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# Reactor Pressure Vessel Embrittlement Monitoring and Prediction in Long-Term Operation

**The Public Meeting Will Begin Shortly …**

**Meeting support contact:**
**Glenna.Lappert@nrc.gov**

- All meeting Information (i.e., notice, slides, transcript, and summary) will be publicly available 30 days after this meeting
- Please log into both webinar and bridgeline. Visuals will be through WebEx and audio will be through the bridgeline
  - https://usnrc.webex.com/usnrc/onstage/g.php?MTID=e544840cfd44284243639a4ad7a9f8974
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  - Passcode: 8425538#

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# Reactor Pressure Vessel Embrittlement Monitoring and Prediction in Long-Term Operation

Public Meeting

October 18, 2021

ADAMS Accession No. ML21270A002

# Agenda

- Welcome / Introductions / Logistics
- NRC Presentation:  Reactor Pressure Vessel Embrittlement Monitoring and Prediction in Long-Term Operation
- Public Presentations
- Discussion and Q&A
- Closing Remarks, and Adjourn

# Welcome & Introductions

**Welcome**

- Robert Taylor, NRR Deputy Office Director

**Introductions**

- David Rudland, NRR Senior Technical Lead
- Allen Hiser, NRR Senior Technical Lead
- Stewart Schneider, NMSS Senior Project Manager
- Joan Olmstead, NRC Facilitator

# Meeting Logistics

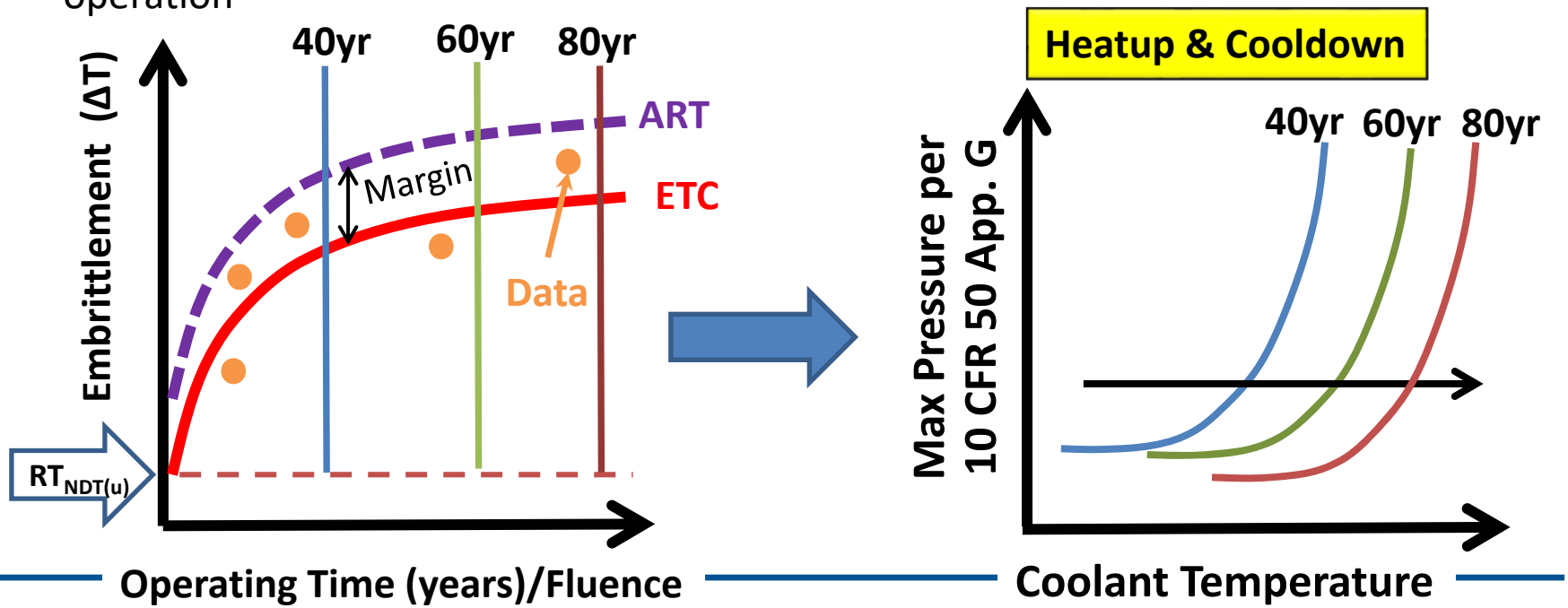
- Meeting visuals are through WebEx.
- Meeting audio is through the bridgeline.
- Participants are in listen-only mode until the question and answer period. The operator will open phone lines during this time.
- This is an Information Meeting with a Question and Answer Session. The purpose of this meeting is for the NRC staff to meet directly with individuals to discuss regulatory and technical issues. Attendees will have an opportunity to ask questions of the NRC staff or give feedback about the issues discussed after all presentations; however, the NRC is not actively soliciting comments towards regulatory decisions at this meeting.
- This meeting is being transcribed. The transcript will be available through the meeting summary.

# Meeting Purpose

- Continue discussion of issues from May 2020 public meeting (ML20168A008)
  - Regulatory Guide 1.99 Rev 2 (RG 1.99) and 10 CFR 50.61 embrittlement trend curve
  - Appendix H surveillance testing
- Discuss a holistic risk-informed analysis of these issues and its potential impact on reactor pressure vessel (RPV) integrity
- This is a technical discussion; no regulatory decisions will be made at today's meeting
- NRC staff would like feedback on analysis approach and results

# Background
# Monitoring and Prediction of Embrittlement

- Embrittlement Trend Curve (ETC) provides estimates of change in fracture toughness ($\Delta T$ or $\Delta RT_{NDT}$) as a function of fluence
- Surveillance capsule testing provides monitoring to ensure ETC predicts plant specific behavior properly
- Together they are used to determine pressure-temperature (PT) limits for normal operation



