An Evaluation of the Status of Diablo Canyon Unit 1 with Respect to Reactor Pressure Vessel Condition Monitoring and Prediction

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PEAI

DCISC Public Meeting

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Engineering
Associates
Inc.

BACKGROUND

Administrative. Embrittlement measurement & prediction. RPV integrity management.

NRC REQUIREMENTS & GUIDANCE

For monitoring & embrittlement prediction. For structural integrity

EVALUATION OF DIABLO CANYON UNIT 1

Surveillance program. NRC's required analysis. Supplemental analysis.

ANSWERS TO PUBLIC'S CONCERNS

Administrative
Documentation

- **Objectives**
  - Address concerns raised by SLOMFP, FOE, & Mr. B. Severance
  - Independent evaluation of Unit 1 RPV condition
- **Two reports**
  - Part 1 addresses public's concerns
  - Part 2 evaluates the embrittlement of the Diablo Canyon Unit 1 reactor pressure vessel (RPV)
- **Reports reviewed by**
  - DCISC consultants
  - DCISC members individually
  - PG&E to check for omissions & errors of fact only

Reviews did not affect report conclusions

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Part 1: Addressing Public Concerns

Consultant's report to the Diablo Canyon Independent Safety Committee

Submitted by
Mark Kirk
Phoenix Engineering Associates Inc.
Unity, New Hampshire, USA

Date: 26 January 2024

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https://www.dcisc.org/

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Letter Transmitting Two Reports by Dr. Mark Kirk to the DCISC. (PDF)

Report Part 1: "Addressing Public Concerns." (PDF)

Report Part 2: "Evaluation of Diablo Canyon Unit 1 Embrittlement." (PDF)

Administrative

Where are public's concerns addressed?

1. BACKGROUND
2. NRC REQUIREMENTS AND GUIDANCE
3. EVALUATION OF UNIT 1
4. ANSWERS TO PUBLIC'S CONCERNS

Concerns

Continued Operating Safety of Unit 1

- Data credibility
- Surveillance capsule withdrawal plan
- Use of similar ("sister plant") data
- RPV beltline inspections
- Alternate testing methods (nano indentation)
- Alternative Charpy analysis method
- Aspects of the RPV analysis methodology
- Deficient materials

1. Background

Part 1 Report
Section 2.1, 2.2, and 2.3

Key terms

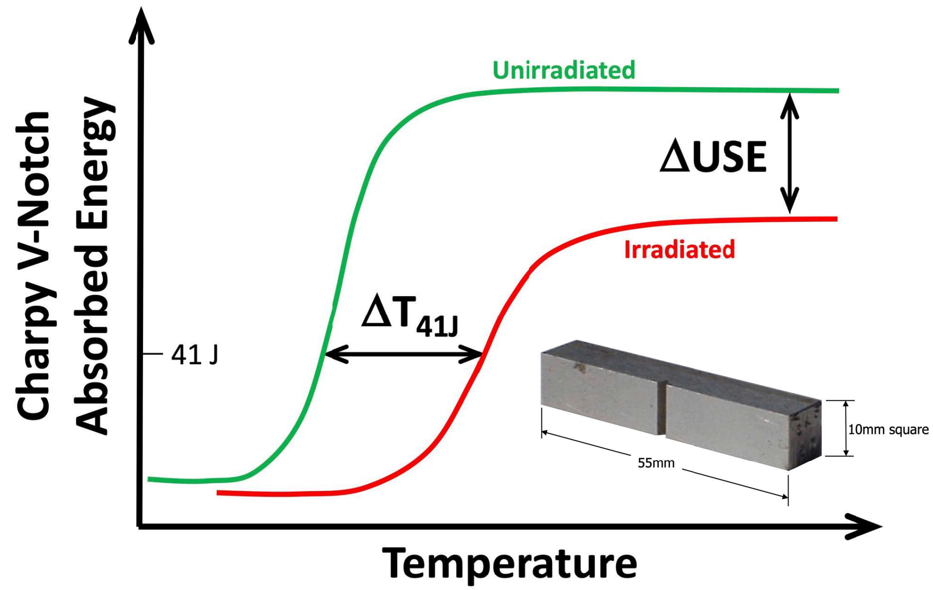
- Embrittlement
- Fracture toughness
- Surveillance program

