Update to Aging Degradation Mechanism Thresholds and Screening

MRP-175, Revision 1

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NRC Public Meeting, Rockville, MD
February 14, 2018

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 2/14/18

- Present discussion of PWR reactor internals material screening criteria including identification of updates and changes from MRP-175, Revision 0 (2005)
- Foster technical discussion with NRC
  - Review and discuss other materials related questions from staff
- Identify future meetings/topics and interactions with NRC

Introduction

- MRP-175, Revision 0 – “Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values” – published December 2005
  - Transmitted to the NRC in 2006 (ML061880278 – non-proprietary)
  - Full version now also available on www.epri.com
  - Meetings held on MRP-175, Rev. 0: ML053270247, ML061290492
  - MRP responded to NRC staff comments on MRP-175, Rev. 0 in April 2007 (ML071500462)
- MRP-175, Revision 1 was published October 2017
  - Requested by the NRC staff on November 29, 2017 (ML17307A156)
  - Transmitted to NRC on December 18, 2017 (ML17361A187)
  - Supports subsequent license renewal (SLR) development of MRP-227
- Discussion today covers the development and contents of the report, including the significant changes

Purpose of MRP-175

- Provides the technical basis for the aging-related degradation mechanisms applicable to the reactor internals
  - Summarizes data available from a large list of literature references
  - Documents the results of expert panel review of literature data, including the resulting aging degradation mechanism screening values
- Creates foundation for the screening step in developing MRP-227
- Supports many of the other MRP-227 development steps
  - Categorization and Ranking
  - Functionality analysis
  - Final conclusions on inspection and evaluation guidelines

Purpose of the Revision to MRP-175

- Original Rev.0 report was published in 2005
  - Significant amounts of new research and results have been published in the decade since original publication
  - New operating experience needed to be considered
  - First step was a literature review to incorporate new data and conclusions
  - Expert panel reviewed the results of this literature search
- Original report was directed at screening for 60 years
  - Limitation is unnecessary in many cases, so revision allowed broader application
  - Cases that were constrained by operating time were updated to apply to SLR and beyond
- MRP-175 is one piece in the overall living program of MRP-227 and is intended to be updated periodically
- Validation of previous report use by a modern review/update

MRP-175 Revision Process

- Process similar to MRP-175, Rev. 0 development
- Step 1: Convene expert panel
  - Define needed document revisions (gaps, clarifications)
  - Review new data, redefine screening criteria etc.
  - Provide direction on sources of and areas with new data
- Step 2: Authors assigned to each degradation mechanism
  - Gather references and updated report sections
  - Provide recommended revisions to the screening values
- Step 3: Reconvene expert panel
  - Review revised report sub-sections and provide comments
  - Provide any additional literature references needed
  - Draw 100% panel consensus agreement on the revised screening
- Step 4: Prepare and issue revised document

Expert Panel

- Panel membership
  - Participants from multiple organizations
    - EPRI, AREVA, EdF, Anatech, Dominion, Southern Nuclear, Westinghouse
  - Domestic and international representation
  - Members with expertise in multiple areas: nuclear materials, materials testing, internals design, simulation, etc.
- Review meetings
  - Two in-person and several teleconference meetings conducted in 2017
  - Opportunity for panel to review approach, assumptions, and revisions
  - Panel provided feedback on additional references and resources
  - Panel evaluated and came to full consensus on screening criteria
- Final document provided to panel and MRP utility members for review and concurrence

General Changes in MRP-175, Revision 1

- Clarified that conservative screening criteria and <u>not</u> threshold criteria are being used
- Incorporated new test data and OE results
- Reviewed and considered technical updates to screening criteria
- Removed references to 60 or 80 years in general
- Addressed NRC comments, RAIs, and A/LAIs
  - Comments and industry responses for MRP-175, Revision 0
  - Cold work
  - Embrittlement of CASS materials
  - Loss of fracture toughness due to irradiation embrittlement
  - Synergy of mechanisms
- Included editorial changes and corrections

Examples of Data Gaps Addressed

- Updated report based on data generated since 2007
  - Including recent data supporting revised IASCC screening criteria
  - Revising fatigue screening to include more recent EAF understanding
  - Referencing the information developed generically for CASS and CW
  - Incorporating new data for other degradation mechanisms which did require screening criteria changes
- Added relevant operating experience for mechanisms
  - Baffle-former bolts, CRGT guide cards, B&W vent valves, etc.
- Evaluated conservatism in estimating onset of effects
  - IE of wrought SS set at 1.5 dpa which is 30% of the 5 dpa allowable
  - VS of SS calculated as <1.5% at screening; conservative relative to actual measurements
- Confirmed void swelling screening with modern cluster dynamics model and data from LWR-relevant testing

Key Concepts for MRP-175

- Definitions
  - **Threshold Value** - The level of susceptibility when an aging *effect is first observable* or quantifiable
  - **Screening Value** - The level of susceptibility when an aging *effect may be significant* to functionality
- MRP-175 provides screening values
  - Defining thresholds for degradation mechanisms typically not possible
  - Screening values have more practical purpose since primary goal is maintaining functionality during PEO
    - Aging effect starting to become relevant to functionality with regards to nuclear safety
- Quantification of the screening criteria requires
  - Knowledge of the specific aging mechanisms
  - Engineering judgment
  - Empirical relations where data may be limited

Applicable Aging Degradation Mechanisms = 8

- Applicable Mechanisms:
  1. Stress Corrosion Cracking (SCC)
  2. Irradiation-Assisted Stress Corrosion Cracking (IASCC)
  3. Wear
  4. Fatigue
  5. Thermal Aging Embrittlement (TE)
  6. Irradiation Embrittlement (IE)
  7. Void Swelling (VS)
  8. Stress Relaxation and Irradiation Creep (SR/IC)

- Other mechanisms not applicable to reactor internals:
  - Loss of material due to pitting or crevice corrosion
  - Loss of material due to general corrosion
  - Loss of material due to erosion or flow accelerated corrosion
  - ➢ Not applicable with proper chemistry control in PWRs

Materials Applicable for Reactor Vessel Internals

- Stainless Steels (SS)
  - Austenitic (e.g., Types 304, 304L, 316, 316L, and 347)
  - Cast Austenitic (e.g., Grades CF8, CF8M, and CF3M)
  - Austenitic Precipitation-Hardenable (e.g., Alloy A-286)
  - Martensitic (e.g., Types 403, 410, and 431)
  - Martensitic Precipitation-Hardenable (e.g., Type 17-4 PH, 15-5 PH)
- Nickel-Base Alloys
  - Austenitic (e.g., Alloys 600, 82, and 182)
  - Austenitic Precipitation-Hardenable (e.g., Alloys 718 and X-750)
- Cobalt-Base Alloys
  - Stellite

MRP-175 Section Overviews and Revisions

MRP-175 Revision 0 Report Structure

- TOC and sections, etc.

Stress Corrosion Cracking (Appendix A)

- Covers most of the relevant forms of environmentally-assisted cracking
  - Stress corrosion cracking (SCC) [transgranular or intergranular]
  - Primary water SCC (PWSCC)
  - Low-temperature crack propagation (LTCP)
- Included IASCC, embrittlement effects, and EAF in other appendices
- Requires three factors to occur:
  - Tensile stress
  - Environmental conditions that cause SCC
  - Material susceptible to the environment
- Assumed that chemistry is controlled per primary water guidelines and that hydrogen additions are present

SCC TOC

SCC Data Sources

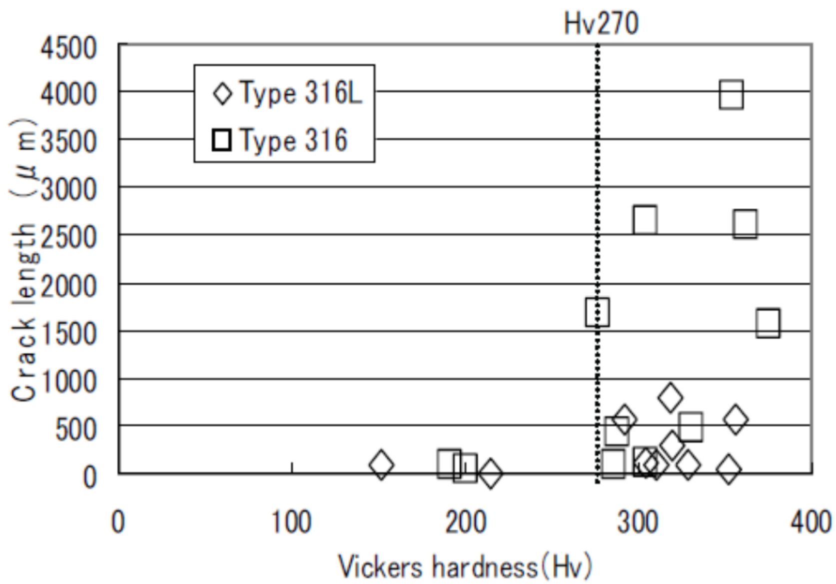
- Includes 85 references on SCC or containing SCC data
- Extensive information on austenitic SS from many sources
  - Multiple sources focused on SCC with contaminants (Navy, Gordon, etc.)
  - Multiple sources summarizing OE and the impact of cold work
- SCC of CASS from limited sources
- PWSCC of austenitic nickel-based alloys covered in multiple studies and summarized in a review by Was
- LTCP summarized in MRP-293
- SCC of A-286 from thermal shield bolts and related testing
- PWSCC of Alloys X-750 and 718 summarized in multiple reports and in experience from CRGT support pins and fuel assembly components, respectively
- SCC of martensitic PH materials from PWR valve part and BWR sources