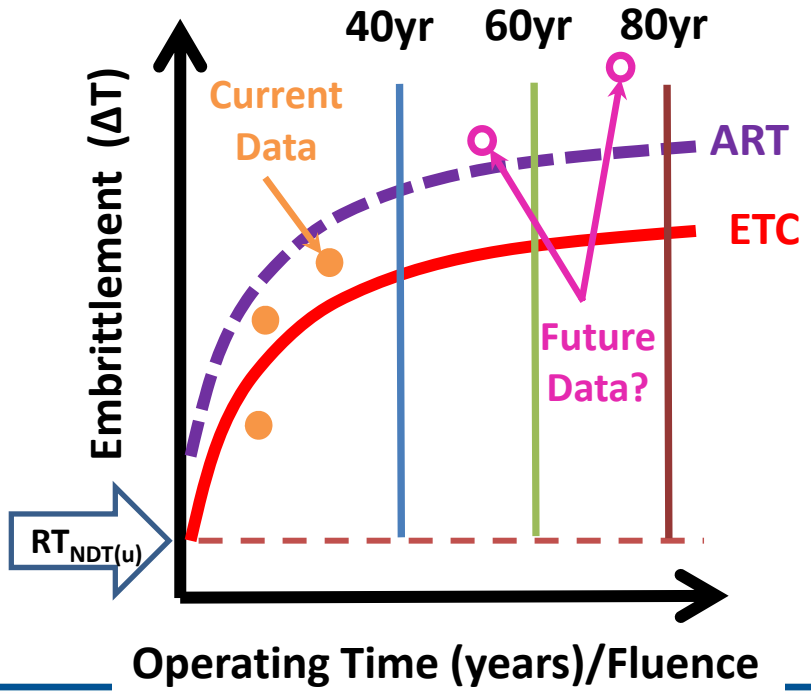

ART = Adjusted Reference Temperature

# Ideal Scenario

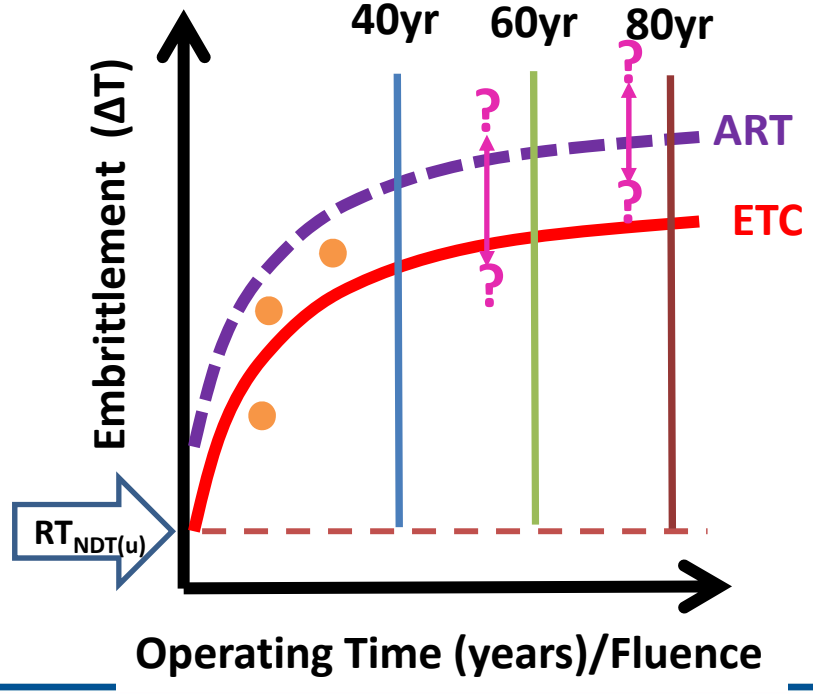
- ETC provides conservative predictions of embrittlement
- Surveillance data covers all operating periods

# Potential Uncertainty Sources

***IF*** **ETC under-predicts measurements**




***IF*** **Limited Surveillance Data is Available**



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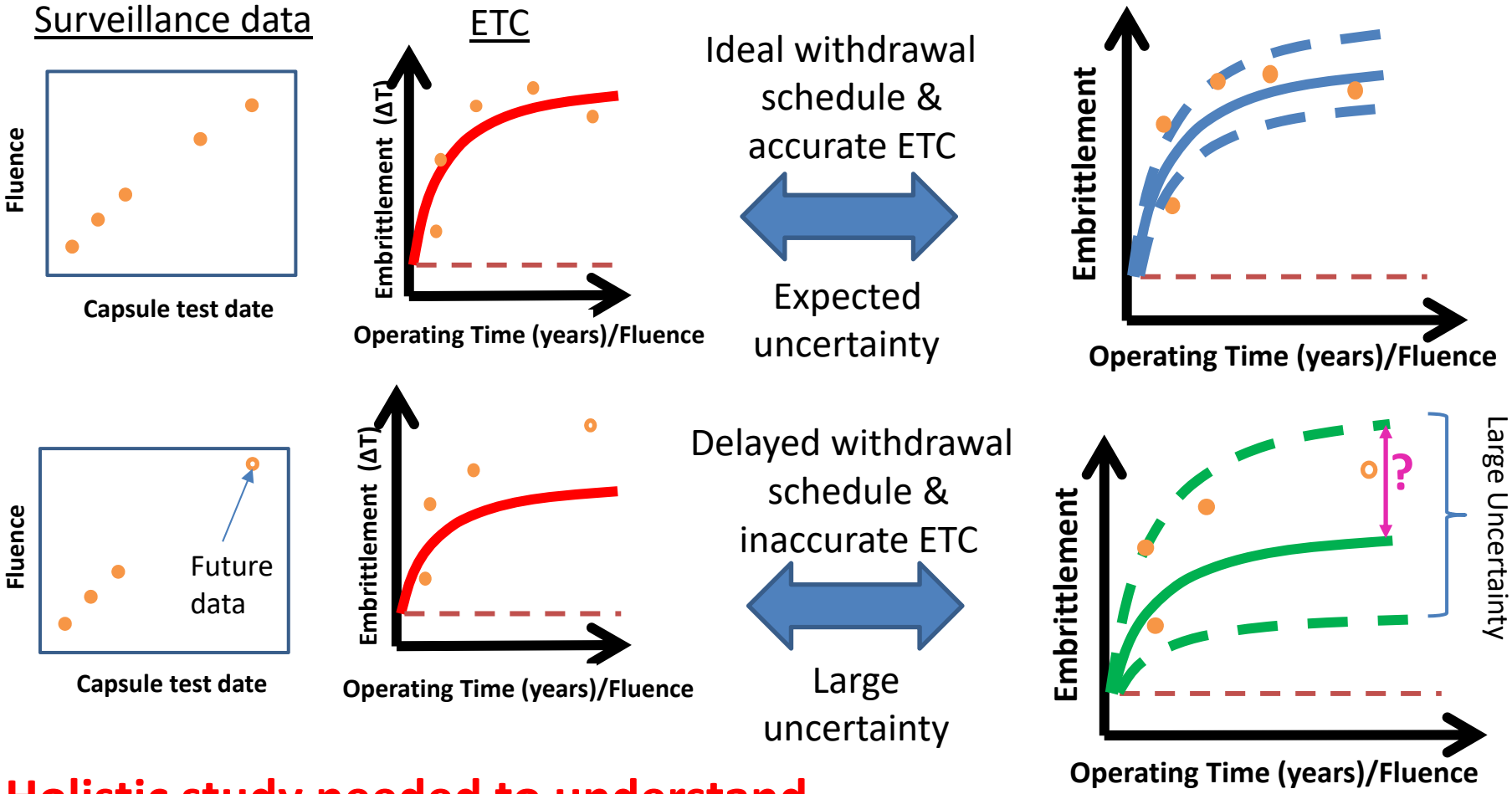
# Embrittlement Uncertainty