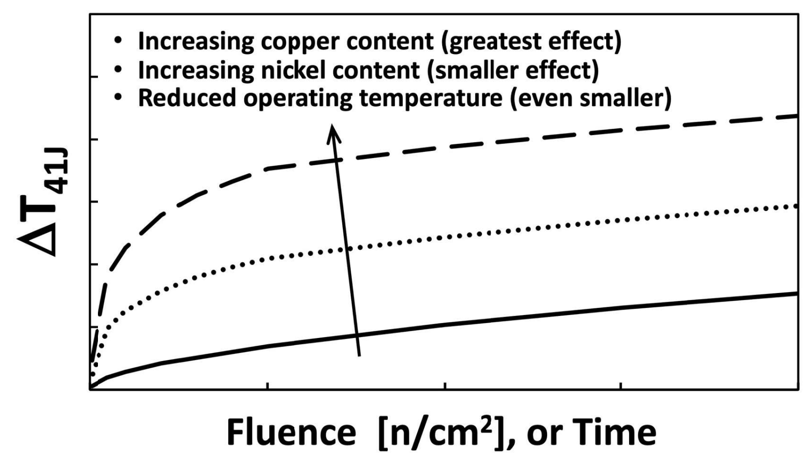
Embrittlement forecasting

Assessment of RPV Integrity (Screening Criteria)

Embrittlement

- **Embrittlement reduces the steel's resistance to failure**
  - Embrittlement changes RPV steel at a microstructural level
    - Embrittlement does not cause cracking
    - Embrittlement does reduce energy absorbed by the steel before fracture
      - Increase in Charpy transition temperature (T_{41J})
      - Reduction in Charpy upper shelf energy (USE)
- **Embrittlement occurs progressively over the plant's operating life**
  - Faster early in life
  - Slower as plants age
  - Steel composition affects embrittlement rate

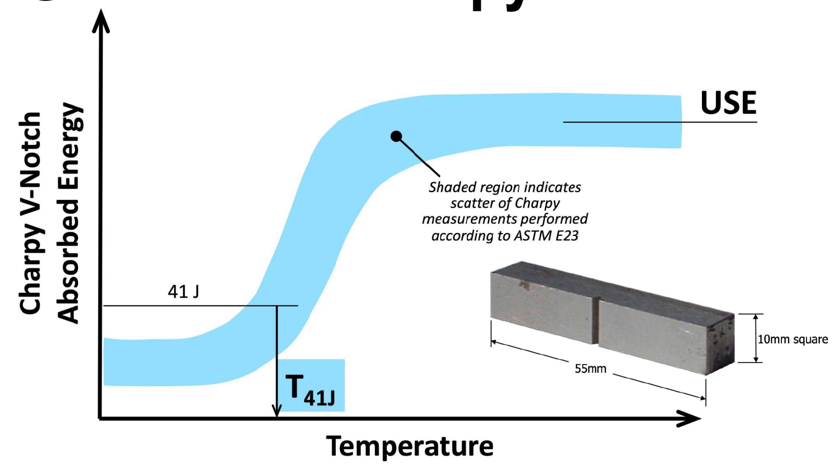




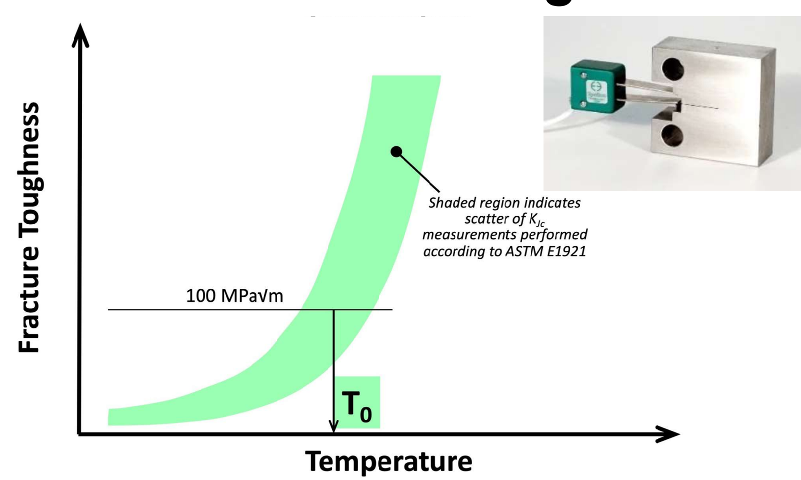
Fracture Toughness

- Fracture toughness needed to assess plant structural integrity (ASME, NRC)
- Charpy specimens have a machined notch, while fracture toughness is measured using a specimen with a sharp fatigue crack
- Charpy T_{41J} data correlates well with fracture toughness