SCC Screening Criteria

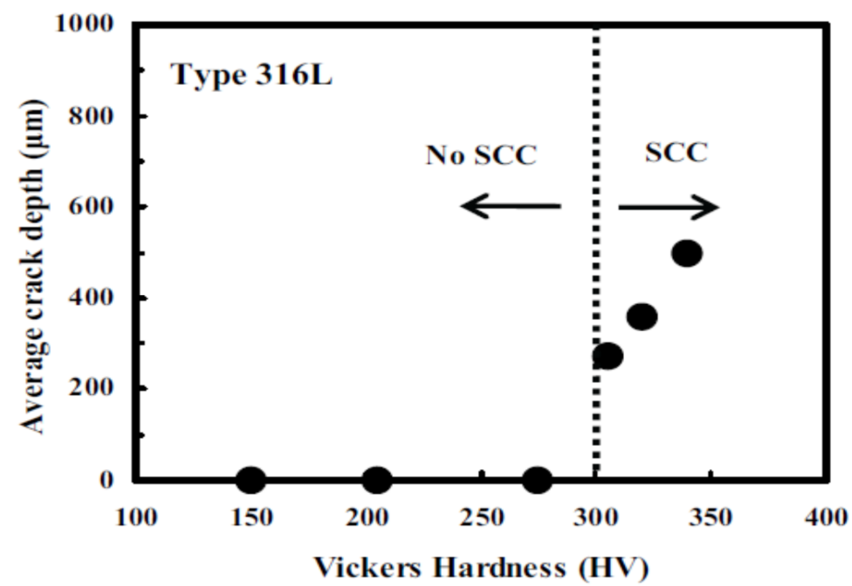
- Must consider all three factors influencing SCC in creating screening criteria:
  - Material: alloy, cold work, ferrite content, weld or base metal
  - Environment: assume coolant chemistry management and H_2 addition
  - Tensile stress: key screening criterion for all cases
- Austenitic SS
  - Environment: oxygen and chloride managed by water chemistry program
  - Material: CW has strong effect, for screening purposes limited to 20%
    - Based on tests showing susceptibility above 270-310 HV
    - Per MRP-236, CW a factor in >85% of PWR SCC events, including all events under free coolant flow conditions
  - Tensile stress: stressed at or near the room temperature yield strength
  - Welds: presence of residual stresses increase susceptibility, while the presence of ferrite decreases susceptibility

Effect of Cold Work on SCC

- Presence of cold work has a strong impact on SCC susceptibility
- Cracking in thick components with surface cold work is expected to arrest in the bulk material
- 20% cold work corresponds to approximately 98 ksi RT YS
- ASME B&PV code and NRC limit yield strength to 90 ksi RT YS (code case N-60-6 and Regulatory Guide 1.84)



MRP-175, Rev. 1, Figure A-9

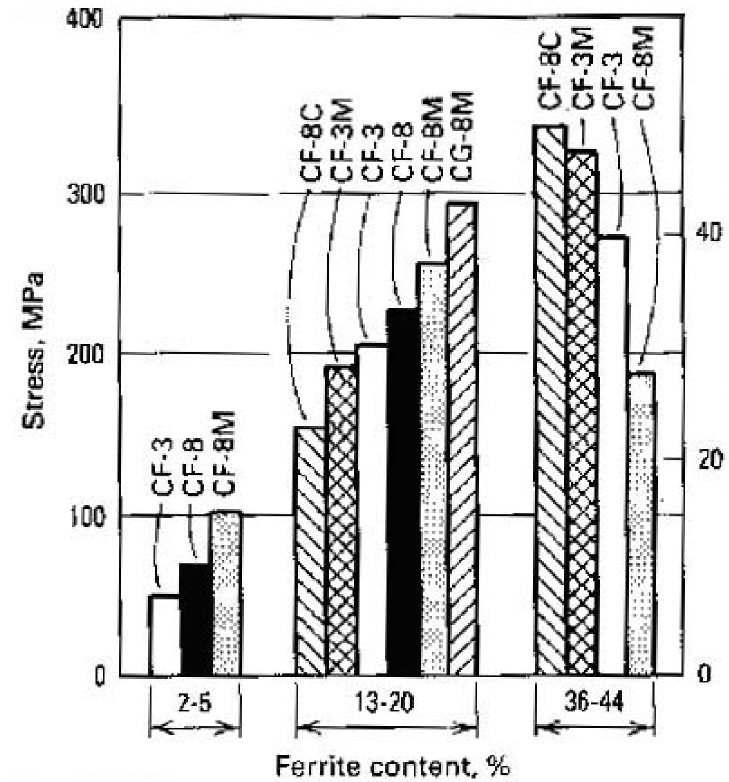


MRP-175, Rev. 1, Figure A-10

ELECTRIC POWER RESEARCH INSTITUTE

SCC Screening Criteria (cont.)

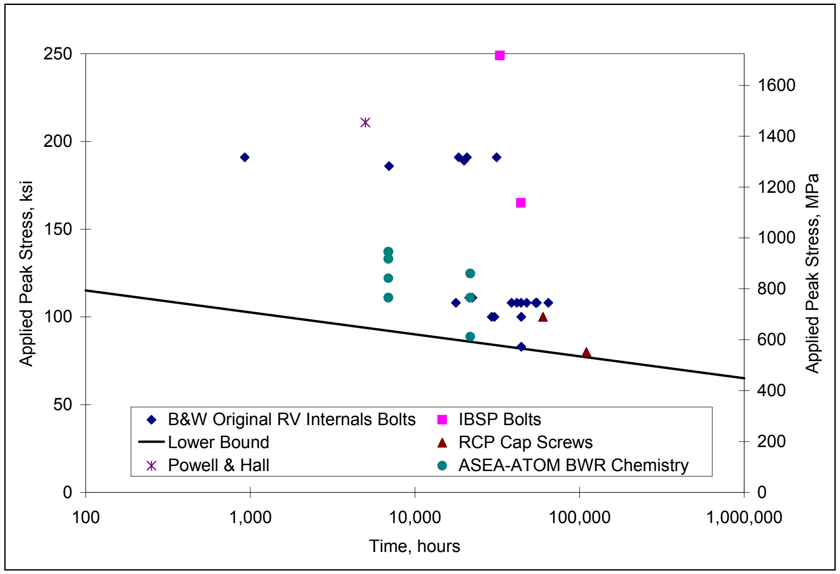
- CASS
  - Limited test results for PWR conditions available
  - Results show increased resistance to SCC with increasing ferrite
  - Stress: similar to wrought material >35 ksi
  - Material: beneficial to have ferrite >5%
  - Impact of TE addressed in TE section
- Austenitic nickel-based alloys
  - Not common in the internals
  - Significant experience with PWSCC
  - Occurs at stress near RT yield strength
  - Conservatively just screened on stress
- Martensitic and martensitic PH SS
  - Generally quite resistant
  - Poor heat treatment can be a problem
  - Screened on stress; above 70% of room temperature yield strength
  - Impact of TE addressed in TE section



