The Role of Test Reactor Irradiations in Implementation of PWR Internals Inspection and Evaluation Guidelines

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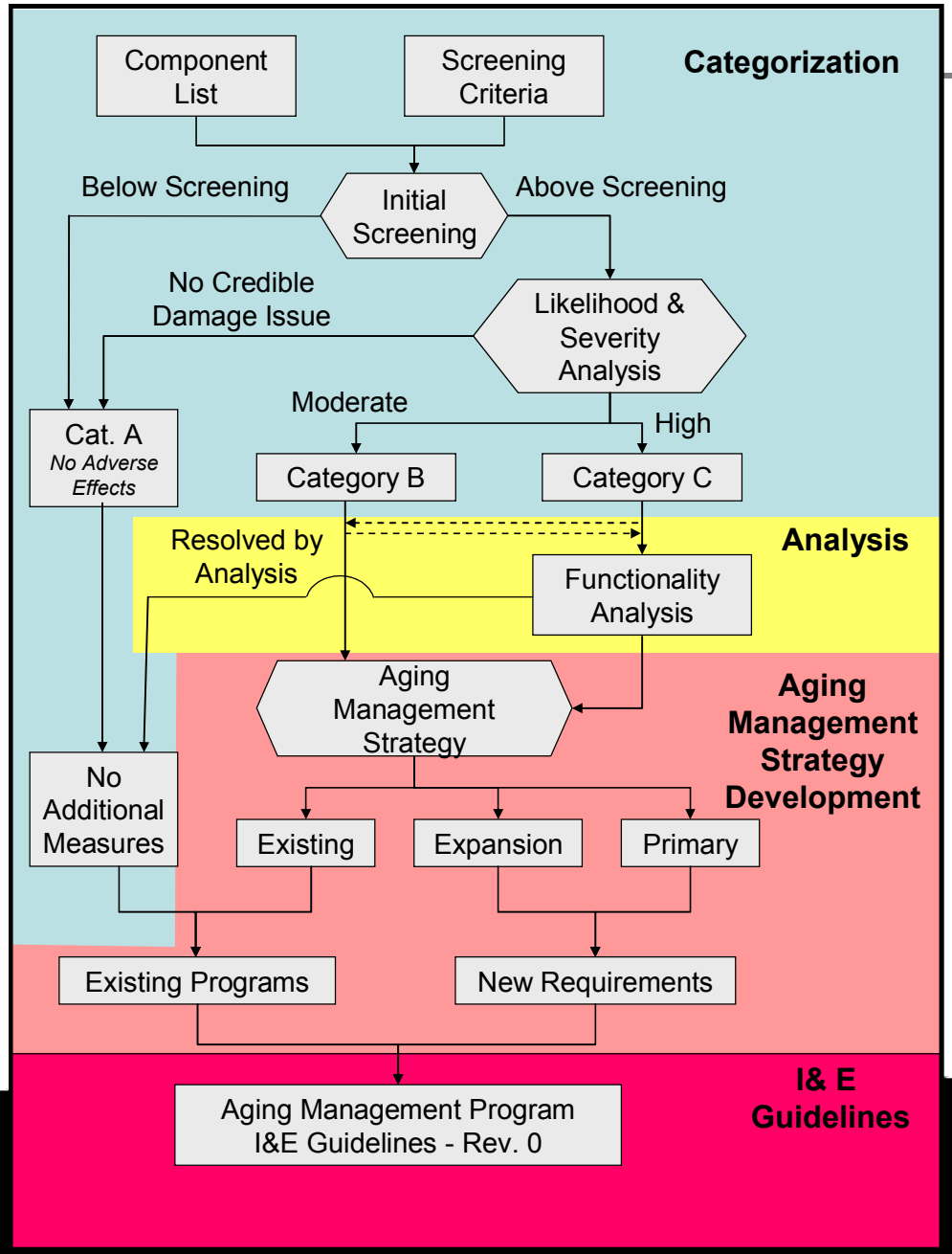
Westinghouse



PWR Internals Aging Management

- Screen all components for eight aging related degradation mechanisms
- Categorize likelihood and severity
- Functionality analysis of most critical components
- Provide Inspection and Evaluation Guidelines (MRP-227)
- Reactor Internals Aging Management Programs

Summary of NUREG-1801 Generic Aging Lessons Learned (GALL)

- Aging Management Program Elements
  - Scope Definition
  - Preventive Actions
  - Parameters to be Monitored or Inspected
  - Inspection (Detection of Aging Effects)
  - Monitoring or Trending
  - Acceptance Criteria (Action Levels)
  - Corrective Actions
  - Confirmation
  - Administrative Controls
  - Operating Experience Review (Reality Check)

Aging Management Strategy Input

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I&E Guidelines

Implementation

The Embrittlement Mechanisms

- Irradiation Embrittlement (Dose & Temperature)
- Thermal Embrittlement (Temperature & Composition)