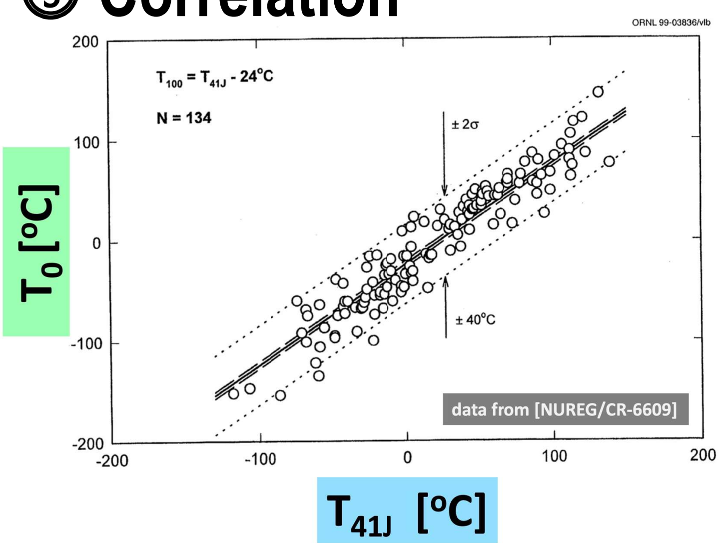
① Measure Charpy



② Want fracture toughness



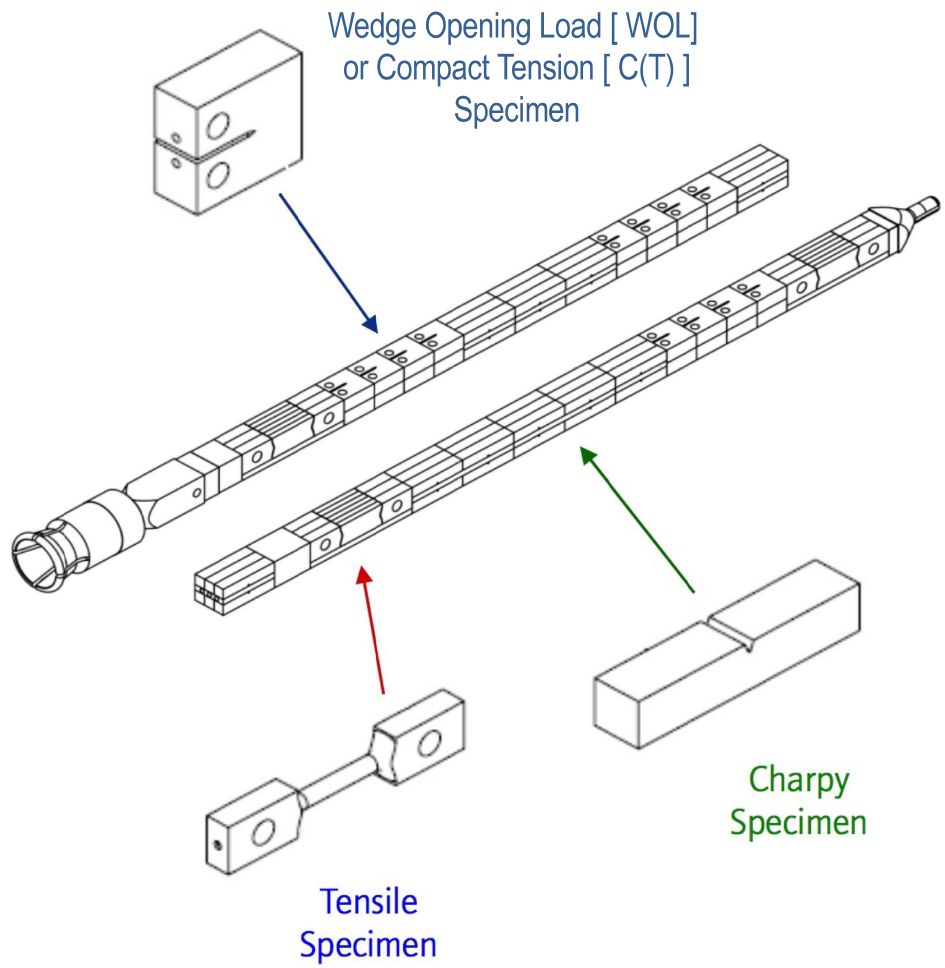
③ Correlation



Surveillance Program

1

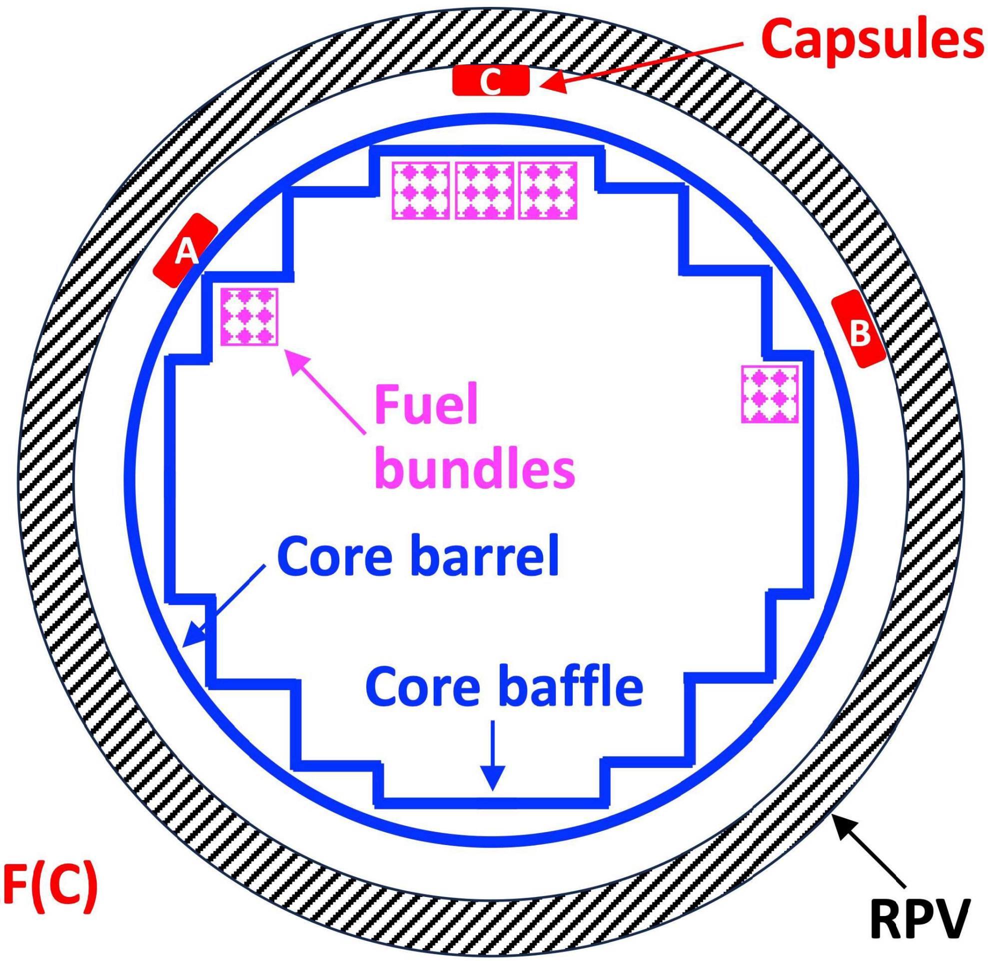
Surveillance capsules are loaded with specimens.



Surveillance Program

