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Reactor Vessel Internals Embrittlement

William L. Server
ATI Consulting, Pinehurst
USA

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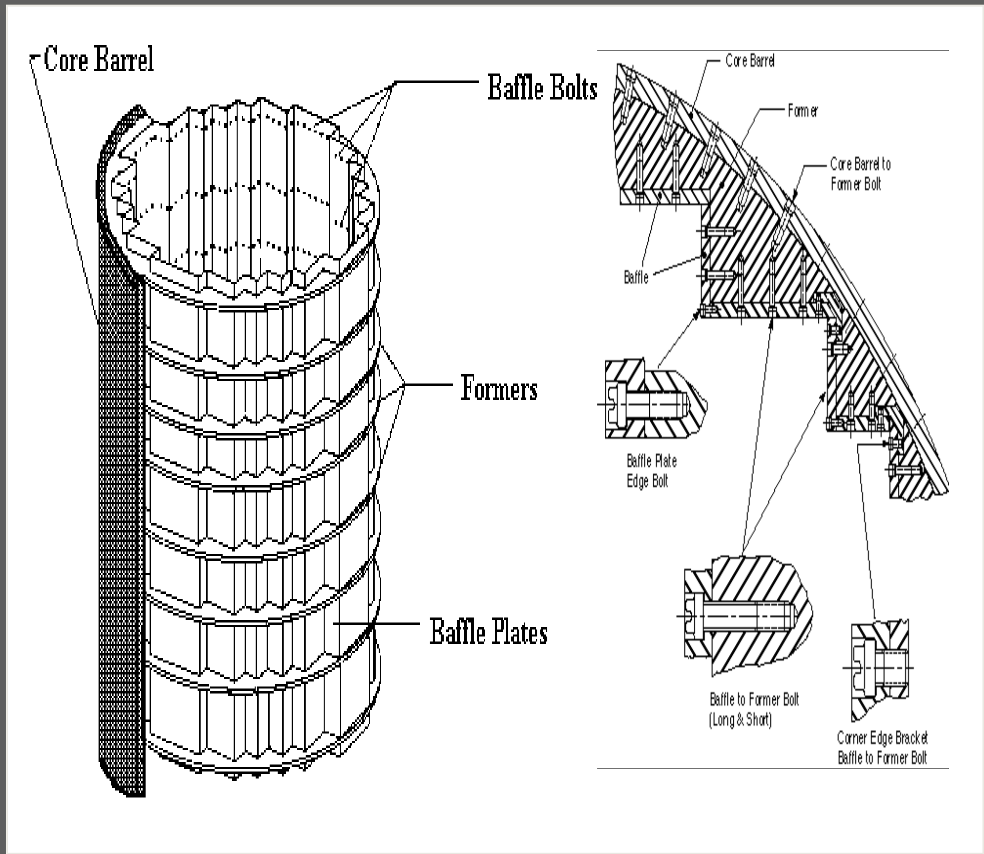
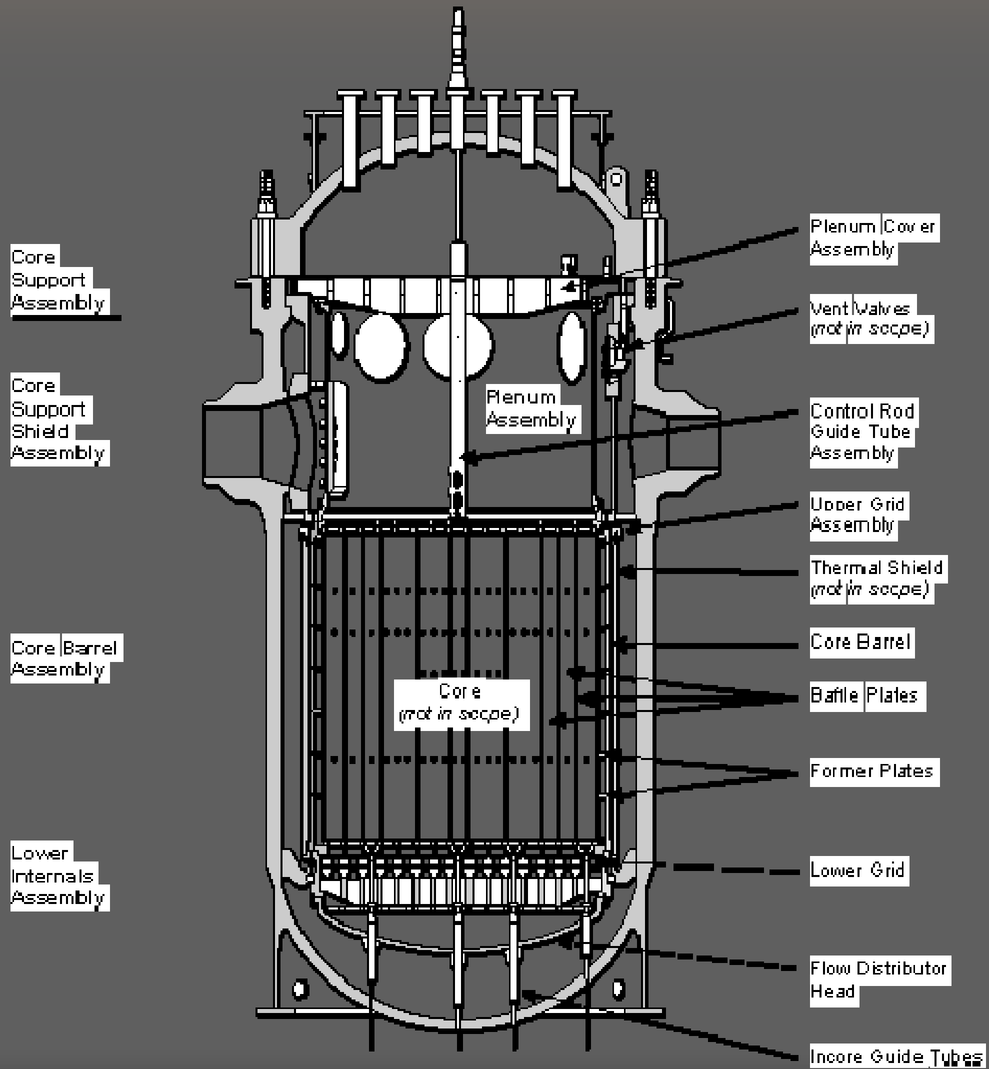
Reactor Internals Overview