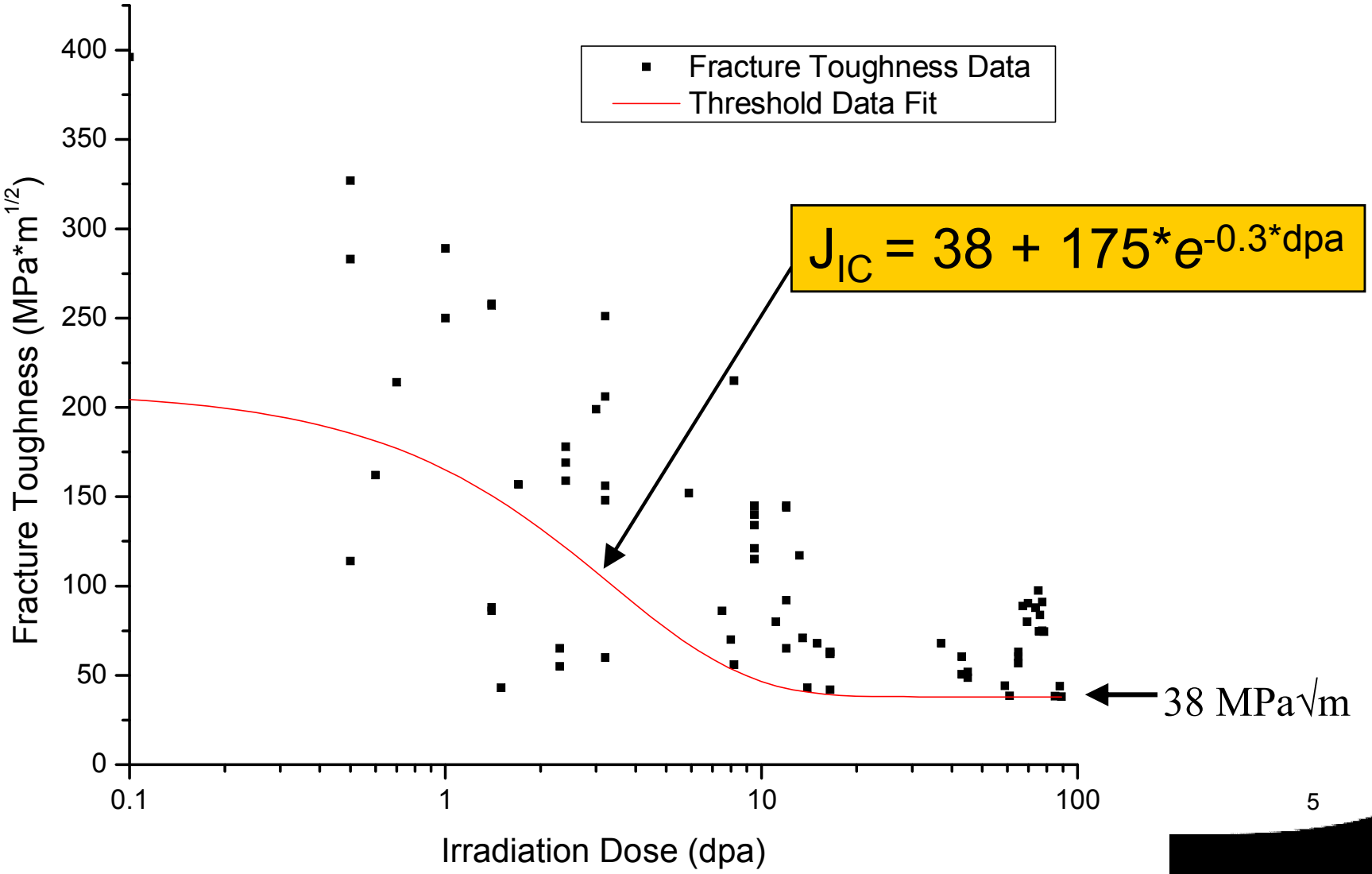
Changes in material properties

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

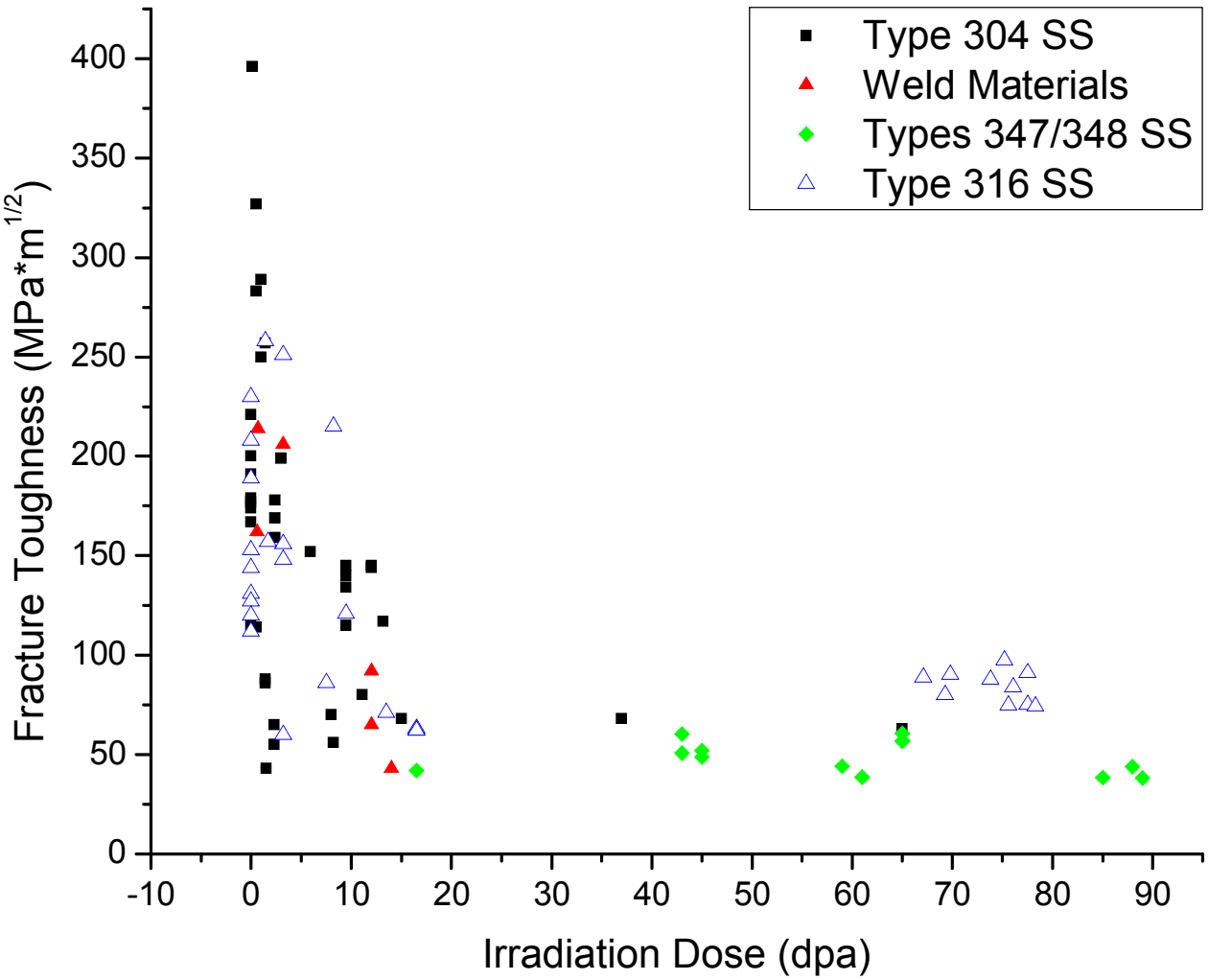
NDE *Direct inspection not required. Industry trending used to predict values.*