**Holistic study needed to understand impact of uncertainty**

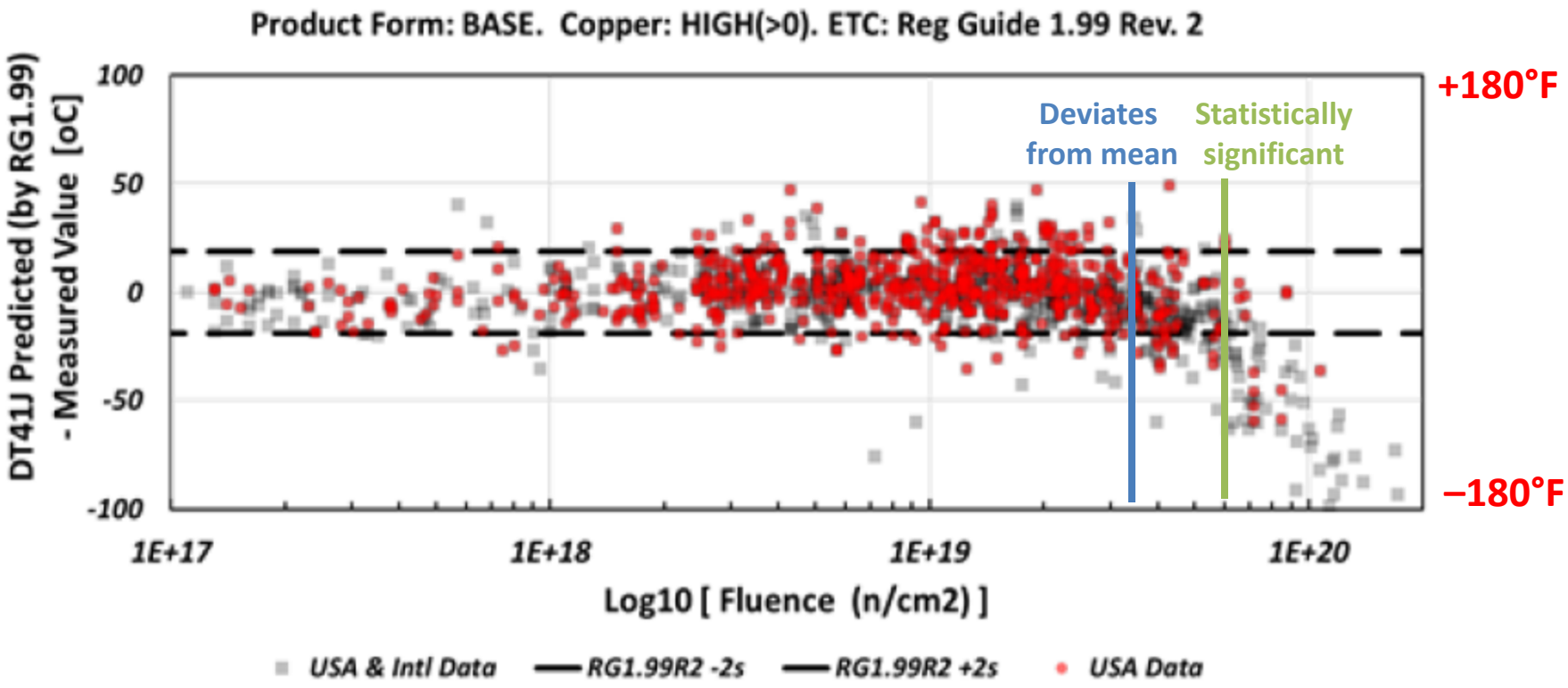
# Current Perspective of Potential Issue

- High confidence that currently operating plants remain safe
- Recent licensing actions remain valid
- Insufficient embrittlement monitoring and under predictions of reactor vessel embrittlement will eventually (after about 10 years) impact the staff's confidence in the integrity of the reactor pressure vessel in long-term operation, i.e., both safety margins and performance monitoring may be impacted
- Further work is needed to determine which plants are impacted by this potential issue

# Embrittlement Trend Curve

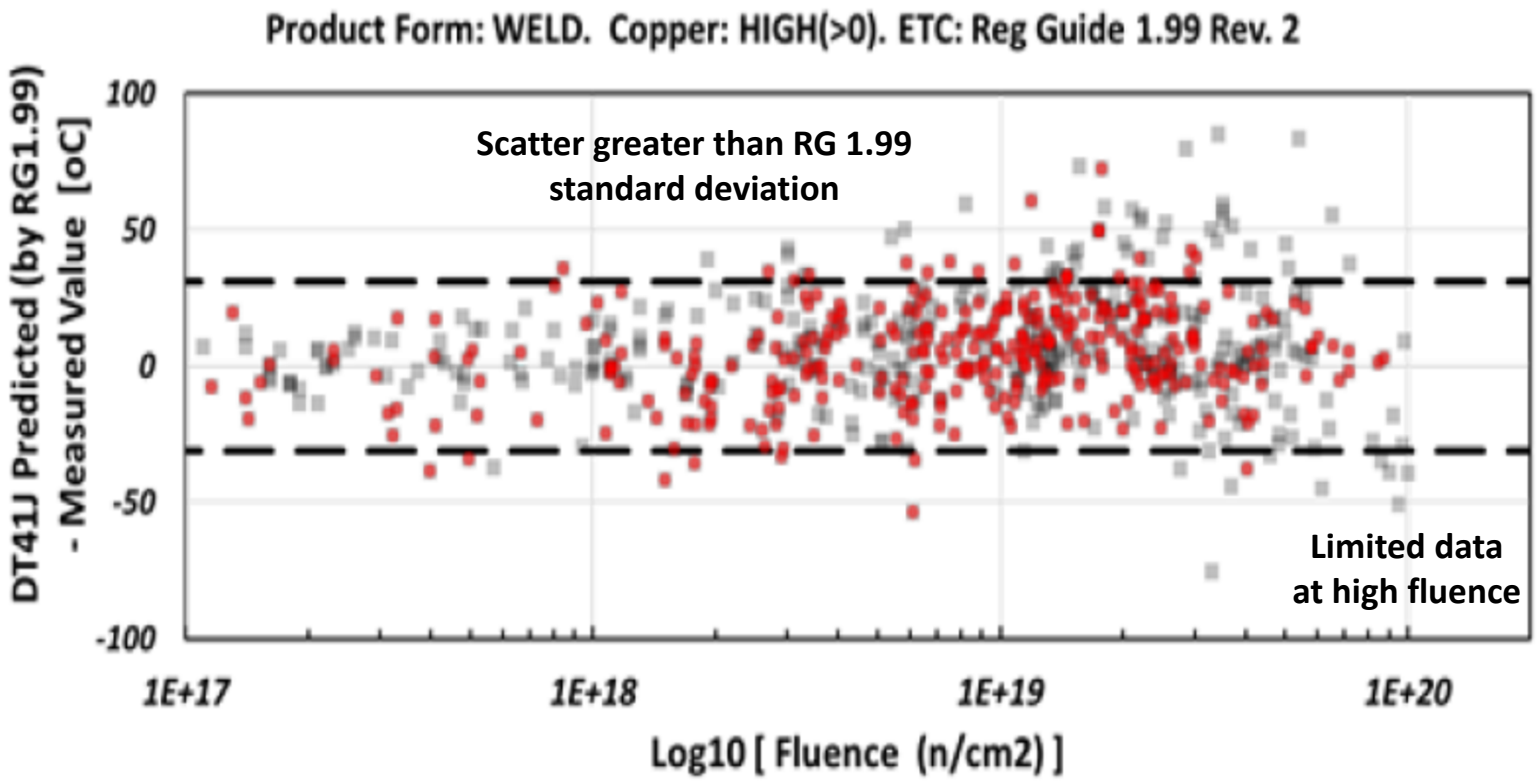
- May 1988, NRC published RG 1.99, which contained an improved embrittlement trend curve (ETC)
  - Fit based on 177 datapoints
- June 1991, NRC updated 10 CFR 50.61 to include the ETC from RG 1.99
  - Addressed lower than measured predictions (up to 60°F) of embrittlement in some vessels
- This ETC was re-evaluated for continued adequacy in 2014 (ML13346A003) and in more detail in 2019 (ML19203A089)

