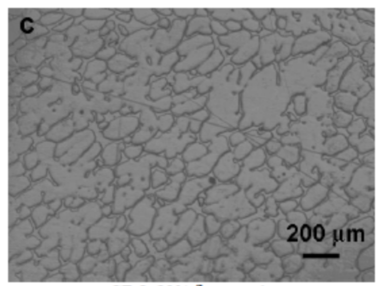


FIG. 3 NEW PWR PRIMARY WATER DISPOSITION CURVE (SOLID) AT 325°C AND 700 MPA (~4.3 DPA) AND MRP-227-A PWR CURVE [11] (DASHED) COMPARED WITH NORMALIZED LOW-ECP CALIBRATION DATA

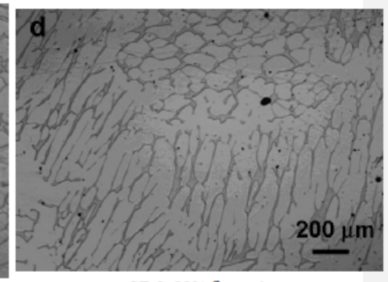
MRP-211, Revision 1 Irradiated Materials Database and Models (Combined TE and IE of CASS)

- TE and IE properties
  - The combined effect of TE and IE is a time and dose dependent process whereby a material undergoes microstructural changes leading to decreased ductility and degradation of toughness and impact properties
  - CASS is a two-phase material consisting primarily of an austenite matrix with the remainder being ferrite
  - Both the austenite and the ferrite are affected by irradiation
    - Only the ferrite is affected by temperature
  - Limited data are available on the combined effects of these two embrittlement mechanisms





CF-8, 23% δ, unaged



CF-8, 23% δ, aged

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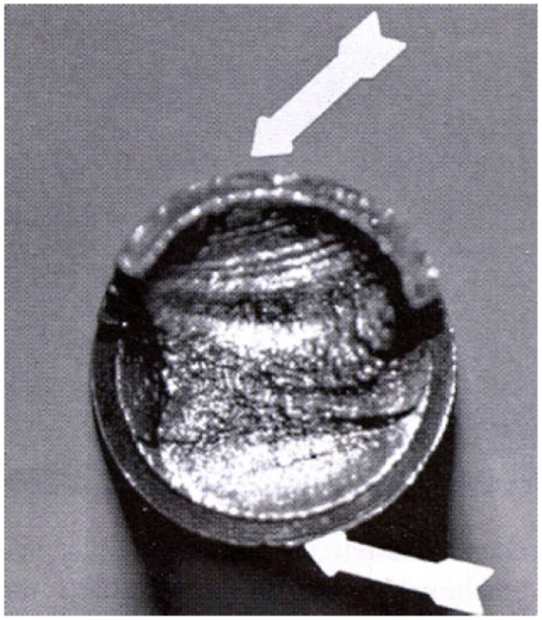
MRP-211, Revision 1 Irradiated Materials Database and Models (Combined TE and IE of CASS)

- TE and IE properties
  - Data compiled in Section 2.7 and Appendix G (New in Revision 1)
  - Figure 2-50 shows data
  - Data from NUREG/CR reports and one paper from conference proceedings
    - Very limited data
  - Differences observed in material types, testing environments, and temperature
    - CF3, CF3M, CF8, and CF8M
    - Air versus water environment
    - Room temperature versus service temperature
  - CASS are shown to be susceptible to loss of toughness by combined thermal embrittlement and IE, which is shown to depend on the extent of the ferrite phase
    - Assessments of CASS have shown that PWR reactor internals components are not significantly impacted by TE/IE (Ref. ML16250A001)
- **No corresponding section in MRP-211, Rev. 0**
  - A lower bounding curve has been developed for MRP-211, Rev.1

BREAK

MRP-211, Revision 1 Irradiated Materials Database and Models (Fatigue Life)

- Fatigue properties
  - Fatigue is a process involving the evolution of persistent slip bands at the surface of a material and subsequent crack formation and propagation during exposure to cyclic stresses
  - Low-cycle fatigue (LCF) is associated with plastic strains
  - High-cycle fatigue (HCF) occurs at stresses below the elastic limit
  - For both cases, the environment can impact the final fatigue results and this is known as environmentally-assisted fatigue (EAF)



(Arrows denote the fatigue initiation sites)

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MRP-211, Revision 1 Irradiated Materials Database and Models (Fatigue Life)

- Fatigue life properties
  - Data compiled in Section 2.8 and Appendix H (New in Rev. 1)
  - Figures 2-51 through 2-53 show data
  - Data from NUREG/CR-6909 report, conference proceeding, and EPRI MRP report
- **No corresponding section in MRP-211 Rev. 0**
  - The expert panel recommended applying existing methods for evaluating fatigue life on irradiated materials with a suggested environmental correction in accordance with NUREG/CR-6909, Revision 1
    - As more test data are gathered, this approach may be updated

MRP-211, Revision 1 Irradiated Materials Database and Models

Irradiated Materials Models

MRP-211, Revision 1 Irradiated Materials Database and Models

- MRP-211, Revision 1 major changes:
  - Void swelling – Updated to incorporate cluster dynamics-based model
  - Irradiation creep – Change in creep compliance terms and removal of incubation period
  - IASCC Initiation – Updated to be consistent with the conservative fit to the database
  - IASCC Growth – Use of 2014 industry developed disposition curves
  - TE & IE (new) – Bounding trend line developed for available CASS data
  - Fatigue life (new) – Based on NUREG/CR-6909 Revision 1 environmental correction factors

Summary and Conclusions

- MRP-211, Revision 0 was published in 2007
  - Summarized current knowledge of irradiated stainless steel properties
  - Provided recommended models for changes to MRP-135 constitutive models
  - Used to refine screening and categorization results through engineering evaluations and assessments
- Additional testing and operating experience data from the last ~10 years since publication have been gathered
- MRP-211, Revision 1 was published in 2017
  - Updated the database to include recent data, address gaps as applicable, update models
  - Include new sections for TE + IE data and fatigue (including EAF) data
  - Recommended model changes including updates and new models
- Revision 1 models will be used in developing MRP-227, Revision 2 for SLR and through a revision to MRP-135, Revision 1

Summary and Conclusions

- Expert panel validated the 2007 materials property assessments in MRP-211 Revision 0
  - Additional testing and operating experience data from the last ~10 years confirm many of the same conclusions in MRP-211, Revision 0
  - Additional testing and operating experience data from the last ~10 years provide improvements to MRP-135 models due to a more extensive database of neutron-induced ARDMs
  - Data for two additional ARDMs (combined TE + IE) and fatigue (including EAF) were added to the database
- Industry provided MRP-211 Revision 1 to NRC for info to foster continued technical exchange with staff

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