Mechanisms:

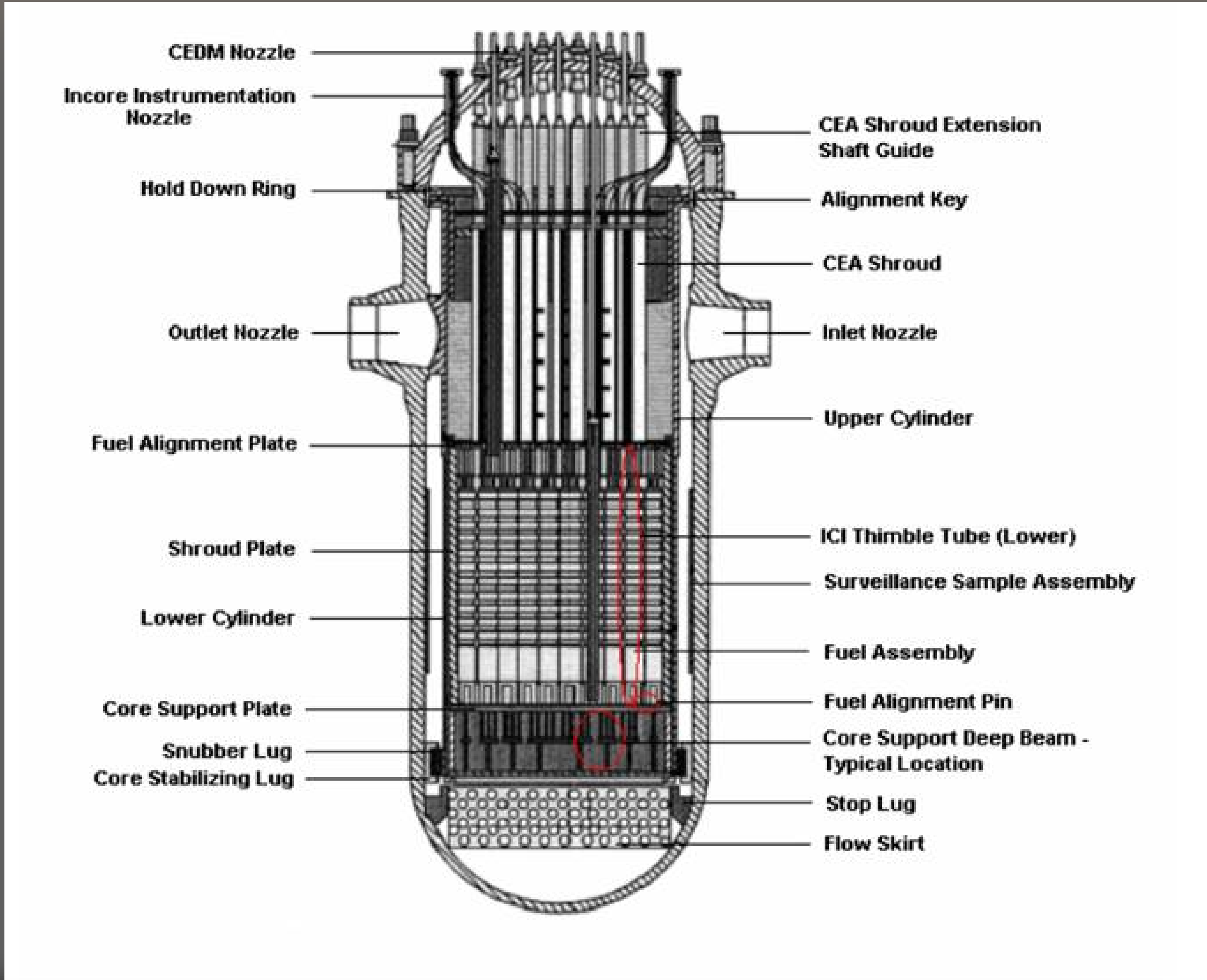
- Cracking due to Irradiation Assisted Stress Corrosion (IASCC) and Stress Corrosion (SCC)
- Reduction of Fracture Toughness due to Irradiation Embrittlement (IE) and Thermal Embrittlement (TE)
- Dimensional Changes due to Void Swelling (VS)
- Loss of Mechanical Closure Integrity due to Stress Relaxation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