# Issue – ETC

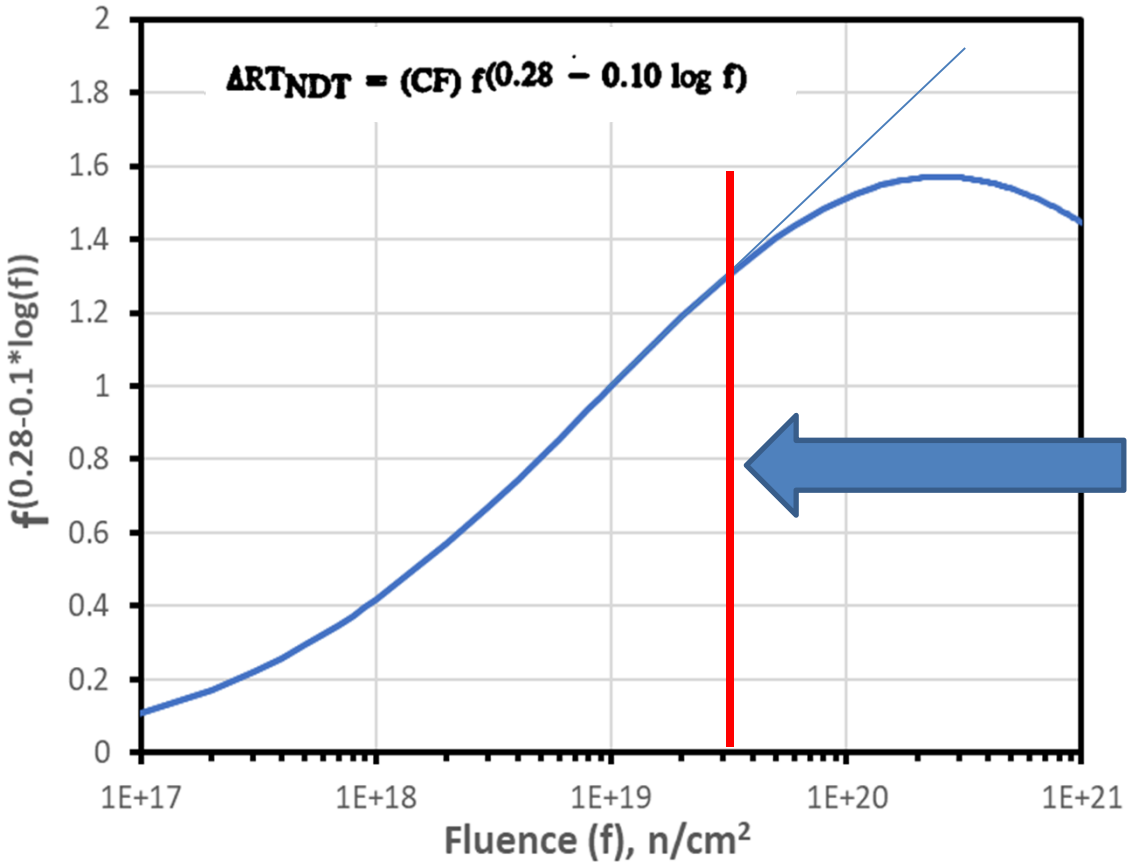



DT41J = $\Delta T_{41J}$ is a measurement of embrittlement representing the shift in transition temperature from brittle to ductile fracture at an impact toughness of 41J

# Issue – ETC



# Issue – ETC Fluence Function




Fluence function begins to "flatten" at the same fluence level underprediction occurs in Slide 12

# Surveillance Capsule Delays

- Appendix H to 10 CFR Part 50 requires periodic monitoring of changes in fracture toughness caused by neutron embrittlement
  - ASTM standard (E185-82) allows final capsule fluence to be 2X RPV "design" fluence – plants change (intended 40-year) design fluence to current license length (e.g., 60 or 80 years)
  - ASTM standard (for 40 years) permits holding last capsule without testing

- Commission finding ("Perry decision" NRC Administrative Letter 97-04) that staff review of requests to change capsule withdrawal schedules is limited to verification of conformance with the ASTM standard (i.e., not based on technical or safety considerations)
  - Capsule withdraw and testing repeatedly delayed in some cases to achieve higher fluence

# License Renewal

- Regulations are unchanged; surveillance program addressed in guidance
  - Guidance provides flexibility for licensees to demonstrate adequate management of RPV embrittlement due to varying plant-specific circumstances

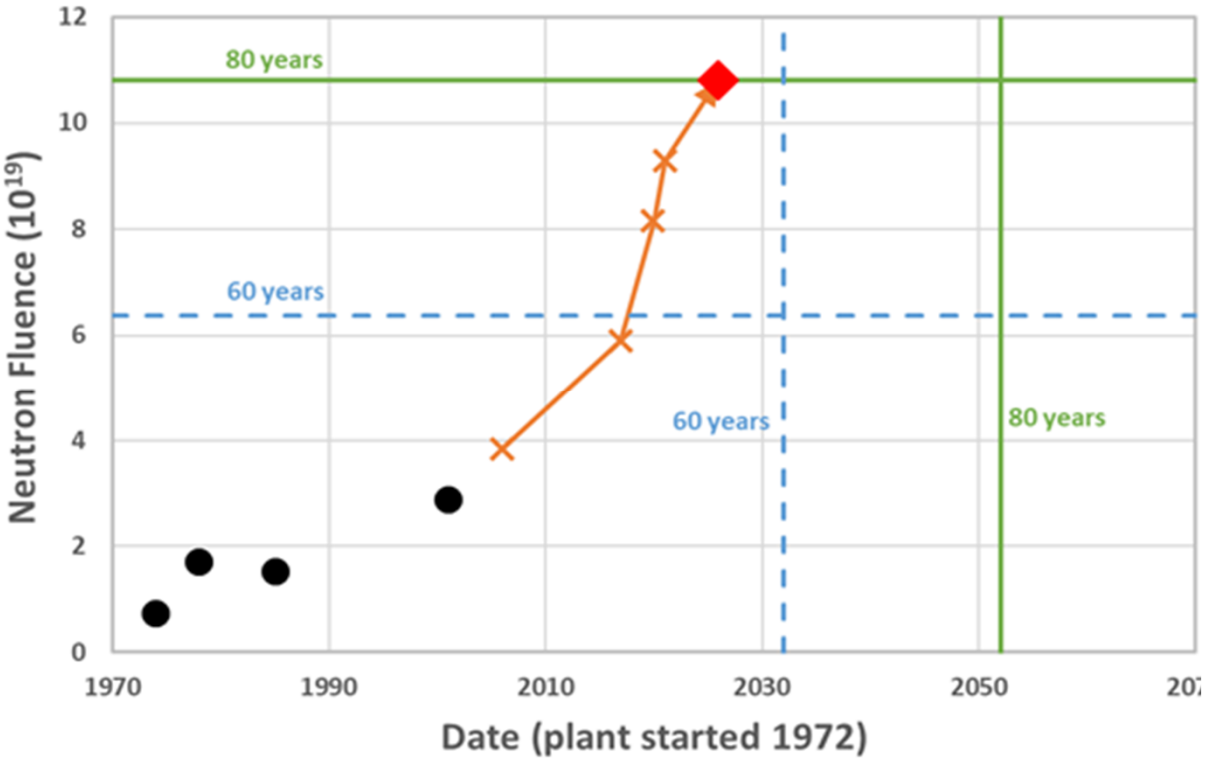
- Aging Management Program XI.M31, "Reactor Vessel Material Surveillance"
  - Continues reliance on Appendix H program using ASTM E185-82
  - GALL Report (NUREG-1801, Rev. 1) for license renewal (40 to 60 years)
    - "shall have at least one capsule with a projected neutron fluence equal to or exceeding the 60-year peak reactor vessel wall neutron fluence prior to the end of the period of extended operation"
    - Describes use of reconstituted specimens and use of operating restrictions (neutron flux, spectrum, irradiation temperature, etc.)
  - GALL-SLR Report (NUREG-2191) for subsequent license renewal (60 to 80 years)
    - "withdrawal and testing of at least one capsule . . . with a neutron fluence of the capsule between one and two times the peak neutron fluence of interest at the end of the subsequent period of extended operation" – or data from a prior tested capsule
    - Specifies – "it is not acceptable to redirect or postpone the withdrawal and testing of that capsule to achieve a higher neutron fluence that meets the neutron fluence criterion for the subsequent period of extended operation"