Evaluation *Based on bounding toughness curves.*

Irradiated Fracture Toughness



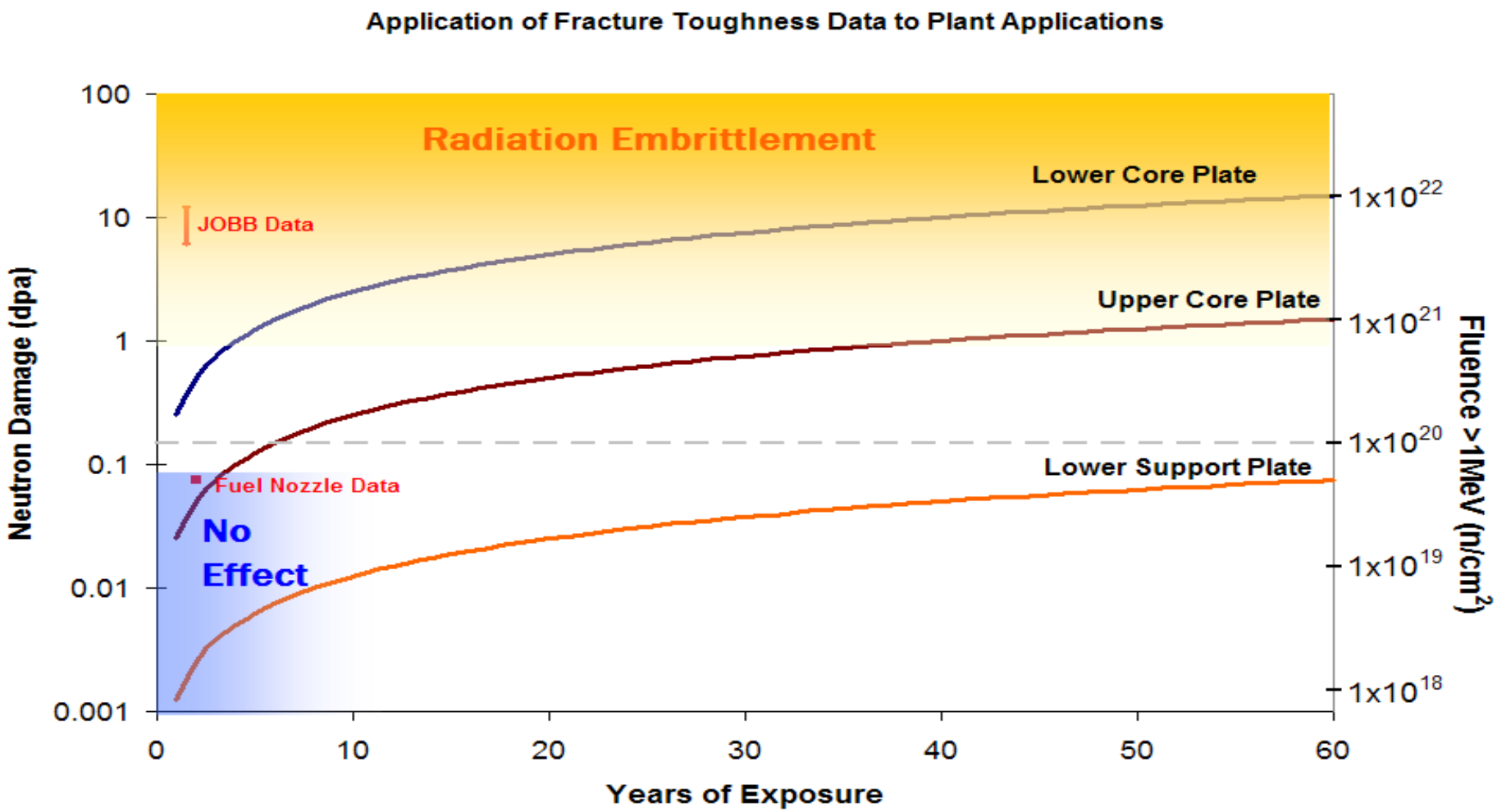
Materials Included:



Gaps

- Highly irradiated weld materials
- More data in general
- Credible data

Comparison of Aging/Irradiation Conditions



Data Needs for Fracture Toughness

- **Weld Data***
  - Required to determine critical crack sizes for inspection and evaluation of core shroud welds and core barrel welds
- **CASS Data on field exposed materials**
  - Difficult to simulate long-term synergistic aging/irradiation effects in high flux experiments
- **High Fluence Effects on R-Curve**
  - Most observations of fracture in high fluence materials indicate ductile fracture with limited resistance to crack extension
  - In large structures credit for crack growth resistance may be helpful
- **Low Temperature Fracture Phenomenon**
  - Multiple observations of low toughness, intergranular failures at room temperature in highly irradiated materials
  - Possible new kind of “ductile-to-brittle” transition?
  - Practical implications of this mechanisms unclear (not being considered in functionality analysis).

*** Analysis of 316 bolting based on fraction of un-cracked bolts. Fracture toughness not required.**

The Dimensional Stability Mechanisms

- Void Swelling (Temperature & Dose)
- Irradiation Enhanced Stress Relaxation/Creep (Dose & Stress)

  Component Distortion

  Modify Stress/Strain Distribution

  – Affects SCC, IASCC and Fatigue

NDE *Distortion and its subsequent effects expected to be observable in visual exam.*

Evaluation *Based on detailed FE analysis to observe combined effects on stress and strain.*

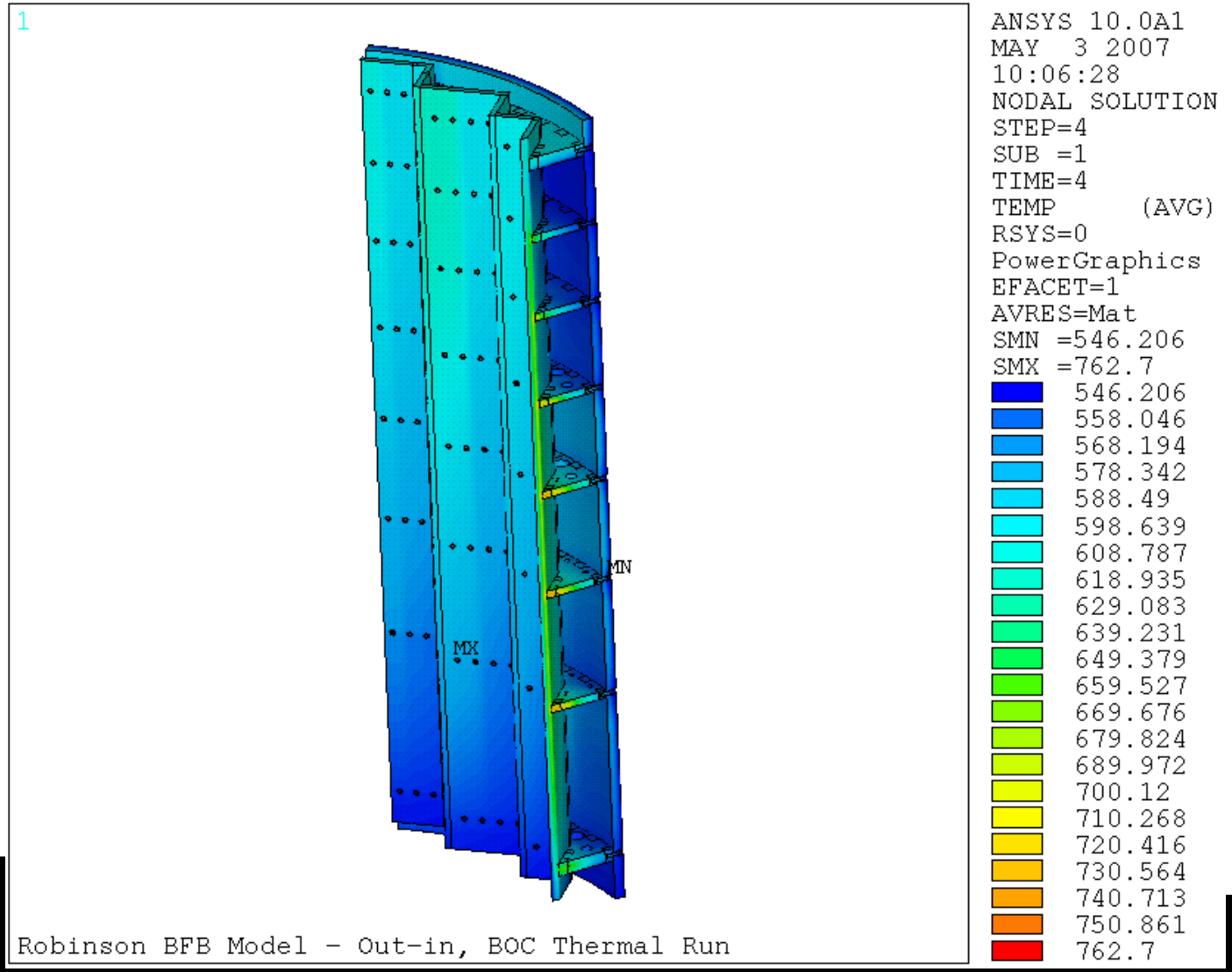
Structural Analysis Model
Iterative Simulation of Aging