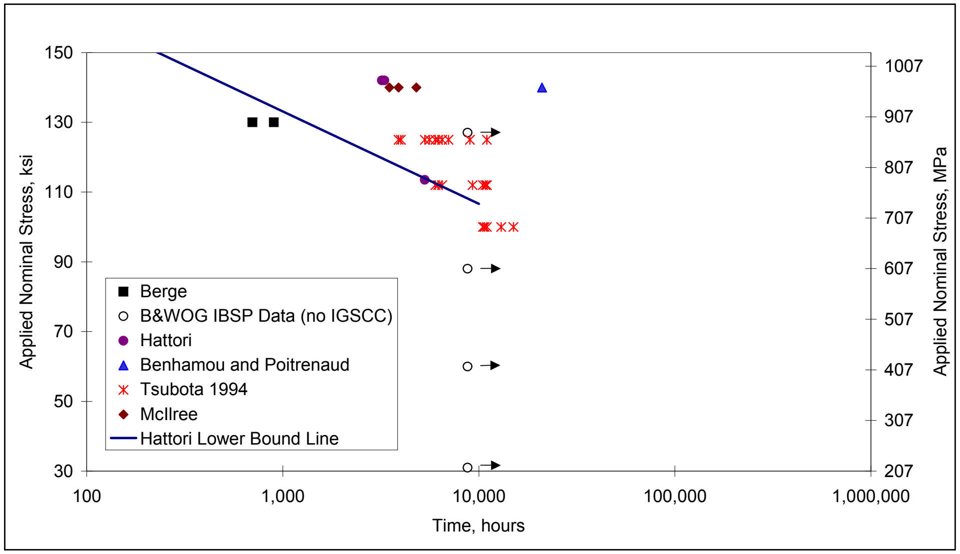
MRP-175, Rev. 0, Figure A-11: Stress required for SCC in several CASS alloys with varying amounts of ferrite

SCC Screening Criteria (cont.)

- Austenitic PH stainless steel (A-286)
  - Operating experience from B&W thermal shield bolts
  - Sensitive to fabrication – cold work, hot heading, surface hardening
  - Screened based on stress – tensile loading > 70% of RT YS
- Austenitic PH nickel based alloys (X-750 and 718)
  - Operating experience from Westinghouse CRGT support pins
  - Alloy 718 experience generally positive – surface defects can initiate cracking
  - Screened based on stress – approximately the yield strength of each material



MRP-175, Rev. 0, Figure A-9: Failure Trend Line for Alloy A-286 SCC



MRP-175, Rev. 0, Figure A-10: Failure Trend Line for Alloy X-750 HTH SCC

SCC Screening Criteria, Rev. 1

No changes from Revision 0

MRP-175, Rev. 0 and 1, Table 2-1

Material	Parameter	Value
Austenitic Stainless Steels	Stress and Material	≥ 30 ksi (207 MPa) and Cold-work ≥ 20% or Welded Locations
Austenitic Stainless Steel Welds	Stress and Material	≥ 30 ksi (207 MPa) and Ferrite < 5%
Martensitic Stainless Steels	Stress	≥ 88 ksi (607 MPa)
Martensitic PH Stainless Steels	Stress	≥ 88 ksi (607 MPa)
Austenitic PH Stainless Steels	Stress and Material	≥ 70 ksi (483 MPa) and Surface cold-work or cold-drawn spring wire
	Hot-headed or shot-peened bolting that meet the stress criterion is to be evaluated for SCC.	
CASS	Stress and Material	≥ 35 ksi (241 MPa) and Ferrite < 5%
Austenitic Ni-base Alloys	Stress	≥ 30 ksi (207 MPa)
Austenitic Ni-base Welds	Stress	≥ 35 ksi (241 MPa)
Austenitic PH Ni-base (Alloy X-750)	Stress	≥ 100 ksi (689 MPa)
	AH and BH condition considered more susceptible than HTH condition.	
Austenitic PH Ni-base (Alloy 718)	Stress	≥ 130 ksi (896 MPa)
Co-base Alloys	Alloys not susceptible in PWR internals locations.	

IASCC (Appendix B)

- Mechanism where materials become more susceptible to SCC as neutron fluence increases
- Exact mechanism of IASCC is not well established
  - Material hardening could be a contributor
  - Radiation-induced segregation (RIS) of alloying elements could contribute
- Requires material, environment, and tensile stress
  - Internals materials near the core are affected (e.g., austenitic SS)
  - Radiation dose is the driving environmental factor
  - Normal operating level of tensile stress must be part of screening
- Relevant IASCC degradation experience to date has generally been with austenitic SS
  - Other materials were also addressed in MRP-175

IASCC TOC

IASCC Data Sources

- Includes 70 references with data or information on IASCC
- Base knowledge comes from BWR and fast reactors
- WOG/JOBB and other programs coordinated by the EPRI MRP and other organizations have accumulated more data
  - Testing programs
  - Failure analysis of degraded internals components
- Operating experience relevant to PWRs
  - Stainless steel fuel cladding
  - Baffle-former bolts
- Limited studies on materials other than austenitic SS
  - One study on martensitic and martensitic PH SS
  - No studies on austenitic PH SS
  - One study on CASS
  - No known studies on austenitic nickel-based alloys
  - Several studies or analyses of austenitic PH nickel-based Alloy X-750
  - Some experience with Alloy 718 (fuel assemblies)

IASCC of Relevant PWR Component Materials

- General material impacts
  - Cold work delays many radiation effects, but cold worked materials at high dose/high stress can still experience IASCC (e.g., baffle bolts)
  - Insufficient data available to separate materials based on type or heat
- Austenitic stainless steel
  - Most information available is for austenitic SS
  - Most highly irradiated components in PWR internals
  - RIS, dislocation density, He concentration at grain boundaries, and changes in mechanical properties all appear to contribute
  - Screening in MRP-175, Rev. 0 was based on crack initiation test results for O-rings fabricated from flux thimble tube material
    - Original Rev.0 screening criteria was selected to be more conservative than the IASCC dataset bounding curve
  - Data from multiple other initiation test sources added in Rev. 1
  - One set of results indicated that data from thimble tubes are more conservative than that from baffle bolts

IASCC of Relevant PWR Component Materials (cont.)

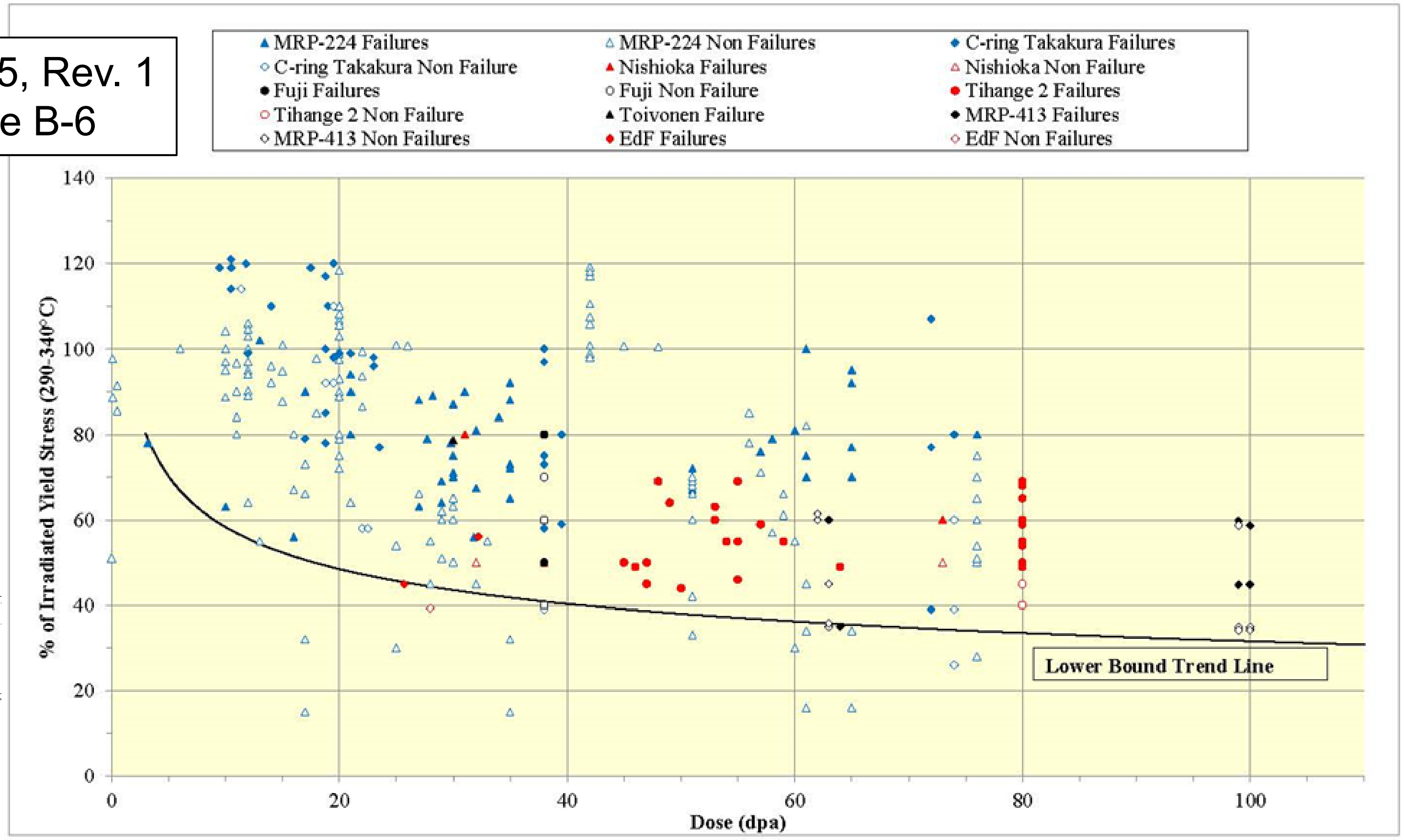
- Martensitic and martensitic PH SS
  - One available study concluded these more resistant than austenitic SS
  - May be skewed by mechanical properties, so considered the same here
- Austenitic PH SS
  - No studies available in open literature
  - Concluded to be potentially susceptible to IASCC in a manner similar to the austenitic SS
- CASS
  - One available study shows potentially higher resistance to IASCC
  - Conservatively concluded to be the same as austenitic SS
- Austenitic nickel-based alloys
  - No known studies, but concluded to be the same as austenitic SS
- Austenitic PH nickel-based alloys
  - Alloy X-750 – limited usage in irradiated areas and limited data, concluded to have similar susceptibility as austenitic SS
  - Alloy 718 – extensive use in fuel assemblies with few failures, some possible evidence of IASCC, concluded to be similar to austenitic SS

