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Reactor Vessel Internals Embrittlement

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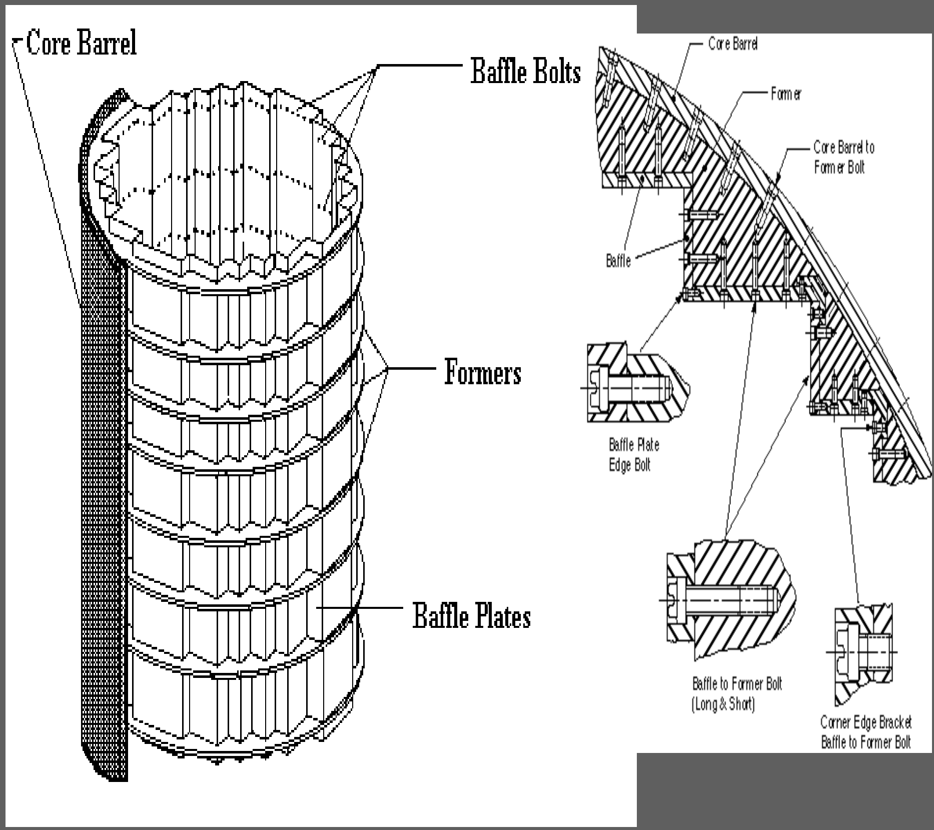
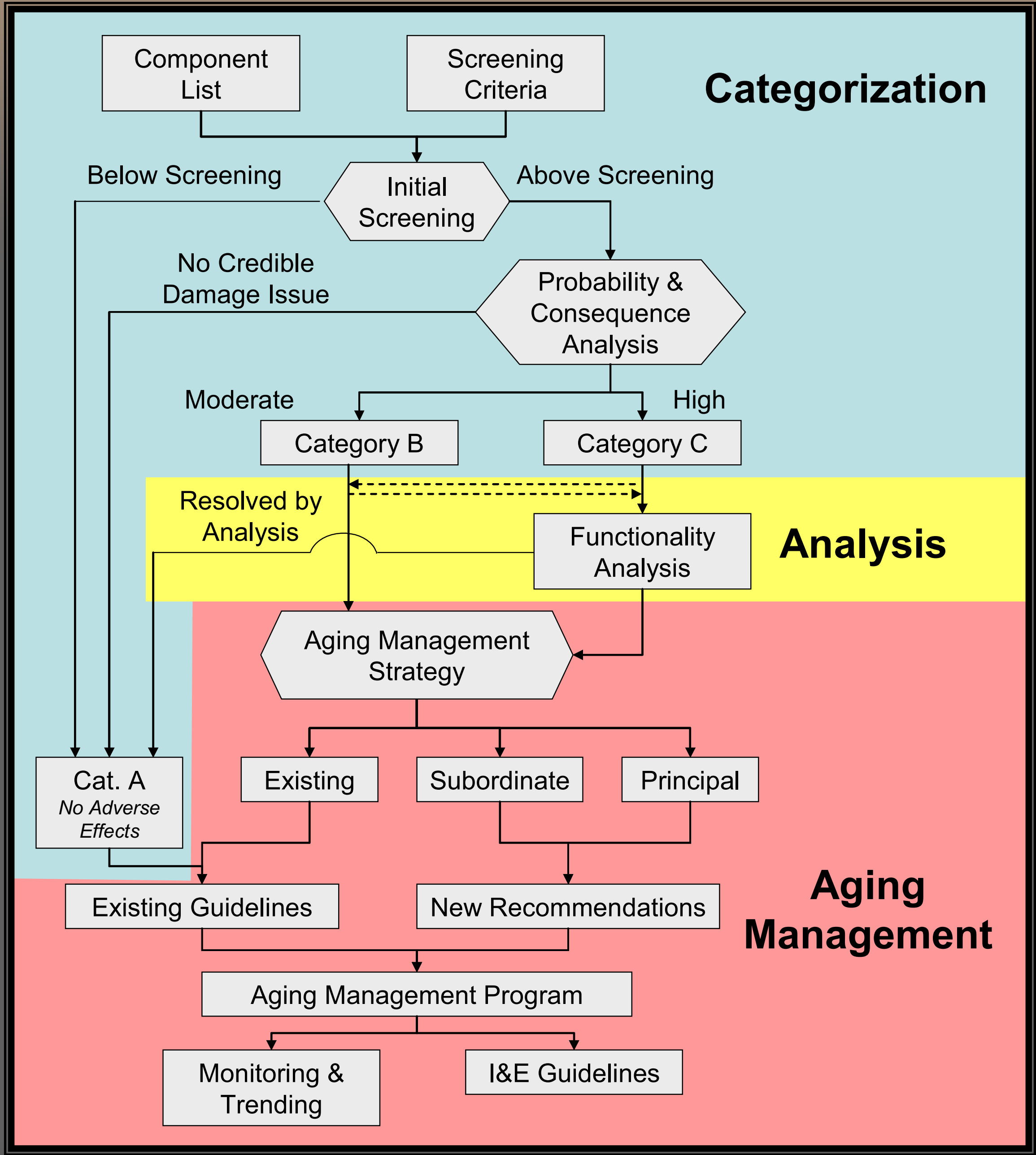
Reactor Internals Overview