# License Renewal in Practice

- Licensees have changed capsule withdrawal schedules prior to application for license renewal or subsequent license renewal
  - Change is evaluated under current approach of "conformance verification"
- (Updated) current licensing basis surveillance program for license renewal/subsequent license renewal is then consistent with the program in GALL/GALL-SLR

# Issue – Appendix H

## Performance Monitoring



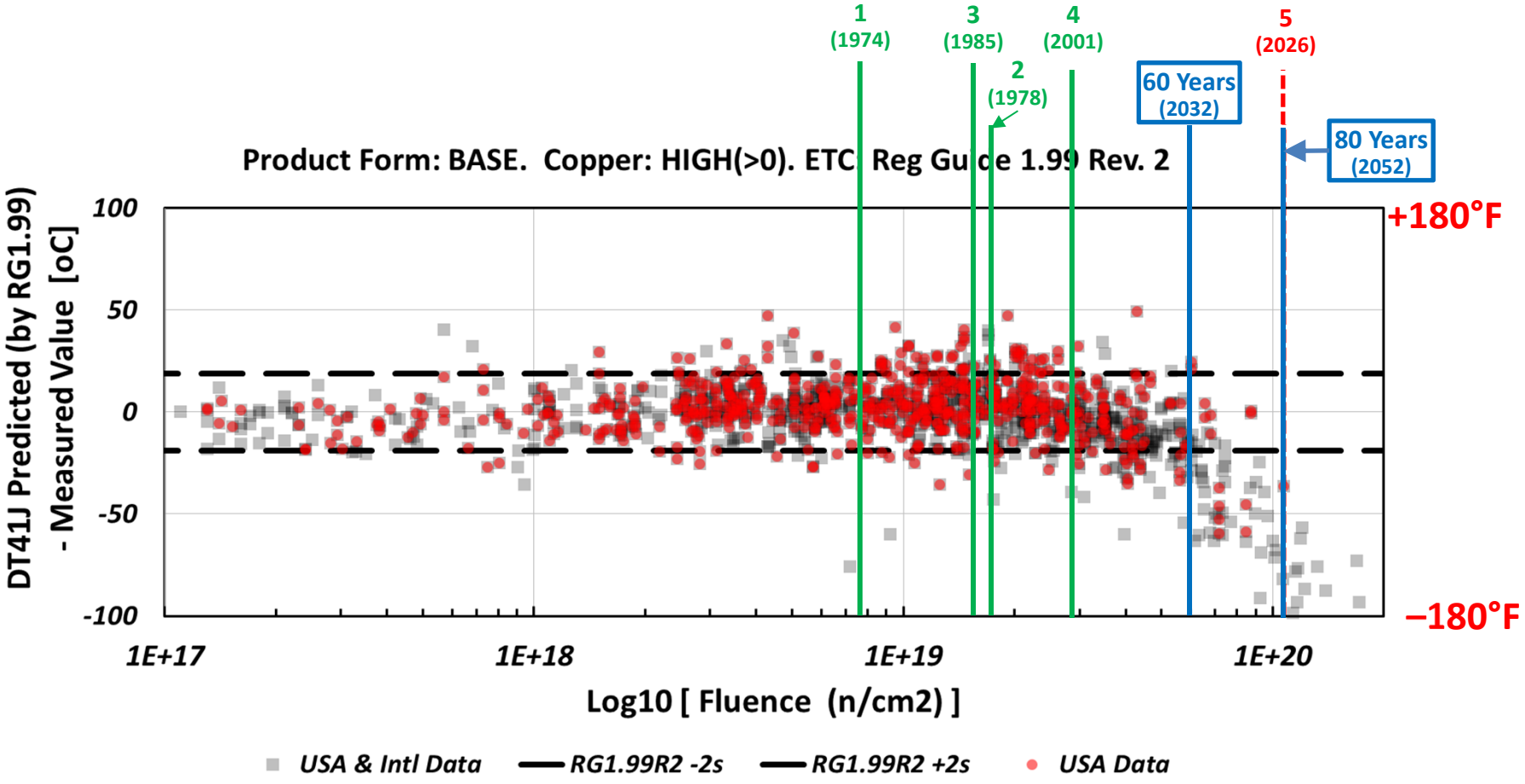

Many licensees have delayed capsules (time and/or fluence), some recent examples:

| Plant | Capsule # | # of times delayed |
|---|---|---|
| Turkey Point | 5 | 4 |
| Robinson | 5 | 2 |
| Surry U1 | 5 | 2 |
| Surry U2 | 5 | 2 |
| North Anna U1 | 4 | 2 |
| North Anna U2 | 4 | 2 |
| St. Lucie U2 | 4 | 1 |
| Point Beach | 5 | 1 |