IASCC Screening Criteria

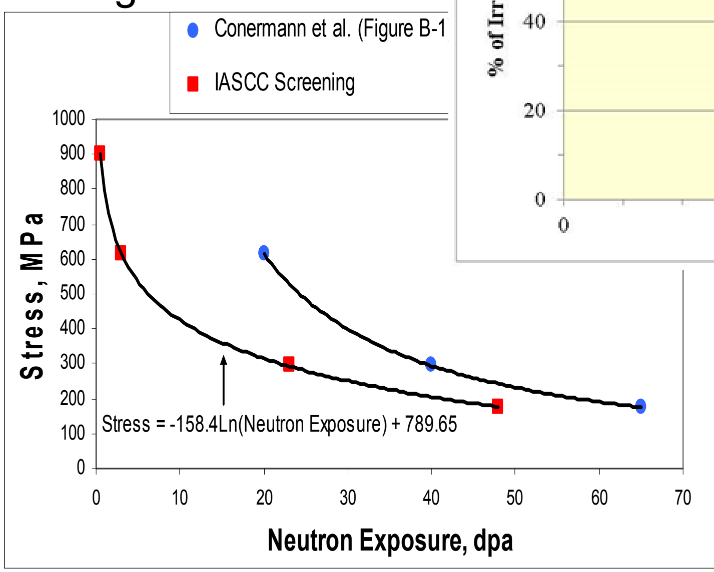
- Focused on austenitic SS but the same screening criteria apply to all alloys (based on material evaluations)
- MRP-175, Revision 0 screening for IASCC (Table 2-2):
  - Criteria where the screening stress varied with accumulated dose
  - < 3 dpa – apply SCC criteria
  - ≥ 3 dpa, stress >89 ksi (616 MPa)
  - ≥ 10 dpa, stress >62 ksi (425 MPa)
  - ≥ 20 dpa, stress >46 ksi (315 MPa)
  - ≥ 40 dpa, stress >30 ksi (205 MPa)
- MRP-175, Revision 1 screening for IASCC (Table 2-2):
  - Crack initiation data gathered and plotted, resulting in a new relationship for the screening value
  - < 3 dpa – apply SCC criteria (same)
  - ≥ 3 dpa → Equation in Table 2-2 of Rev. 1
  - Criteria for screening stress vary continuously with accumulated dose (see next slide)
    - Low dose end (< 3 dpa) Screening for > 90 ksi (620 MPa);
    - Highest dose end (~100 dpa) Screening for > 35 ksi (240 MPa)
    - NRC Suppl. Question #2

Revised IASCC Screening Basis

MRP-175, Rev. 1
Figure B-6



MRP-175, Rev. 0
Figure B-3



Includes data from multiple austenitic SS types

Similar behavior expected after radiation is accumulated

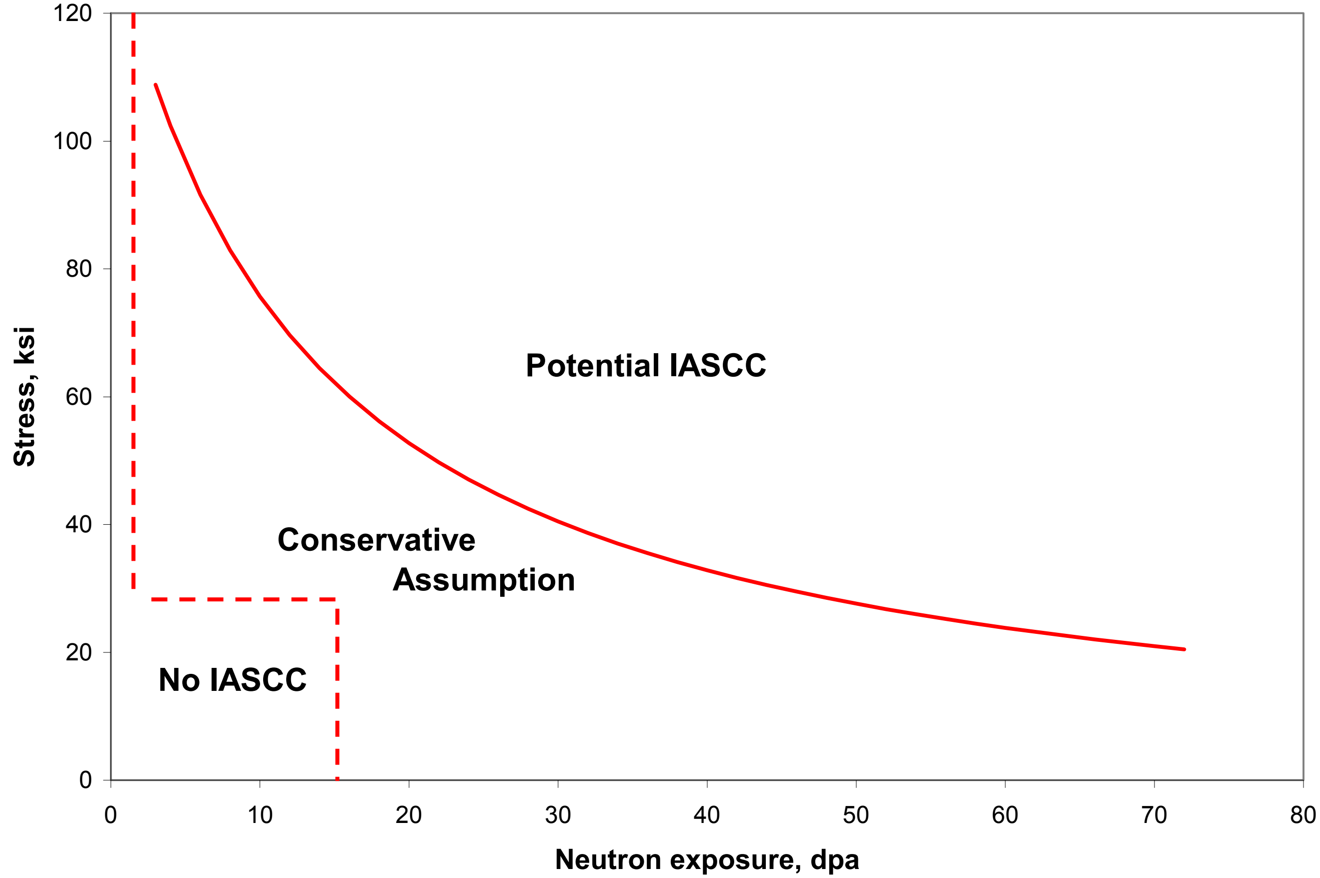
Bounding approach addresses potential small differences

Conservatism of MRP-191 IASCC Screening

- Screening criteria developed from the IASCC initiation database as shown in previous figure (Figure B-6 of MRP-175, Revision 1)
  - IASCC initiation never observed in lab testing below the threshold line
  - “Applied stress” does not differentiate between active and residual stress
- Weld residual stress uncertainty
  - Weld residual stress value not precisely known
  - Residual stress in large, constrained welds expected to be at or above the unirradiated yield strength of the material (30 ksi minimum)
- Potential uncertainties evaluated in applying screening criteria
  - Accounted for in step documented in MRP-191 FMECA by using a more conservative value (shown on the next slide)
  - Lower limit dose of 1.5 dpa used instead of 3 dpa
  - From 1.5 to 15 dpa, components were screened in above 30 ksi
  - All components above 15 dpa screened in, regardless of applied stress
  - Criteria used in original screening for MRP-191, Rev. 2 as well

Conservatism of MRP-191 IASCC Screening (cont.)