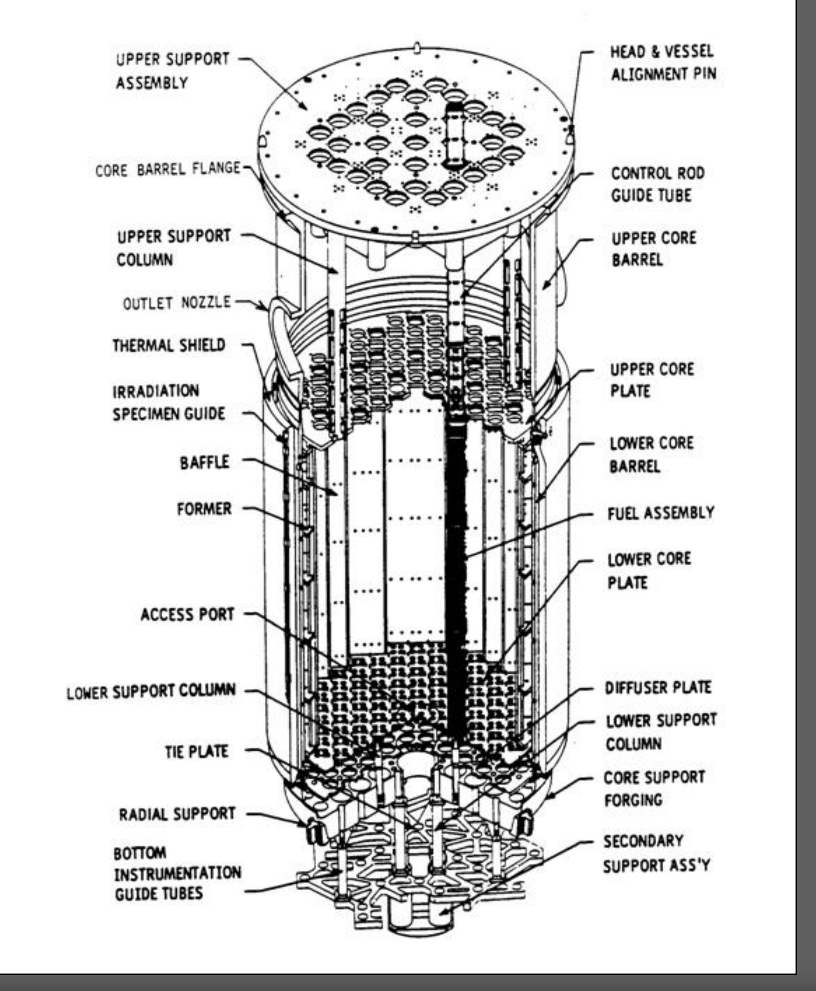
B&W Typical Internals Layout



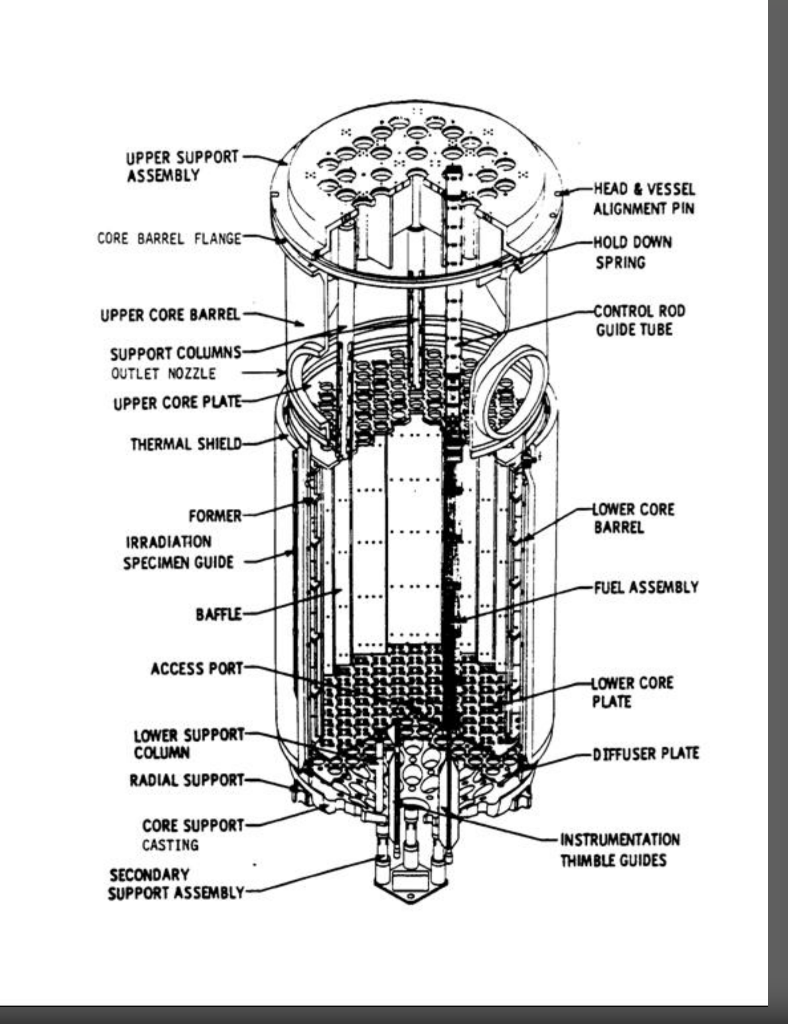