Capsule withdrawal schedule changes include delays in both time and/or fluence

Not all plants have delayed withdrawal of capsules

# Potential Impact of Issue



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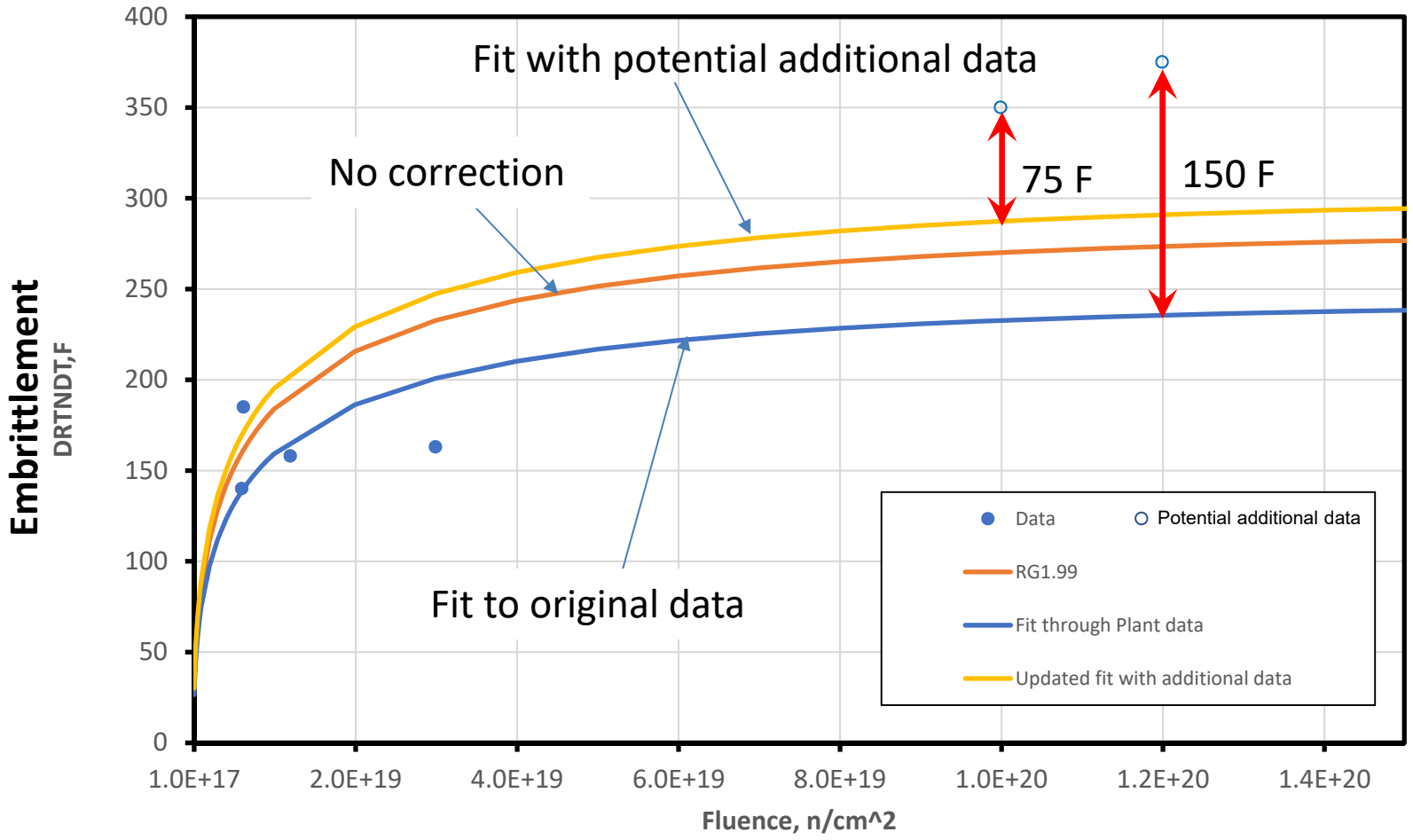
# Potential Impact of Issue



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# Risk-informed Analysis

- Considered combined effects of surveillance and embrittlement predictions
- Leveraged 5 principles of risk-informed decision making
- Targeted sample of plant data used, but much plant specific information not available




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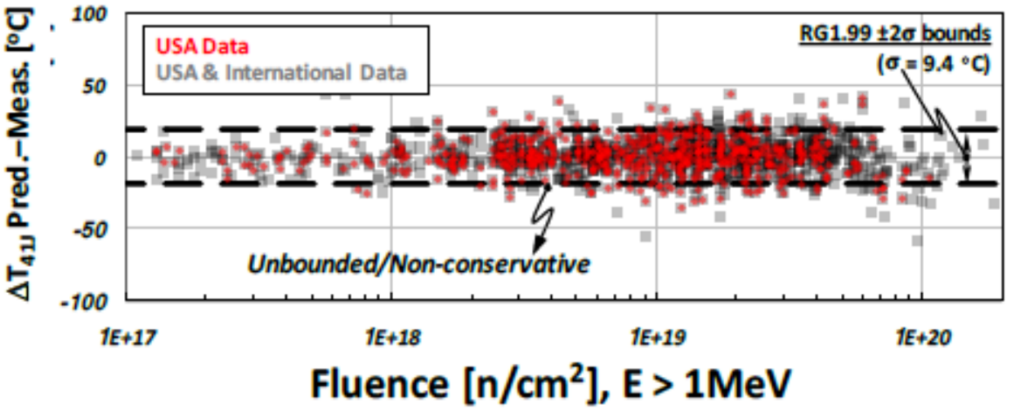
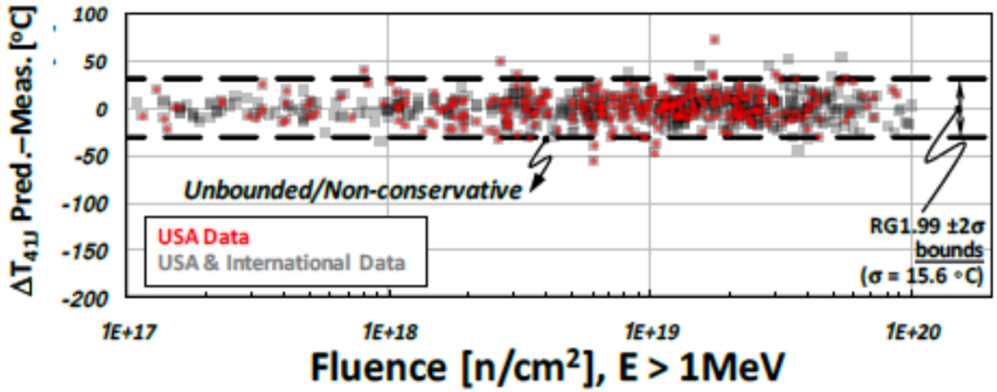
# Analysis Assumptions

- Comparisons based on ASTM E900-15 ETC
- The NRC staff found that the ASTM E900-15 ETC provided the most accurate characterization of this database*

**Plates and Forgings**




**Welds**




*"Basis for a Potential Alternative to Revision 2 of Regulatory Guide 1.99," TLR-RES/DE/CIB-2020-11, ML20345A003

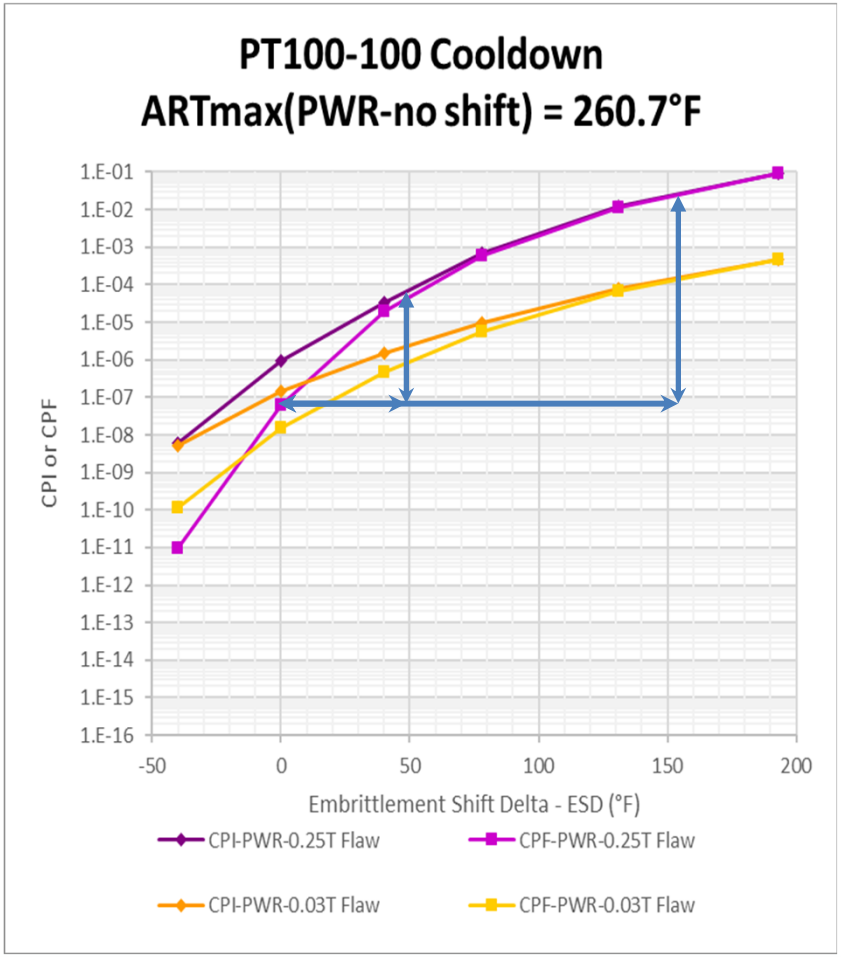
# Analysis Assumptions – Fleet Impact Study