Example of conservative screening criteria used by WEC in MRP-191, Section 5.1.2



MRP-191, Rev. 1, Figure 5-1 (EPRI TR 1013234)

BREAK

Wear (Appendix C)

- Defined as the localized loss of material due to relative motion between two surfaces in contact
- Governed by the types of materials in contact, the type of relative motion, the normal force, and the environment
- Wear relationships for a particular case can be developed but prediction and choosing a lead item are difficult
- Operating experience plays a key role in screening

Wear TOC

Wear Sources

- General references providing background information on wear, including a review of the wear types
- Operating experience references for PWR events or ongoing wear issues
  - Westinghouse: flux thimble tubes, CRGT guide cards, clevis inserts
  - B&W: vent valve jacking screw lock cup, surveillance holder tubes (redesigned)
  - CE: hold-down ring (redesigned)
- Connection to the ISR/IC section for components that require preload, since wear could occur with loosening

Wear Screening Criteria

- MRP-175, Revision 0 screening for wear (Table 2-8):
  a. Locations where relative motion may occur between component items (such as control rod guide tubes)
  b. Locations where clamping force is required (such as the mating ledge between the internals and the RV)
  c. All bolted or spring locations subject to irradiation-enhanced stress relaxation/creep
- Essentially any location that could conceivably experience wear – determined by expert review
- Same reasoning continues to apply for MRP-175, Revision 1, so the expert panel retained the same screening values
  - Screening in Table 2-8
  - References were updated to include new OE since 2005

Fatigue (Appendix D)

- Fatigue section addresses the potential for both high-cycle fatigue and low-cycle fatigue
  - HCF characterized by the expected stress cycles (endurance limit)
  - LCF characterized by the cumulative usage factor (CUF)
- EAF was included in MRP-175, Rev. 0 by reducing the allowable 40-year CUF to 0.1 (instead of $\frac{40\ years}{60\ years} = 0.67$)
- Components subject to irradiation exposure that also require a preload for function could loosen and experience fatigue
- MRP-175 assumes:
  - Water chemistry conditions are maintained
  - Fatigue can occur in all materials subjected to cycling (mech. or therm.)
  - Fatigue can occur at all relevant PWR internals temperatures
- Goal is to not exceed a CUF_{EN} of 1 at the end of life

Fatigue TOC

Fatigue Data Sources

- General fatigue background information references
- ASME Code fatigue design criteria
  - Governs the design of later reactors
  - Relevant to the rest
- Recently published summary of the impact of environmental effects in LWR environments in NUREG-6909, Rev. 1
  - Compilation of relevant environmental effect fatigue data
  - Data available are still limited when considering the broad range of potential environments and loading conditions in the internals
  - Worst case factor calculated for internals is ~14
- References for the impact of neutron irradiation on fatigue
  - Mixed results
  - Inconclusive given the current limited data

Fatigue Screening Criteria

- MRP-175, Revision 0 screening for fatigue (Table 2-3):
  - Any component with CUF at 40 years >0.1
  - Locations requiring a preload that experience ISR/IC (bolts, springs)
  - Components with fatigue life originally qualified through testing
- Changes for MRP-175, Revision 1 only impacted the CUF
  - Screening criteria in Table 2-3
  - $CUF_{EN} \leq 1$ at end of life
  - CUF ≤ 0.036 at 40 years for 80 years planned life
  - CUF ≤ 0.028 at 40 years for 100 years planned life
  - Locations requiring a preload that experience ISR/IC (bolts, springs)
  - Components with fatigue life originally qualified through testing

Fatigue Screening Criteria (cont.)

- Revised CUF reduced by additional time and an EAF factor
  - Factor of 14x reduction based on NUREG/CR-6909
  - Worst case parameters applied (temperature, strain rate, environment)
- CUF with EAF is lower than that used in Revision 0
  - Many more components screen in with revised EAF
  - Components most affected by fatigue are still the lead items
    - These were previously identified as the Primary components
  - MRP-227 lead component approach covers this change → still managing the correct leading components
    - Change does <u>not</u> invalidate MRP-227-A, Rev. 0, or Rev. 1

Thermal Embrittlement (Appendix E)

- Results in several material property changes:
  - Hardening
  - Increase in yield and tensile strength
  - Reduced toughness and ductility
- Occurs for a limited number of PWR internals materials
  - Cast austenitic stainless steel (CASS)
  - Austenitic stainless steel welds
  - Martensitic stainless steels
  - Martensitic precipitation-hardenable stainless steels
- Does not affect wrought alloys
- TE needs to be considered in conjunction with cracking mechanisms due to reduced critical crack sizes

Thermal Embrittlement TOC

- E.1 – General Description of Thermal Aging Embrittlement
  - E.1.1 – CASS
  - E.1.2 – Austenitic Stainless Steel Welds
  - E.1.3 – Martensitic Stainless Steel
  - E.1.4 – Martensitic Precipitation-Hardenable Stainless Steel
- E.2 – TE Summary and Discussion
- E.3 – TE Threshold and Screening Criteria
- E.4 – TE References

Thermal Embrittlement Data Sources

- Includes 56 references related to TE
- Relies on MRP-80 as a key reference summarizing much of the data
- CASS materials
  - National lab data from Chopra and others
  - NUREG documents
  - Safety evaluation on BWRVIP-234
  - Industry CASS materials statistical assessment (PWROG-15032-NP)
- Austenitic stainless steel welds
  - MRP-80 and MRP-276
  - Multiple other references
- Martensitic stainless steel
  - Data from various sources, mainly for Type 410 SS
  - Considered to be equivalent to the other grades
- Martensitic PH stainless steel
  - Data available for 17-4 PH and applied to other grades (15-5)
  - Multiple sources available, including LWR-relevant studies

CASS

- Received significant attention under MRP-227-A
  - A/LAI 7: Plant-specific evaluation of CASS components
  - Potential synergistic effects of TE and IE on CASS
  - Screening criteria for TE and IE
- MRP-175, Rev. 1 references the documents resolving this
  - PWROG-15032-NP and NRC staff assessment ML16250A001
    - Detailed search of CASS CMTR records
    - Statistical evaluation of ferrite contents showing generally < 20%
    - TE saturation fracture toughness of CASS > 255 kJ/m^2
  - SE on BWRVIP-234
    - Screening criterion for fracture toughness at 200 kJ/m^2
    - Internals do not require same toughness as pressure boundary
    - Screening criterion for potential combination of IE and TE at 1 dpa for CASS with < 0.5% Mo

Thermal Embrittlement Screening Criteria

- Temperature threshold for all susceptible materials at 260°C (500°F) – affects all PWR internals
- CASS and austenitic SS welds depend on the Mo content and the ferrite content
  - These were calculated for many components in responding to A/LAI 7
  - PWROG-15032-NP addressed CASS TE using a statistical approach
- Martensitic SS and martensitic PH SS are always screened in for TE
- Only change from Revision 0 was to increase ferrite screening level for centrifugally cast CASS with ≤ 0.50% Mo from 20% ferrite to 25% ferrite

Thermal Embrittlement Screening Criteria Rev. 1

MRP-175, Revision 1, Table 2-4

Material	Criteria	
	Parameter	Value
Austenitic SS Austenitic PH SS Austenitic Ni-Base Alloys Austenitic PH Ni-Base Alloys Co-Base Alloys	TE is not applicable to these materials.	
CASS (Centrifugal Castings)	Molybdenum and Ferrite	≤ 0.50% and > 25%
	Molybdenum and Ferrite	> 0.50% and > 20%
CASS (Static Castings)	Molybdenum and Ferrite	≤ 0.50% and > 20%
	Molybdenum and Ferrite	> 0.50% and > 14%
Austenitic SS Welds	Molybdenum and Ferrite	≤ 0.50% and > 20%
	Molybdenum and Ferrite	> 0.50% and > 14%
	TE is not anticipated as an issue due to ASME Code procurement requirements for low levels of ferrite (5-15%) and low Mo levels.	
Martensitic SS	All component items considered susceptible to TE.	
Martensitic PH SS	All component items considered susceptible to TE.	