Typical CE Internals Layout



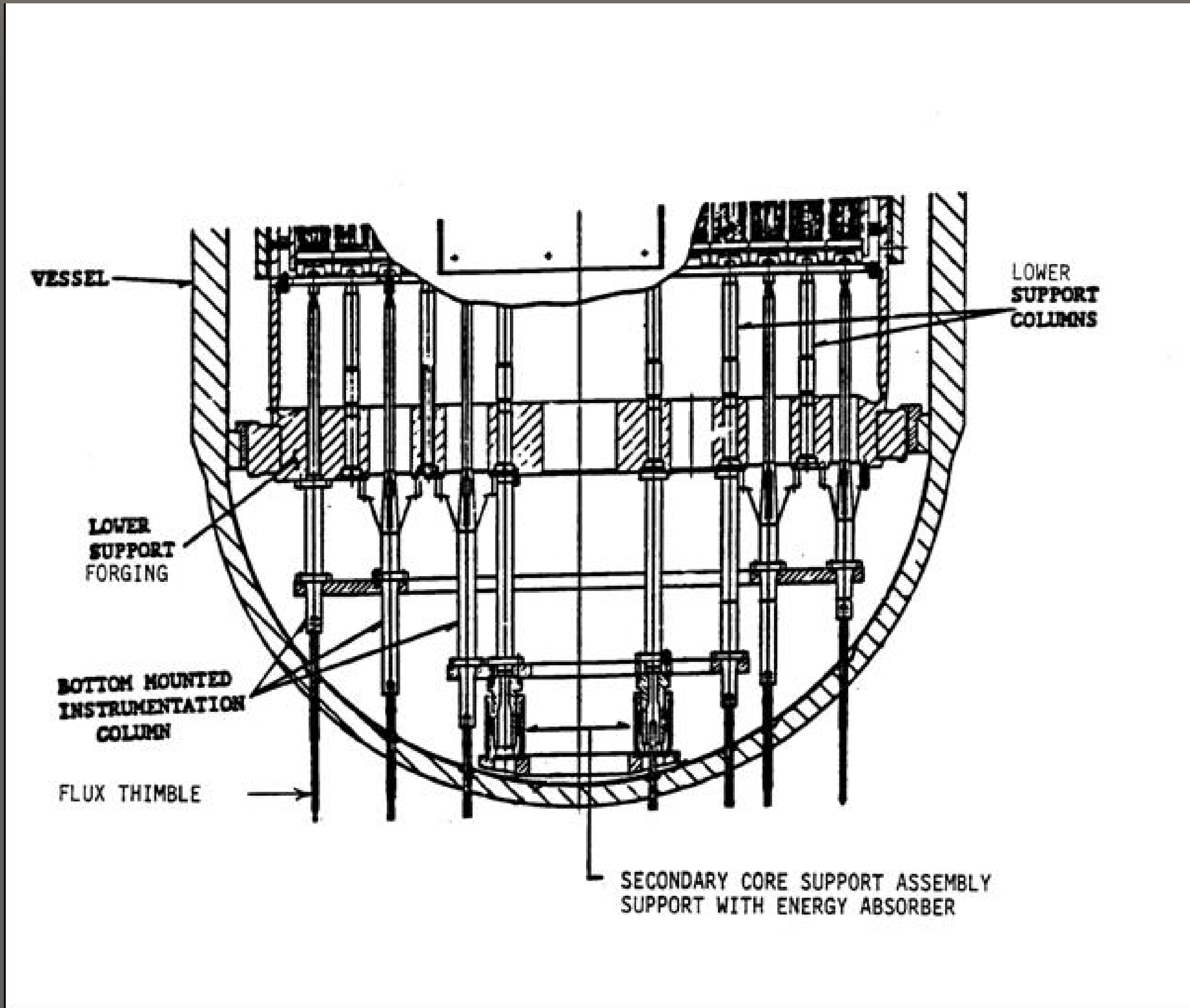
Westinghouse Internals with Forged Lower Support