Iterate

- FEA – ANSYS (elastic-plastic)
  - ANATECH Materials Subroutine
    - Calculate Properties at time t
      - Irradiated Mechanical Properties
      - Creep Rate
      - Swelling Rate
    - Update σ / ε for Δt

Baffle & Former

Temperatures and Doses Sufficient for Swelling

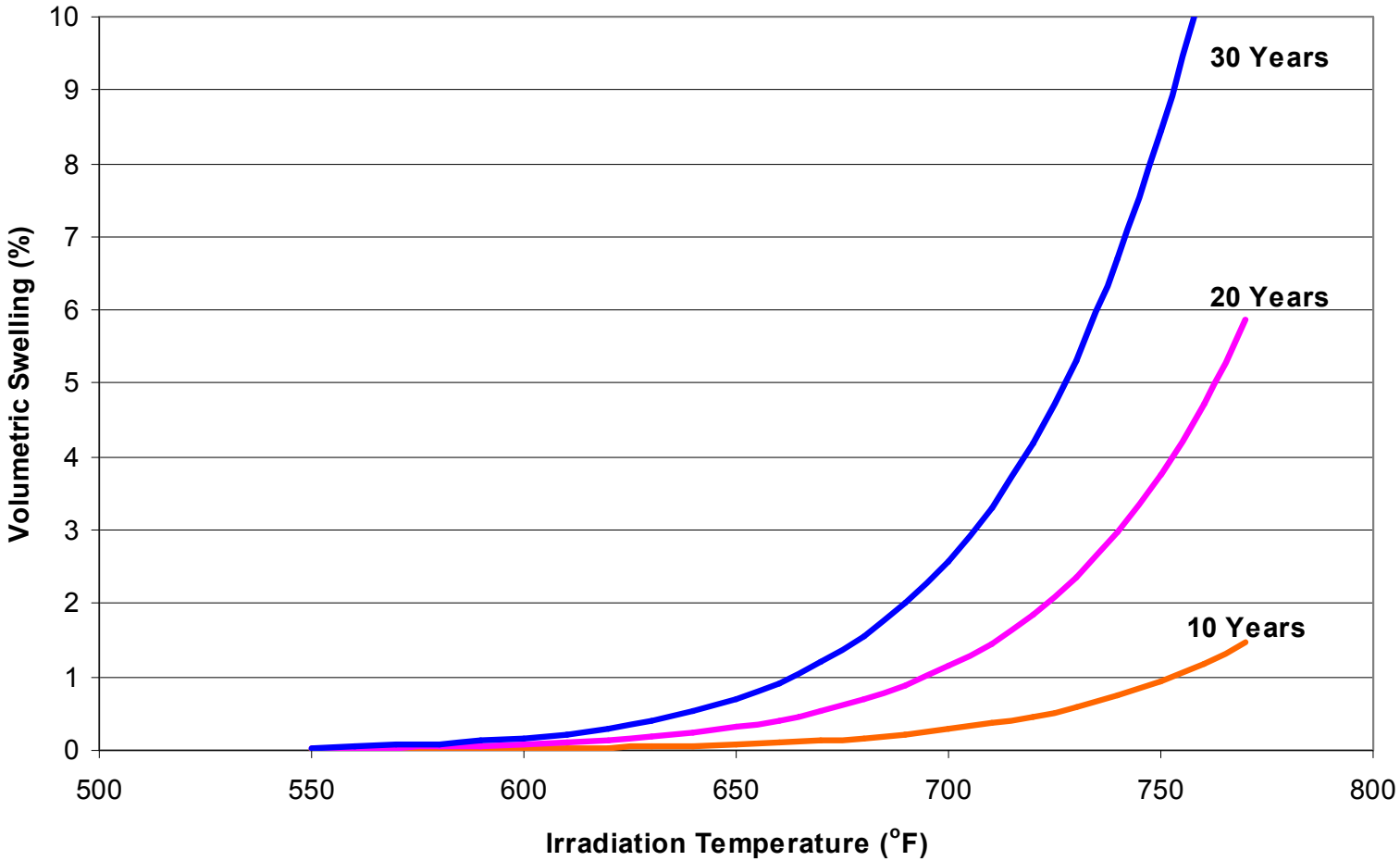


Surface Temperature Results:

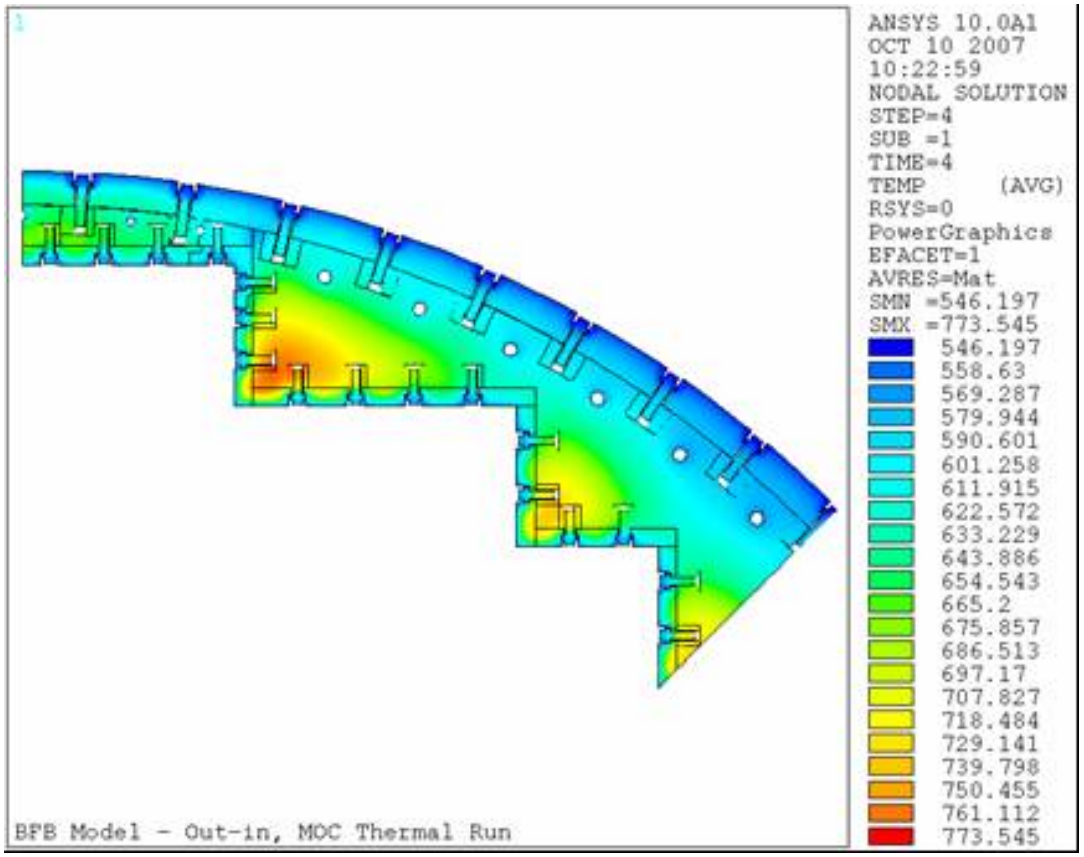
Out-In, BOC

Swelling Model (304 SS)

Dose Rate = 1dpa/year



Temperature in Baffle-Former-Barrel Assembly (30 EFPY "Out-in" Core Load)



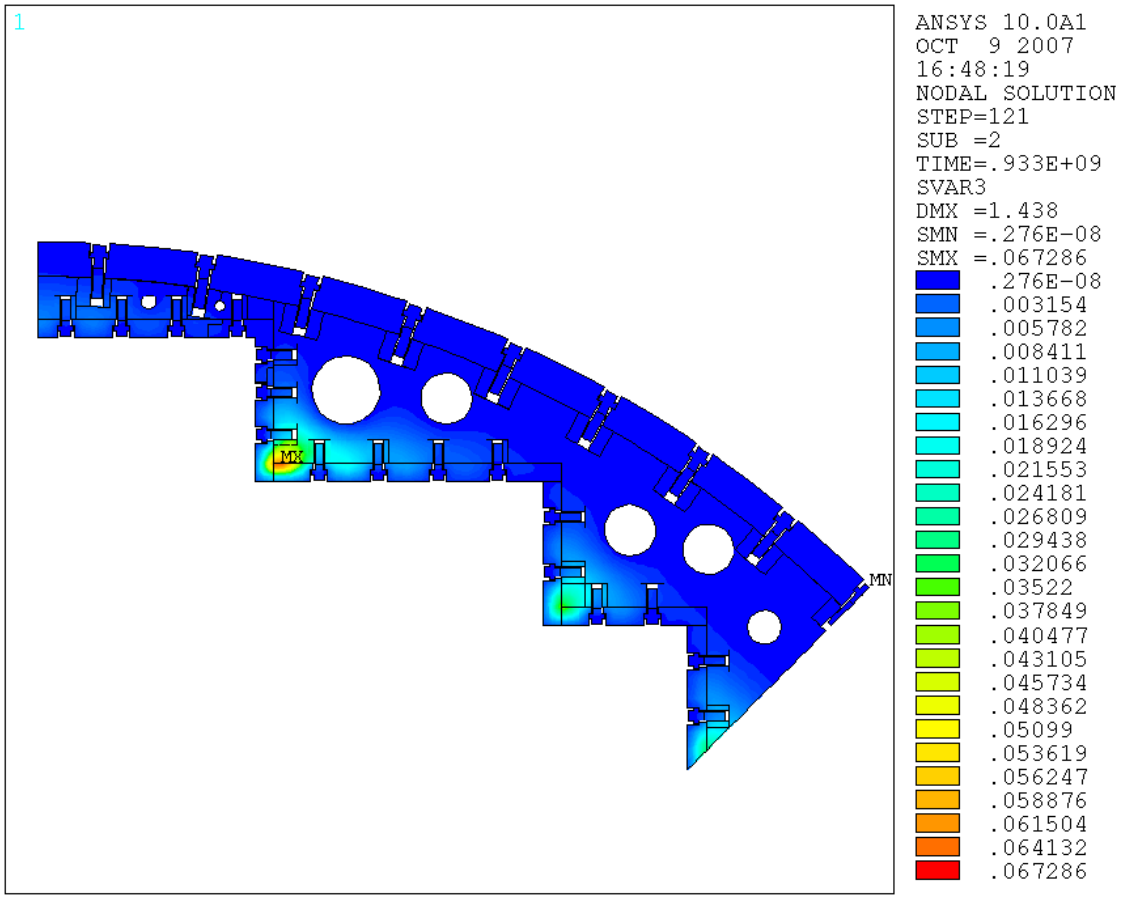
Cut through mid-plane of former level 4.

Dark Blue ~ 550°F

Light Blue ~ 600°F

Orange ~ 760°F

Swelling in Baffle-Former-Barrel Assembly (30 EFPY)



Cut through mid-plane of former level 2.