Irradiation Embrittlement (Appendix F)

- Results in several material property changes:
  - Hardening
  - Increase in yield and tensile strength
  - Reduced toughness and ductility
- Impacts a limited group of materials due to location
  - Austenitic stainless steel
  - Austenitic stainless steel welds
  - CASS

Irradiation Embrittlement TOC

Irradiation Embrittlement Data Sources

- Austenitic SS
  - MRP-79: fast reactor up to 80 dpa and LWR up to 18 dpa
  - National lab fast reactor – NUREG/CR-6826 and NUREG/CR-6960
  - MRP-160 and other references
- Austenitic SS welds and CASS
  - MRP-79: fast reactor up to 15 dpa, BWR and PWR
  - MRP-276, NUREG/CR-7185

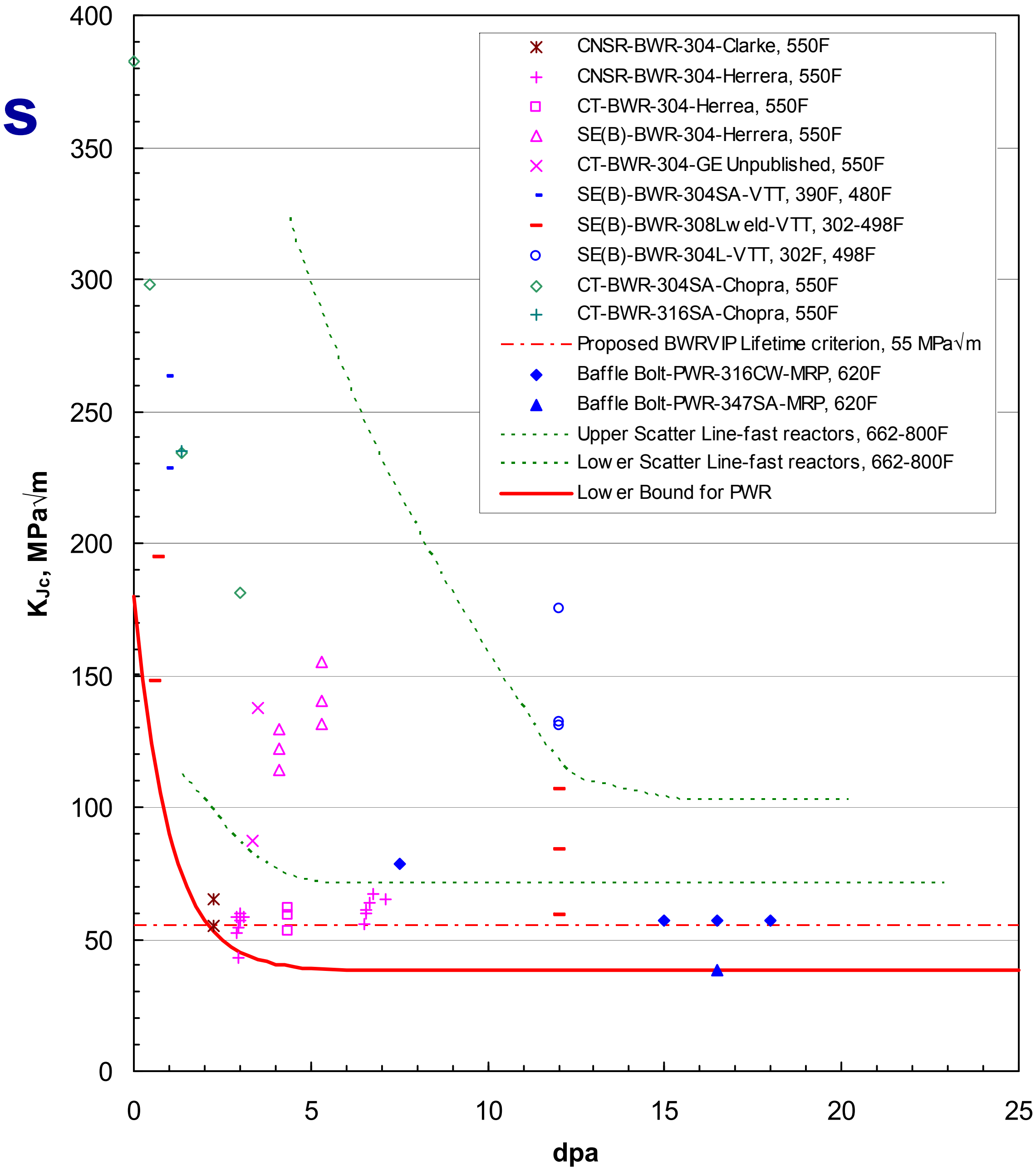
Irradiation Embrittlement Screening Criteria

- Austenitic stainless steel
  - Types 304, 316, 347, and 348 show similar enough behavior to be treated together for IE
  - Fracture toughness decreases rapidly between 0 and 10 dpa
    - Data fitting shows plateau as early as 5 dpa (around 40 MPa$\sqrt{m}$)
  - Screening level was set at 30% of this level for conservatism: 1.5 dpa
- CASS and austenitic stainless steel welds
  - Show greater variability than wrought materials
  - Potentially susceptible to TE
  - Screening level set slightly lower at 1 dpa (consistent with BWRVIP-234-A)
- No changes from MRP-175, Revision 0 to Revision 1
- Screening levels additionally supported by discussion in TE section

Impact of IE on Fracture Toughness

MRP-175, Rev. 0, Figure F-2

Elevated Temperature Fracture Toughness K_{Jc} of Austenitic Stainless Steels and Welds Irradiated in BWRs or PWRs as a Function of dpa



Irradiation Embrittlement Screening Criteria Rev. 1

MRP-175, Revision 0 and 1, Table 2-5

Material	Criteria	
	Parameter	Value
Austenitic PH SS Austenitic Ni-Base Alloys Austenitic PH Ni-Base Alloys Martensitic SS Martensitic PH SS Co-Base Alloys	These materials are generally used in relatively low fluence locations; therefore, IE is not an applicable age-related degradation mechanism for component items fabricated with these alloys.	
Austenitic SS	Dose	$\geq 1 \times 10^{21}$ n/cm^2 (E > 1.0 MeV) [≥ 1.5 dpa]
Austenitic SS Welds CASS	Dose	$\geq 6.7 \times 10^{20}$ n/cm^2 (E > 1.0 MeV) [≥ 1 dpa]
	Lower screening value accounts for the large variability in initial fracture toughness and possible combined effects due to simultaneous occurrence of IE and TE	

Void Swelling (Appendix G)

- Occurs at relatively high radiation dose levels
  - Generally only applicable to austenitic stainless steels
  - Only relevant in components near the core
- Dependent on dose, dose rate, temperature, and material
- Little data exist for LWR conditions and materials
  - Most data are from fast flux reactors due to the higher dose rate
  - Few components have been extracted and tested
- Data for LWR conditions to date have shown very low swelling
  - Temperature in most components is much lower than test reactors
  - Swelling expected to be localized to regions of highest temperature (gamma heating) with surrounding cool material not swelling as much
- Screening value based on potential for significant mechanical impacts and possible embrittlement

Void Swelling Appendix G TOC

- G.1 – General Description of Void Swelling (VS)
- G.2 – Descriptions of Austenitic Stainless Steels VS Models
  - G.2.1 – Published Equations Modeling VS
  - G.2.2 – Stress-Free Model for Types 304 and 316 Stainless Steel
  - G.2.3 – Cluster Dynamics Model
- G.3 – VS of Russian Stainless Steels in Other Fast Reactors
  - G.3.1 – VS Reported in BN-350, BOR-60, and BR-10
  - G.3.2 – BN-350 VS Database Analyzed by Yilmaz et al.
- G.4 – JOBB VS Data
  - G.4.1 – JOBB BOR-60 Density Measurement
  - G.4.2 – JOBB TEM Measurement of EBR-II Specimens
- G.5 – Density Measurement Results from the LWRS Program
- G.6 – VS in Removed PWR Components
- G.7 – Factors Affecting VS
- G.8 – VS Threshold and Screening Criteria
- G.9 – References