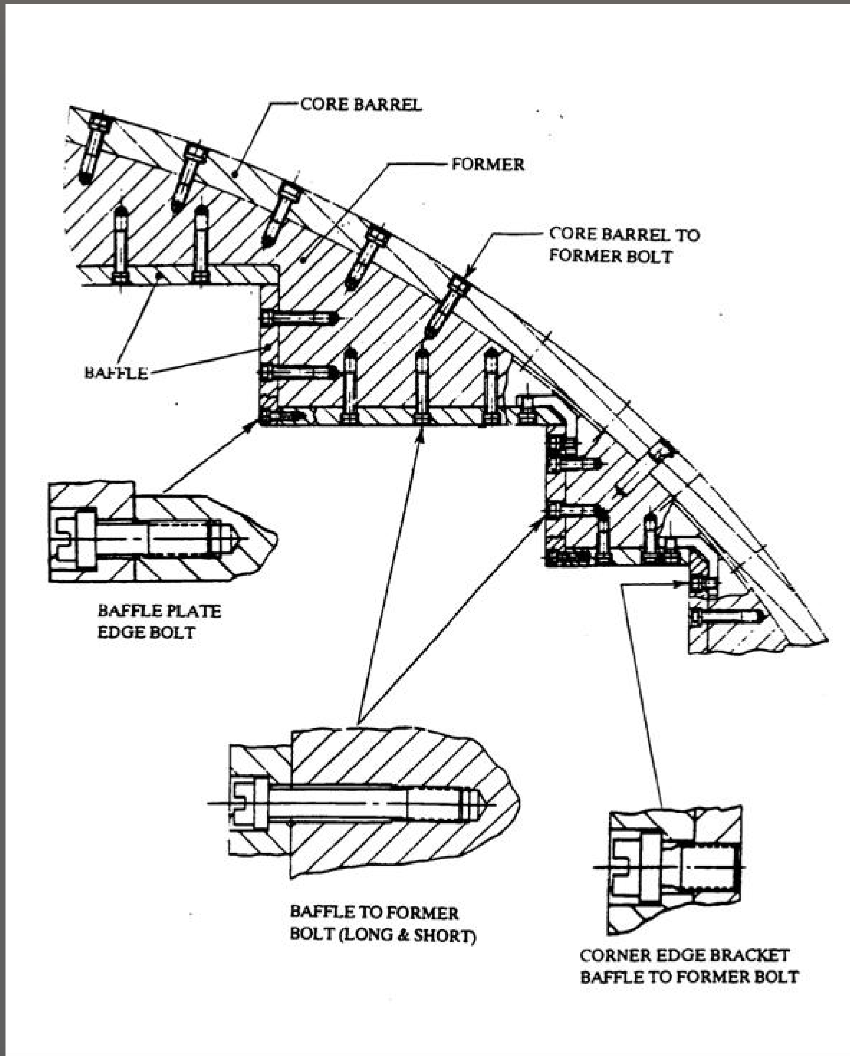
Westinghouse Internals with Cast Lower Support