- A targeted sample of 21 plants
- Emphasis on high fluence plants, with a few low Cu plants and BWRs to round out
- Determined changes in adjusted reference temperature resulting from switching ETCs – “embrittlement shift delta” (ESD)
- Results used to benchmark ESD range of risk analysis

# Results – Fleet Impact Study

- There is a tendency for material reference temperatures to increase when switching from RG 1.99 to ASTM E900-15.
- Base materials are more likely to see increases in reference temperatures than weld materials.
- Only a handful of plant limiting materials will have ESDs > 50 °F, and these tend to be at fluences ~$6x10^{19}$ $n/cm^2$ .
- Range of ESDs assumed in risk study bounds fleet impact findings.

# Risk of Failure




ESD represents the underprediction of $\Delta RT_{NDT}$

## Large Uncertainties:

- Unknown frequency of transient
- Actual plant fluence variations
- Are these analyses bounding?
  - Unknown plant-specific considerations
- How much protection do administrative and other operational limits provide against violating the PT limit?

"RG 1.99 Revision 2 Update FAVOR Scoping Study," May 6, 2021, TLR RES/DE/CIB-2020-09, Rev. 1, ML21126A326

# Through-Wall Crack Frequency Results

| Transient Type | Shallow Flaw | 1/4T Flaw | Comment |
|---|---|---|---|
| **BWR P-T Limit Cooldowns** | CPF ≤ $1x10^{-6}$ for all ESDs | CPF ≥ $1x10^{-6}$ for ESD > 40 °F | BWRs must cooldown on saturation curve, so cooldown on licensed limits not plausible. |
| **BWR Saturation Cooldown** | CPF ≤ $1x10^{-6}$ for all ESDs | CPF ≤ $1x10^{-6}$ for all ESDs | |
| **BWR Leak Test, Cooldown rate ≤ 50 °F/hour** | CPF ≤ $1x10^{-6}$ for all ESDs | CPF ≥ $1x10^{-6}$ for ESD > 100 °F | Additional information is desired to determine if high cooldown rates are possible, or ASME Code action will be pursued to prohibit. |
| **BWR Leak Test, Cooldown rate > 50 °F/hour** | CPF ≤ $1x10^{-6}$ for all ESDs | CPF ≥ $1x10^{-6}$ for ESD > 100 °F | |
| **PWR P-T Limit Cooldowns** | CPF >$1x10^{-6}$ for ESDs ≥ 50 °F | CPF > $1x10^{-6}$ for ESD ≥ 20 °F | Additional information on event frequencies is desired to confirm TWCF< $1x10^{-6}$ /year. |
| **PWR Cooldown, Actual Transients** | CPF < $1x10^{-6}$ for most transients | n/a | |

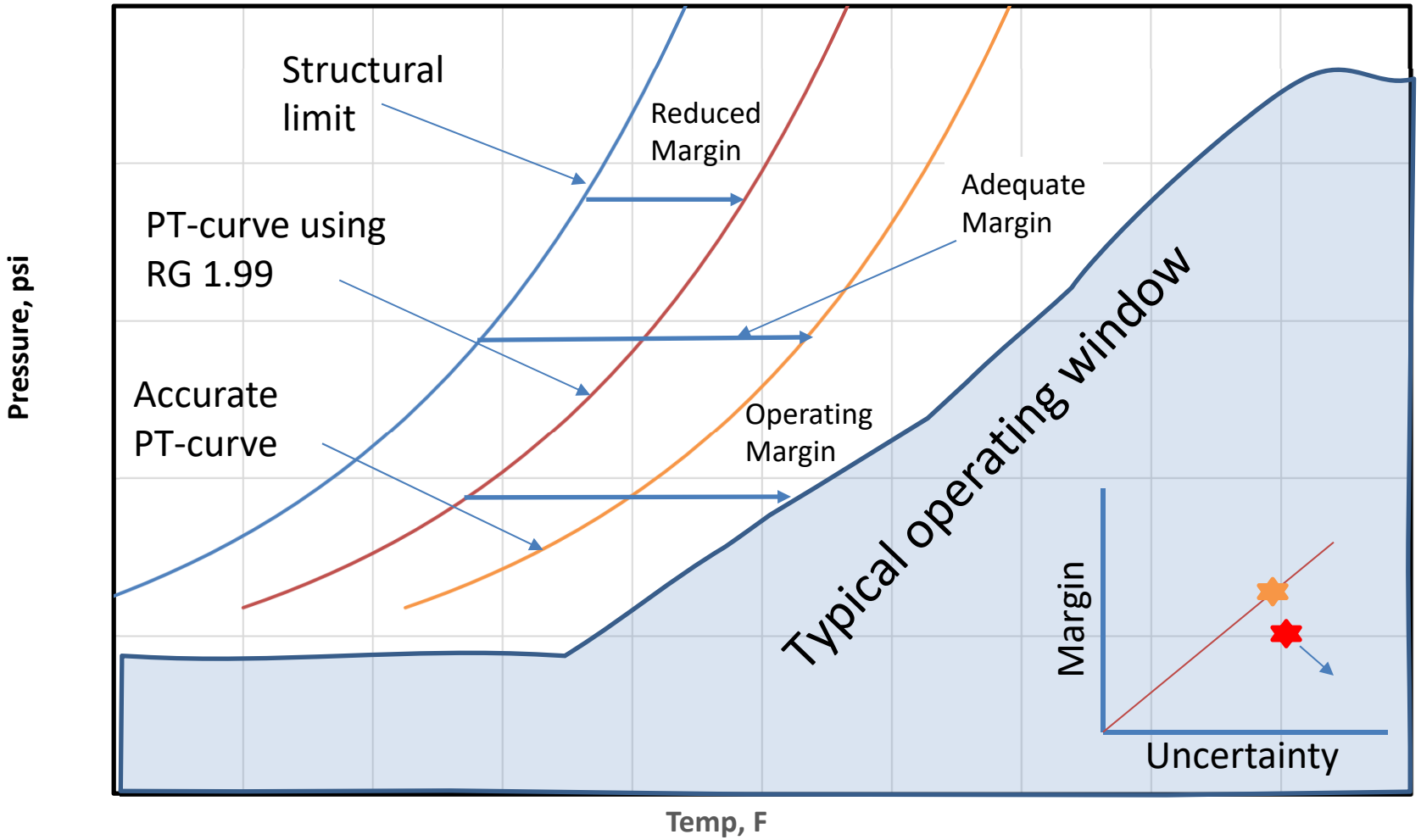
# Pressurized Thermal Shock Considerations

- 10 CFR 50.61 uses ETC from RG 1.99
- $RT_{PTS}$ from 10 CFR 50.61 might be impacted
  - Limits of 270 °F for plates, forgings, and axial weld materials, and 300 °F for circumferential weld materials
- However, through-wall crack frequency calculated with corrected embrittlement less than $1x10^{-6}$ for all cases investigated