Dark Blue < 1%

Light Blue ~ 3%

Orange > 15%

Data Needs for Void Swelling

- **Evaluate Results of Functionality Analysis**
  - ANATECH Usermat routine contains a “consensus” swelling model based on minimal PWR data.
  - Incubation Doses
    - Temperature 300-400°C
    - 316SS, 304SS, 347SS
    - Cold work

Data Needs for Irradiation Creep/Stress Relaxation

- **Low Dose Transient Behavior**
  - Bolt load data indicates a delay before onset of stress relaxation
- **Material sensitivity**

The Cracking Mechanisms

- SCC (Stress & CW)
- IASCC (Stress and dose)
- Fatigue (Transient Loading)

Produce observable cracks
Most probable in regions of stress concentration

NDE

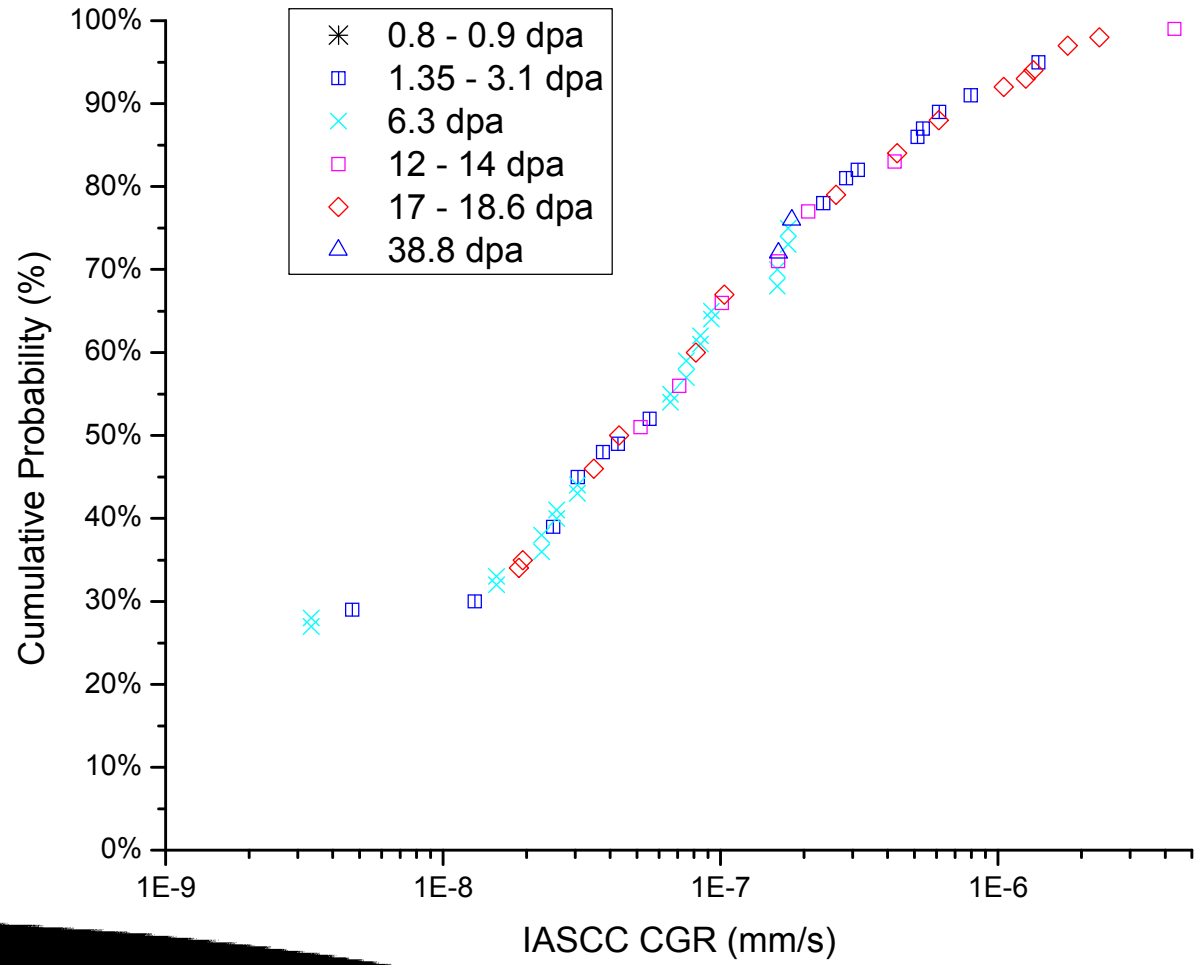
Crack observation generally implies VT-1/EVT-1 or UT (EC not a primary tool for internals). Broken parts/gross cracking detected with VT-3.