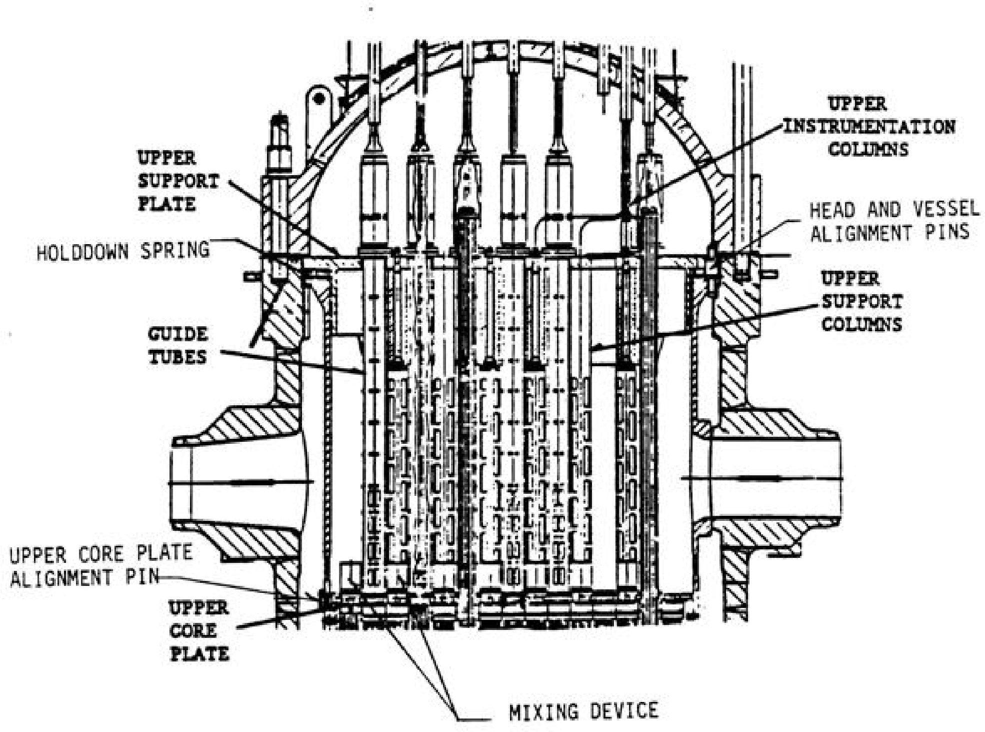
Reactor Internals Lower Section

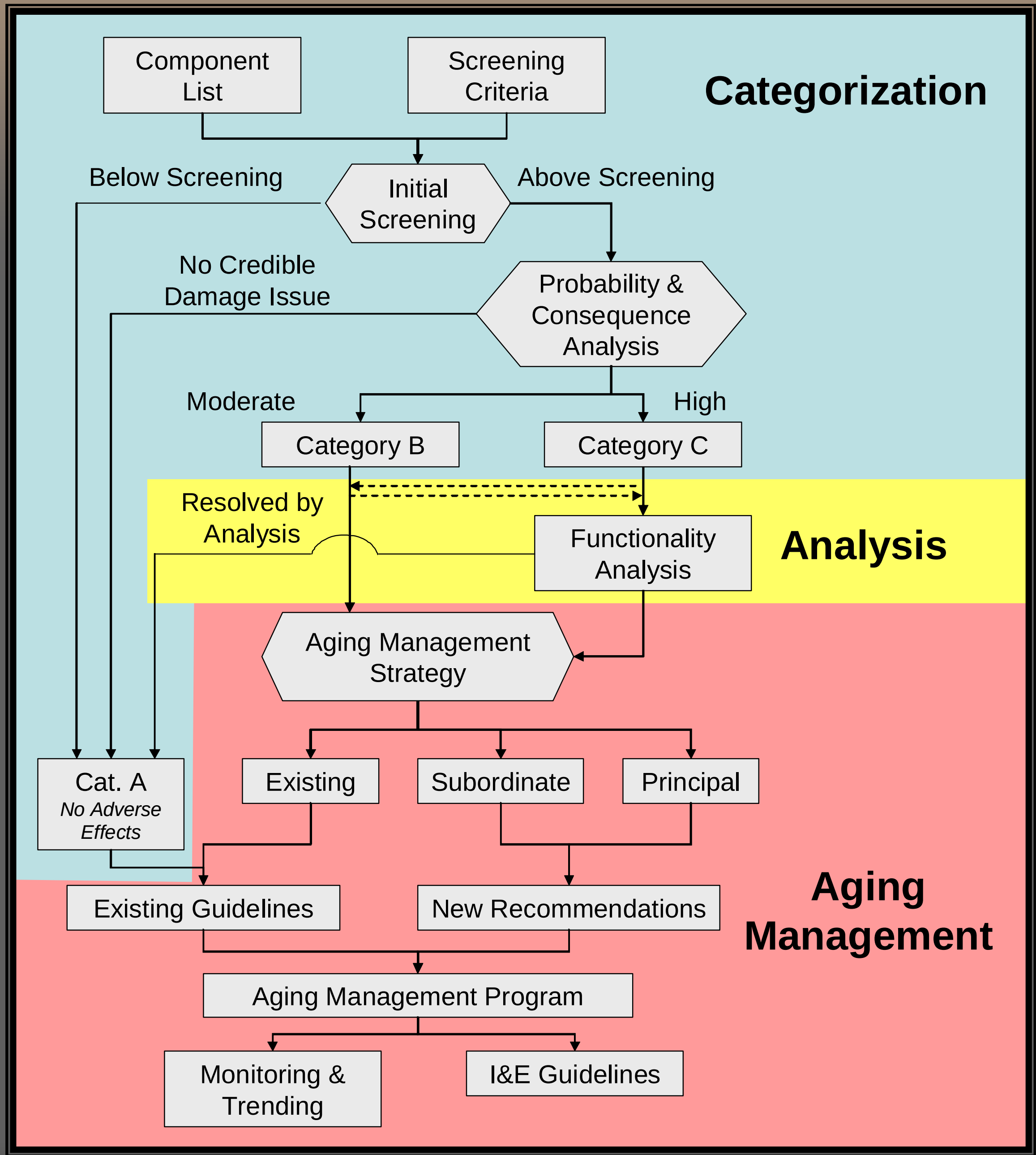


Typical Baffle Former Assembly



Reactor Internals Upper Section





Categorization
Component List
Screening Criteria
Below Screening
Initial Screening
Above Screening
No Credible Damage Issue
Probability & Consequence Analysis
Moderate
High
Category B
Category C
Resolved by Analysis
Functionality Analysis