Void Swelling Data Sources

- Total of 111 references with VS data or discussing VS
- Sources of VS data:
  - Fast Reactors
    - FFTF, EBR-II, ORR, HFIR, BN-350, BOR-60, BR-10
    - Represents majority of available data—higher dose rates, more swelling observed, generally operated for research
    - Many operate above 370°C (698°F)– not PWR relevant
    - Russian reactors have provided data at 280-365°C (536-689°F)
    - Data focused on austenitic SS: Types 304, 316, 321
    - Available empirical equations based on these data
  - LWRs
    - Limited data set available—removed baffle-former bolts and flux thimble tubes
    - Maximum PWR swelling observed was 0.24% for a baffle bolt

LWR Empirical Data for Screening Criteria

- Baffle-former bolts
  - Temperature range: 282-363°C (540-685°F)
  - Dose range: 7 to 26.8 dpa
  - Volumetric swelling range: none to 0.24%
  - Key entries include no swelling at 320°C and 26.8 dpa
  - Maximum swelling below 320°C was 0.029% after 7-8 dpa at 302-341°C (576-646°F)
- Flux thimble tubes
  - Temperature range: 290-320°C (554-608°F)
  - Dose range: 1 to 73 dpa
  - Volumetric swelling range: very small
  - ~1 nm Helium bubbles detected above 1 dpa and number density increased with increasing dose

Void Swelling Screening Criteria

- MRP-175, Revision 0 screening for void swelling (Table 2-6):
  - Applicable to austenitic SS and welds since they receive enough dose
  - Temperature ≥608°F (320°C) and Dose ≥ 20 dpa (≥ 1.3×10^{22} n/cm^2 (E > 1.0 MeV))
- No changes made for MRP-175, Revision 1 (Table 2-6)
  - Data still show that little swelling will occur below these values
  - Swelling is low in LWR-relevant swelling test data gathered since 2005
  - Recent data and reviews show that the saturation swelling rate should be less than 0.1%/dpa for LWRs (compared to 1%/dpa for fast flux)
  - Simulations using a cluster dynamics approach predicted less than 1.5% swelling at the 20 dpa screening values – acceptable from mechanical and embrittlement standpoint
  - Screening addresses the possibility of embrittlement which starts around 5% swelling and reduces significantly at 10% swelling
  - Industry conservatively retained MRP-175 Revision 0 screening values

Stress Relaxation and Creep (Appendix H)

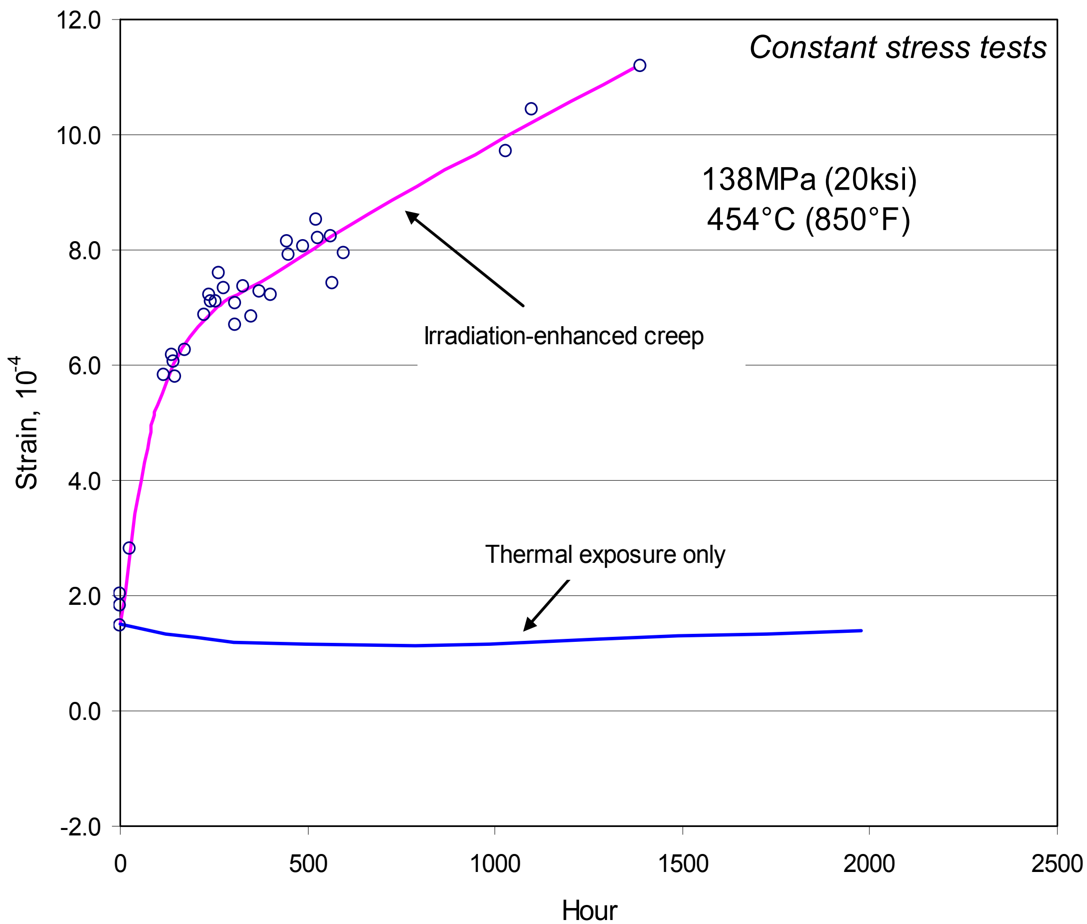
- Manifestations of the same process:
  - Stress relaxation – decrease in stress under a constant deflection
  - Creep – inelastic deformation under a constant load
- Thermal stress relaxation and creep were considered, but the amount is small at PWR temperatures, below 0.5 T_m
  - Components with critical hold-down functions could experience these at long lifetimes (e.g., license renewal)
- ISR/IC can occur in any internals component alloy exposed to adequate radiation dose
  - Data are not available to differentiate between different alloys
  - ISR/IC is only an issue for components that rely on preload or hold-down force for functionality
  - ISR can be rapid, achieving 50% lower stress by 0.2 dpa

Stress Relaxation and Creep TOC

Stress Relaxation and Creep Data

- Manjoine's thermal stress relaxation tests from 1970s
  - Show limited transient stress relaxation for Type 304 SS
- MRP-50—compilation of SR and creep through ~2001
- EBR-II samples evaluated for thermal creep and IC
  - Thermal creep level was essentially zero
  - At the same applied stress, irradiation creep occurred (next slide)
- Pressurized tubes in fast reactors – BOR-60, OSIRIS, EBR-II
  - Have provided data for creep relationships
- Data for thermal reactor spectrums shows creep below 1 dpa
- Sample types: pressurized tubes, uniaxial tensile, bent beams, helical springs, etc.

EBR-II Thermal and Irradiation Creep Comparison



MRP-175, Revision 0, Figure H-6: Comparison of Thermal Creep and Irradiation-Enhanced Creep of a 20% Cold-Worked Type 316 Stainless Steel Irradiated in EBR-II

Irradiation Stress Relaxation of Welds

- Irradiation enhanced stress relaxation at low fluence in test welds has been demonstrated by Obata, et.al (ASTM STP 1475)
- Testing performed on Type 304 SS specimens
  - Irradiated in the Japan Material Test Reactor
  - Neutron irradiation to ~3 dpa (2×10^{21} m/cm^2 (E > 1 MeV))
  - Residual stress measured by neutron diffraction
- Results showed significant stress relaxation
  - Relaxed both parallel and perpendicular to the weld axis
  - Started with an initial steep relaxation – transient stage
  - Slower than mechanical stress relaxation due to three dimensional aspect but still occurs

Screening Criteria Selection

- Impact of aging effect depends on application
  - ISR/IC can reduce applied stresses, reducing cracking degradation
  - ISR may reduce desired preload in threaded fasteners and springs
  - IC may distort components and interfere with fluid flow or heat transfer
- Irradiation Stress relaxation:
  - Loss of preload in a bolt can result in static or cyclic loads that may cause cracking
  - Screening criteria for ISR should include potential for loss of preload or spring force
- Uncertainty in transient behavior at low fluences dominates ISR prediction
  - Limits the capability to determine a reliable threshold criteria for PWR components
  - Potential creep of 50% by 0.2 dpa considered reasonable functionality threshold

Stress Relaxation and Creep Screening Criteria

- MRP-175, Rev. 0 screening for Stress Relaxation and Creep (Table 2-7):
  - Thermal: bolts or springs that require preload to function
  - ISR/IC: Dose ≥ 0.2 dpa (≥1.3×10^{20} n/cm^2 (E > 1.0 MeV)) and a bolted or spring location
- No significant changes made for MRP-175, Revision 1 (Table 2-7)
  - Impact of ISR/IC on components requiring preload to function is still applicable, dose level still applicable
    - Literature review provided some additional ISR/IC data
    - New data did not contradict prior conclusions about the radiation level required to experience ISR/IC – relatively low fluence levels
  - Thermal stress relaxation clarified to apply to components providing spring forces in the core hold-down load line (hold-down spring)