# Safety Margins

- Uncertainties in risk calculations are high and increasing with time
- Even though the risk appears low, resolving these issues will help maintain the fundamental safety principles that are the basis of plant design and operation
- Safety margins, as provided by regulations and current license bases, provide reasonable assurance against brittle fracture

# Safety Margins Illustration




Uncertainties increasing due to lack of surveillance, but margin is less due to embrittlement underprediction

# Performance Monitoring

- Performance monitoring ensures
  - Analysis results remain valid with time
  - No unexpected (or unmodelled) adverse safety issue occurs
- Delaying capsule withdrawal for an extended period with the possibility of no future data represents a lack of performance monitoring

# Analysis Summary

- With the current state of knowledge, a generalized analysis suggests the overall risk of brittle fracture is low
- The uncertainty in these results is high and increases with time
  - Plant specific details not considered
- Under certain conditions, safety margins are impacted and are decreasing as uncertainty increases
- Delaying capsules at high fluence represents a lack of sufficient performance monitoring
- Issues are plants with fluences > $6x10^{19}$ $n/cm^2$

# Who is Impacted?

- Embrittlement Underprediction

| Percentage of Fleet Surpassing Fluence Levels | | | Percentage of PWRs Surpassing Fluence Levels | |
|---|---|---|---|---|
| Year\Fluence | 6 x $10^{19}$ n/cm$^2$ | 8 x $10^{19}$ n/cm$^2$ | 6 x $10^{19}$ n/cm$^2$ | 8 x $10^{19}$ n/cm$^2$ |
| 60 years | 6% | 0% | 9% | 0% |
| 80 years | 22% | 10% | 34% | 15% |

  - Plant specific details (e.g., limiting material, etc.) may contribute to which plants are impacted
  - More work is needed to determine which plants are impacted

- Lack of Surveillance Data
  - Any plant renewing license that chooses to delay last capsule

# Staff Goals

- Currently, regulations are sufficient for reasonable assurance of adequate protection against brittle fracture of vessel
- Staff wants to ensure continued reasonable assurance in long-term operation
  - Provide remedies for the identified issues with RPV surveillance requirements and embrittlement predictions, on a risk-informed, performance basis
- Do not impact those plants that are not adversely affected by the issues
  - Plant-specific surveillance data that covers end of license fluence level
  - Projected fluence at end of license < ~3 x $10^{19}$ n/$cm^2$

# Options to Meet Goal

- Plant-specific action
- Focused regulatory action
- Generic communication
- No action

# Discussion Topics

- Is the staff's approach to determine the safety impact of the surveillance and embrittlement issues appropriate?
- What other options could be considered to address these issues?
- Are there other potential adverse impacts to plant operations (e.g., unnecessary updates to PT limits) that should be considered?
- Is now the right time to pursue these issues?

*Reminder: The NRC is not actively soliciting comments towards regulatory decisions at this meeting.*

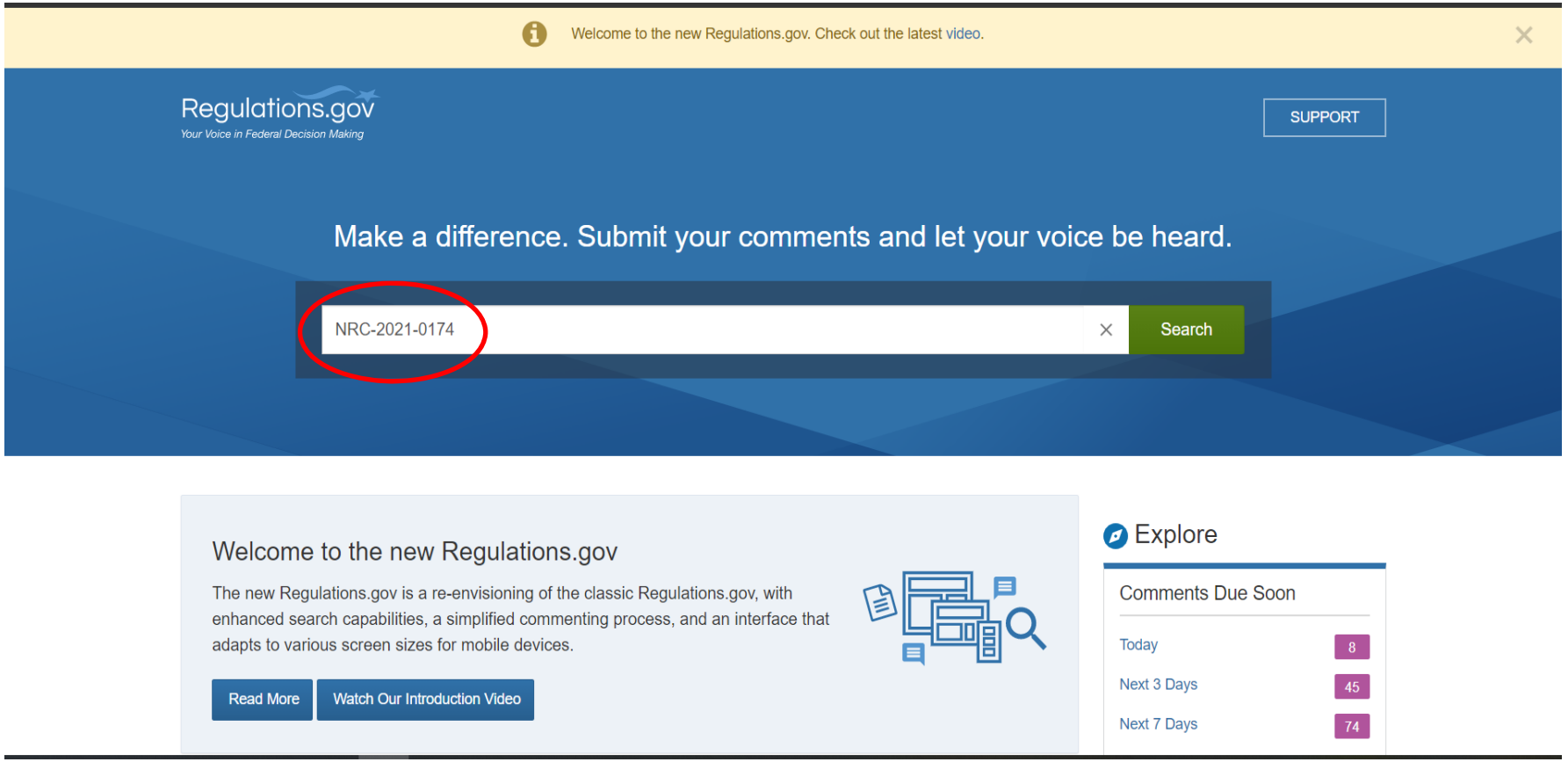
# Summary

- High confidence that currently operating plants remain safe, and recent licensing actions remain valid
- Issue will eventually (after about 10 years) impact the staff confidence in the integrity of the reactor pressure vessel in long-term operation, i.e., both safety margins and performance monitoring may be impacted
- Further work is needed to determine which plants are impacted by this issue
- Proactively ensure continued reasonable assurance though a risk-informed, performance-based solution
  - Staff is considering options – desires focused solution to only those conditions adversely impacted by this issue

# Public Presentations

# Discussion

# Where to Find Information




**Search for docket ID NRC-2021-0174**



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# How did we do?

- NRC Public Meeting Feedback Form
  - Link to the form is available at the public meeting website under the meeting notice

- Or use this QR code:

# NRC Contacts


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# Thank You