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 inspection 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

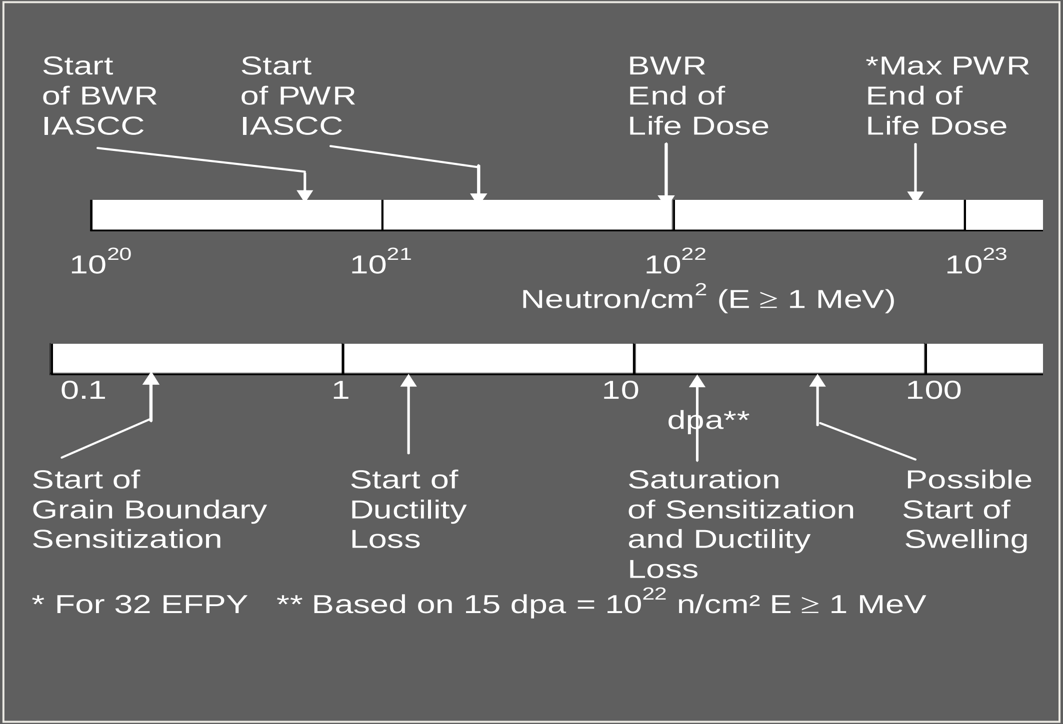
IASCC Waterfalls

- Stainless steel bolts (316 SS)
  - FEA intended to provide basis for ranking of time to failure
  - Limited number of CE plants with bolted baffles
- Stainless steel plate (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 x 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