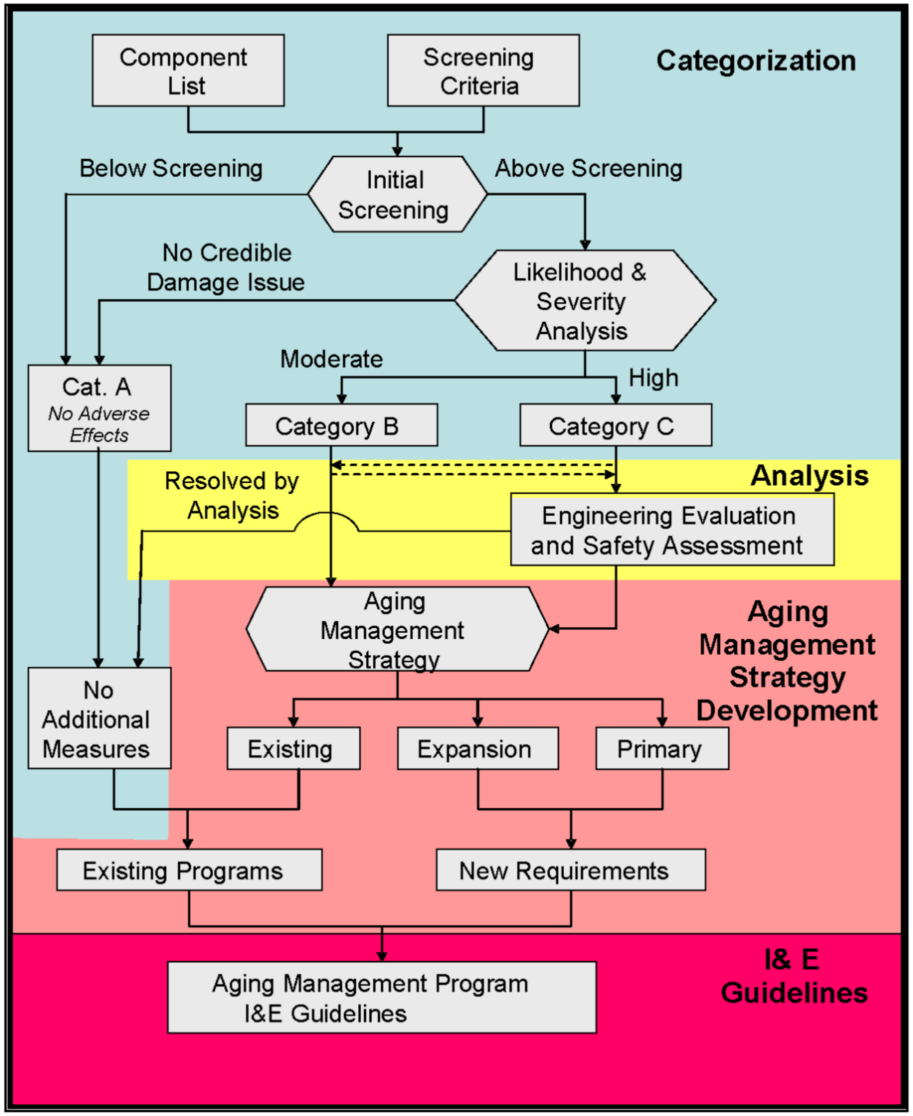
Summary of MRP-175, Revision Screening Criteria

Mechanism	MRP-175, Revision 1
Wear	No Change
Stress corrosion cracking (SCC)	No Change
Irradiation embrittlement (IE)	No Change
Thermal embrittlement (TE)	Centrifugally-cast ferrite screening increased from 20% to 25% for low Mo No other changes
Irradiation-enhanced stress relaxation/creep (ISR/IC)	No Change
Void swelling (VS)	No Change
Irradiation-assisted SCC (IASCC)	Changed based on testing developments: MRP-175, Rev. 0: < 3 dpa – no IASCC > 3 dpa and stress > stress bins (89 to 30 ksi) MRP-175, Rev. 1: < 3 dpa – no IASCC > 3 dpa and stress vs. dose relationship in Table 2-2 (90 to 35 ksi)
Fatigue	Reduced for environmentally-assisted fatigue: MRP-175, Rev. 0: CUF $\leq$ 0.1 at 40 years MRP-175, Rev. 1: CUF $\leq$ 0.036 at 40 years

Application of MRP-175

Application of the Report in Developing MRP-227

- Results of screening provide a basis for further categorization
- Components are designated as **Category A** and **Non-Category A** based on screening
- Failure modes, effects, and criticality analysis (FMECA) performed on Non-Category A components
  - Results in categorization as A, B, and C components
- MRP-227-A Roadmap:
  - Initial Screening per MRP-175
  - Likelihood and Severity Analysis
  - MRP-191 for Westinghouse and CE
  - MRP-189 for B&W



Required Screening Inputs

Parameter	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Material Type	☑		☑		☑	☑		
Operating Stress	☑	☑						
Operating Temperature							☑	
Fluence		☑			☑	☑	☑	☑
CUF				☑				
Fatigue Factor				☑				
Wear Potential			☑					

Screening Examples

WARNING: Data here is for screening demonstration purposes only. It provides reasonable values for the example components, but does not necessarily represent actual screening output.

Assembly/ Component Name	Material Category	Material Type/Grade	Effective Stress ≥ Threshold	Structural Weld	Wear Potential	CUF > 0.036	Preload Required?	Max Neutron Dose @ 80yrs (dpa)
Baffle, former bolts	Austenitic SS	316 SS	Yes	No	No	Yes	Yes	>75
BMI column cruciforms	CASS	CF8	No	No	Yes	Yes	No	75
SCS housing	Austenitic SS	304 SS	No	No	No	No	No	< 0.15
UCP	Austenitic SS	304 SS	Yes	No	No	Yes	No	15

Assembly/Component Name	Material Type/Grade	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Baffle, former bolts	316 SS			IASCC	Wear (I)	Fatigue		IE	VS	ISR/IC
BMI column cruciforms	CF8			IASCC	Wear	Fatigue	TE	IE	VS	
SCS housing	304 SS	X								
UCP	304 SS			IASCC		Fatigue		IE		

Conclusions

- MRP-175, Revision 0 was developed in 2005
  - Summarized current knowledge at that time
  - Provided screening criteria for the reactor internals inspection & evaluation guidelines (MRP-227, Rev. 0/1 and MRP-227-A)
- Additional testing and operating experience data have been gathered in the decade since
- MRP-175, Revision 1 was developed in 2017
  - Updated the degradation mechanism sections with recent references
  - Adjusted the screening criteria to not be limited to 60 years
  - Validated the updates through review by a panel of experts
  - Only changes required were for IASCC and fatigue
  - Retained the original screening criteria for all other mechanisms
- Screening criteria have changed little since 2005
  - Reasonable and appropriate technical basis for aging management
  - Shows that MRP-175, Revision 0 was on-target
- Foundation of the MRP-227 I&E guidelines is validated
  - Technical aspects of first license renewal do not need to be re-addressed

Next Steps

- Revision 1 screening criteria will be used in developing MRP-189, Revision 3, and MRP-191, Revision 2 for SLR
  - Degradation mechanisms have not changed; no new mechanisms
  - Screening criteria have not changed significantly
  - Accumulated fluence, time, and fatigue usage increasing with age
  - Changes are incremental and the overall MRP-189/-191 changes will also be incremental; no “cliffs” identified based on aging effects
- Expected scale of changes for SLR
  - Time of operation only increases by 33%
  - Fatigue usage will increase by the same or less
  - Fluence increase will move screening boundaries by a few cm at most
  - Components near the boundary may have additional mechanisms and this will be addressed through expert panel FMECA review again
- MRP-227, Rev. 2 for SLR will be developed on this basis

Q&A and Open Discussion

Future Technical Exchanges with NRC

- Provide technical progress with NRC on:
  - FMECA ranking results -189/191 – late-2018
  - Engineering analysis results -229/230 – mid-2019
  - Aging Management Suggestions -231/232 – late-2019
  - Final MRP-227-Rev.2 I&E Guideline Results – end of 2020

- Intent is to keep NRC technical staff informed on industry efforts to generically address GALL-SLR requirements

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