Mechanisms:

- Cracking due to Irradiation Assisted Stress Corrosion (IASCC) and Stress Corrosion (SCC)
- Reduction of Fracture Toughness due to Irradiation Embrittlement (IE) and Thermal Embrittlement (TE)
- Dimensional Changes due to Void Swelling (VS)
- Loss of Mechanical Closure Integrity due to Stress Relaxation (SR)
- Synergistic Effects of These Mechanisms




Categorization
Component List
Screening Criteria
Initial Screening
Below Screening
Above Screening
No Credible Damage Issue
Probability & Consequence Analysis
Moderate
High
Category B
Category C
Analysis
Resolved by Analysis
Functionality Analysis
Aging Management Strategy
Cat. A
No Adverse Effects
Existing
Subordinate
Principal
Existing Guidelines
New Recommendations
Aging Management
Aging Management Program
Monitoring & Trending
I&E Guidelines

Approach for Evaluating Functionality Analysis Results for I&E Guidelines

- **What?**
  - Damage mechanisms of concern?
  - Metrics used to characterize a damage mechanism?
  - Observable effects/consequences on functionality?
- **Where?**
  - Location of degradation?
- **When?**
  - Estimate the likelihood and timing of future damage?
- **How?**
  - Inspection, monitoring or trending technique

Task is to utilize representative plant results and apply to entire fleet

The Cracking Mechanisms

- SCC
- IASCC
- Fatigue

Produce observable cracks
Most probable in regions of stress concentration

Expect to manage through an integrated <u>inspection</u> program.

SCC Waterfalls

- **Austenitic stainless steel**
  - No experience with SCC in 300 series stainless steel under normal primary water conditions
    - *No model to evaluate or rank potential for SCC*
  - Large structural welds identified due to large potential residual stresses
- **X-750**
  - Programs for guide tube support pins in place
- **Cast austenitic stainless steel**
  - Verify that specifications meet minimum ferrite requirements