Evaluation

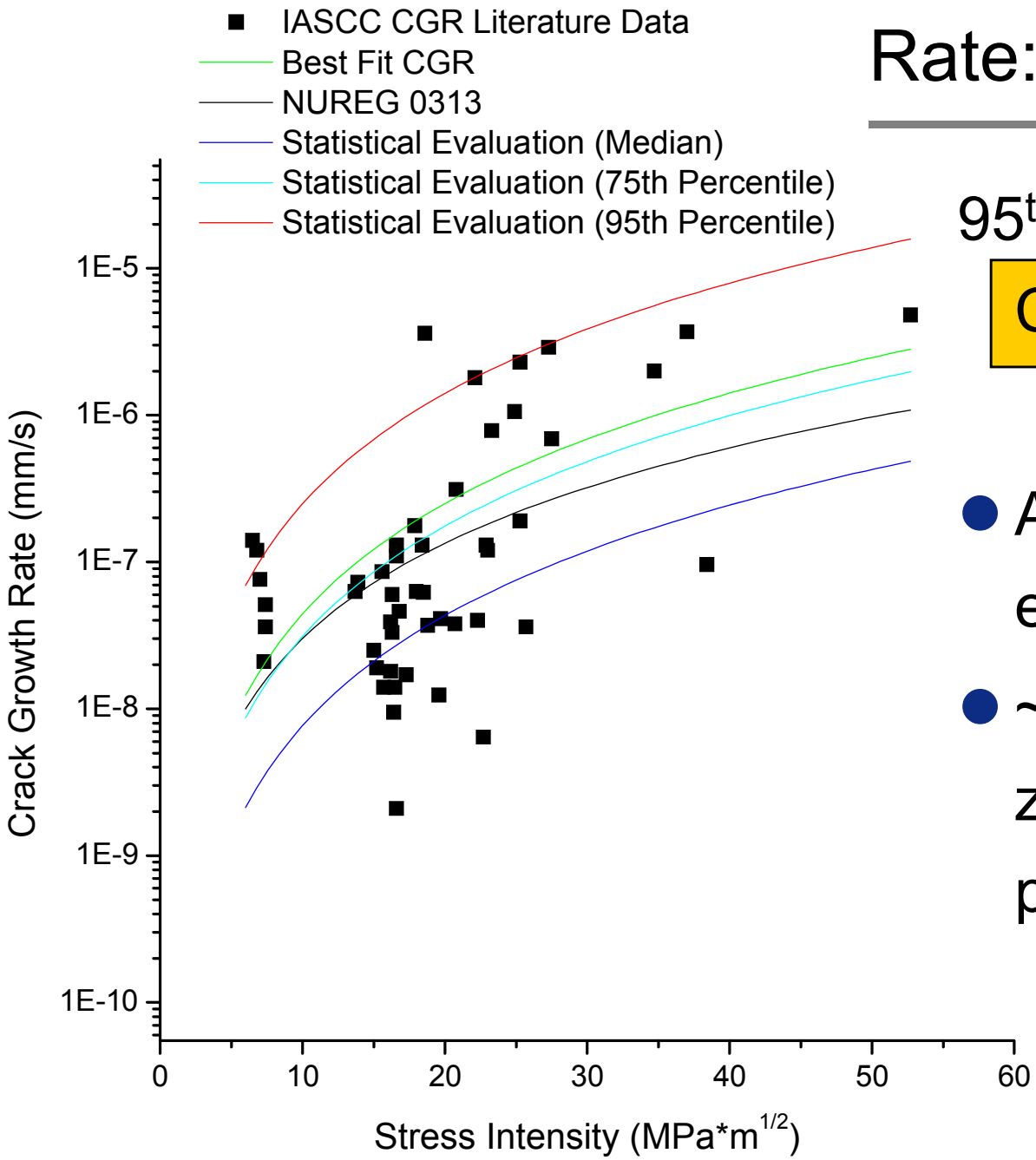
Predict crack growth prior to next inspection.
Calculate critical crack size.

IASCC Effect of Irradiation Dose

- Irradiation dose shows no apparent effect on CGR in the results from this dataset



IASCC Crack Growth Rate: Curve Fitting



95th Percentile Curve:

$$CGR = (7.84E\text{-}10)*K^{2.5}$$

- Assumed stress intensity exponent of 2.5
- ~30% of the data was zero and is not on this plot

Data Needs for IASCC Initiation

- **Exposure Times Greater than 3,000 hours**
  - 85% of 60yrs = 447,000 hrs
  - Most existing lab failures < 600hrs
  - No lab failures 1000-4000 hrs (<50% σ_y)
- **High Fluence 304 SS Plate Data**
  - Most lab data on 316 thimble tubes and bolts
  - Applicable to baffle/shroud plates or former plates
- **High Fluence Weld Data**
  - Most lab data on 316 thimble tubes and bolts
  - Applicable to core shroud welds
- **Standardized Test Methods**

Data Needs for IASCC Crack Growth Rates

- **High Fluence Weld Data**
  - Most lab data on 316 thimble tubes and bolts
  - Applicable to core shroud welds
- **Low/Intermediate Fluence Weld/HAZ Data**
  - Applicable to core barrel beltline welds
- **High Fluence 304 SS Plate Data**
  - Most lab data on 316 thimble tubes and bolts
  - Applicable to baffle/shroud plates or former plates
- **No requirement to analyze flaws in bolts**
  - Calculate minimum bolting patterns and replace as necessary

Data Needs for Fatigue

- **Determine Realistic Fatigue Parameters for Reactor Internals (lower core plate, baffle bolts, …)**
  - Low cycle (high strain) conditions have largest potential irradiation and environmental effects
- **Evaluate Impact of Draft Regulatory Guide DG-1144**
  - NRC intends to apply to license extension evaluations
  - Small changes in SS fatigue curve
  - New environmental correction factor (F_{en})
- **Environmental Correction Factors for Irradiated Materials**
  - Targeted testing to determine applicability of F_{en} to Irradiated Materials
  - Test conditions based on realistic fatigue parameters with maximum expected F_{en} value

The Wear Mechanism

- Difficult to compare or rank wear potential in identified components

Wear scars may be observed visually. Physical measurements may be required for evaluation.

Summary

- I&E Guideline recommendations available (MRP-227)
- Implementation of MRP-227 Guidelines will require additional irradiated material properties for:
  - Fracture Toughness
  - Void Swelling
  - IASCC Susceptibility
  - IASCC Crack Growth Rates

Questions?

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