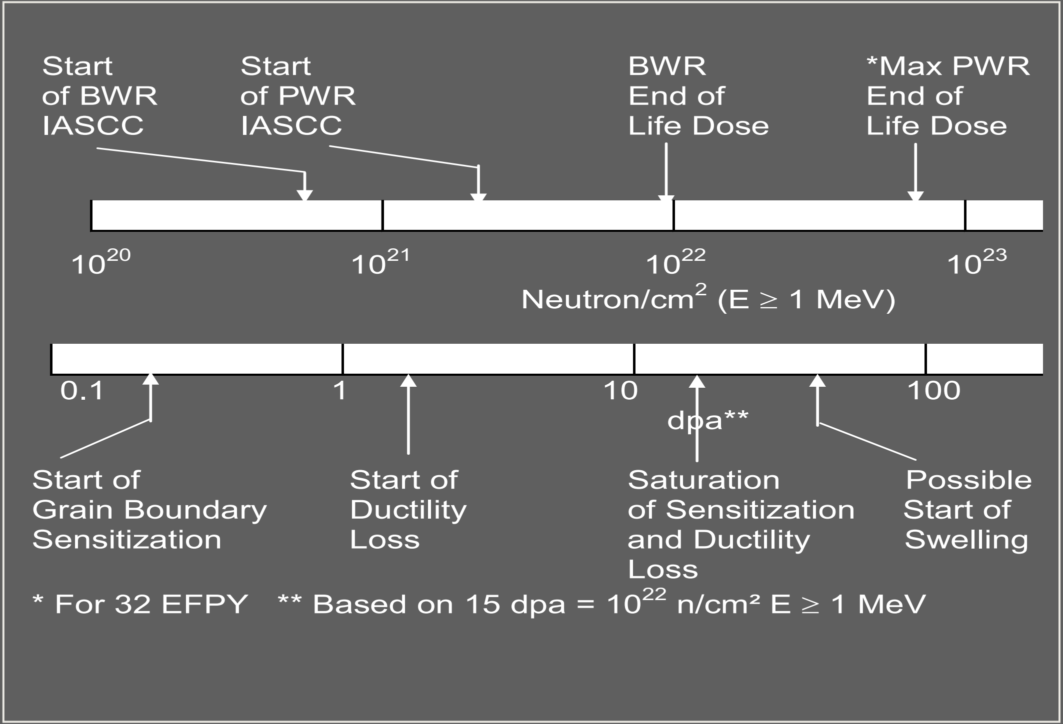
IASCC Waterfalls

- **Stainless steel solts (316 SS)**
  - FEA intended to provide basis for ranking of time to failure
  - Limited number of CE plants with bolted baffles
- **Stainless steel slate (304 SS)**
  - CE shroud welds included in plate waterfall
  - Will identify locations with IASCC susceptibility from FEA
- *Eliminated components associated with Westinghouse lower core plate on basis of completed analysis*

Parameters Influencing IASCC

Fluence

- IASCC in PWRs occurs above a threshold fluence of ~ 2×10^{21} n/cm^2, E > 1 MeV
- This fluence level is higher than in BWRs by about an order of magnitude
- The threshold fluence level does not correlate directly with the onset or saturation of radiation-induced materials changes such as grain boundary segregation or hardening



Fatigue Waterfalls

- Expect that fatigue evaluation will be required to justify extended life
  - Real vs. assumed stress history
  - Realistic stress/strain amplitudes
  - Potential environmental effects
- Two waterfall groups
  - Additional evaluation required
  - Addressed via SCC, IASCC, etc.

The Embrittlement Mechanisms

- Irradiation embrittlement
- Thermal embrittlement

Changes in material properties

- Strength (increase)
- Ductility (decrease)
- Toughness (decrease)

Expect to manage through an industry trending program

Irradiation Embrittlement Waterfalls

- Industry trend curves for strength and ductility are embedded in computer codes
  - Westinghouse lower core plate
  - Westinghouse baffle-former-barrel
  - CE core shroud
- Extrapolate to remaining components based on fluence and temperature
- Fracture toughness estimates required for components with active cracking mechanisms

Thermal Embrittlement Waterfalls

- Evaluate composition and temperature to determine susceptibility to thermal embrittlement
- Fracture toughness estimate required if there is an active cracking mechanism

Dimensional Stability Mechanisms

- Void swelling
- Irradiation induced stress relaxation/creep

Component distortion

Modify stress/strain distribution

– Affects SCC, IASCC and fatigue

Expect to manage through industry trending and inspecting for distortion

Void Swelling Waterfalls

- FEA analysis provide ranking based on swelling model in computer codes
  - Westinghouse baffle-former-barrel
  - Westinghouse lower core plate
  - CE core shroud
- Components not included in FEA that can be easily compared to analyzed components
  - Westinghouse lower core support structure
  - CE baffle bolts

Irradiation Induced Stress Relaxation/ Creep Waterfalls

- FEA model incorporates stress relaxation and creep effects (can rank effect)
  - Stress relaxation may have significant impact on other stress related mechanisms (e.g., IASCC)
  - Loss of bolt preload must be considered as contributing to wear and fatigue waterfalls

Wear Mechanism

- Difficult to compare or rank wear potential in identified components

Match inspection/trending monitoring program to component requirements

Wear Waterfalls

- Existing wear management programs
  - Westinghouse flux thimbles
  - CE thermal shield positioning pins
  - CE In-core Instrumentation thimble tubes
- Monitored through control rod drop times
- Inspect & monitor neutron noise
- Inspection requirements combined with integrated crack monitoring programs

What is a Reactor Internals Aging Management Program (AMP)?

- A document (procedure, instruction, specification) that describes a plant's program to ensure the long-term integrity and safe operation of PWR internal components

- Why is it required?
  - Previously required only for plants applying for license renewal
  - With publication of MRP-227, now required for all plants (Mandatory requirement under NEI 03-08)

Contents of an AMP

- What are the required contents of an AMP?
  - MRP-227 Appendix A defines the 10 elements which constitute an acceptable AMP
  - These elements are from NUREG-1801 (Generic Aging Lessons Learned [GALL] Report)

Regulatory Requirements for Reactor Internals AMPs

- GALL report NUREG-1801 Rev. 0, April 2001, XI.M16, "PWR Vessel Internals" identified 10 Attributes/Elements necessary for the Evaluation and Technical Basis

1. Scope of Program
2. Preventive Actions
3. Parameters Monitored/Inspected
4. Detection of Aging Effects
5. Monitoring and Trending
6. Acceptance Criteria
7. Corrective Actions
8. Confirmation Process
9. Administrative Controls
10. Operating Experience