2

Surveillance capsules are placed inside the RPV. Capsule distance from fuel determines capsule lead factor (LF).

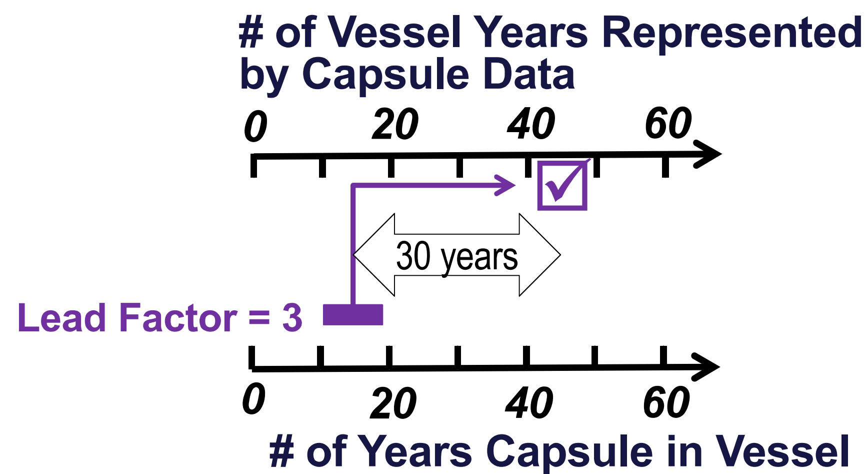


LF(A) > LF(B) > LF(C)

Surveillance Program

3

Higher lead factors provide greater advance information on RPV embrittlement.



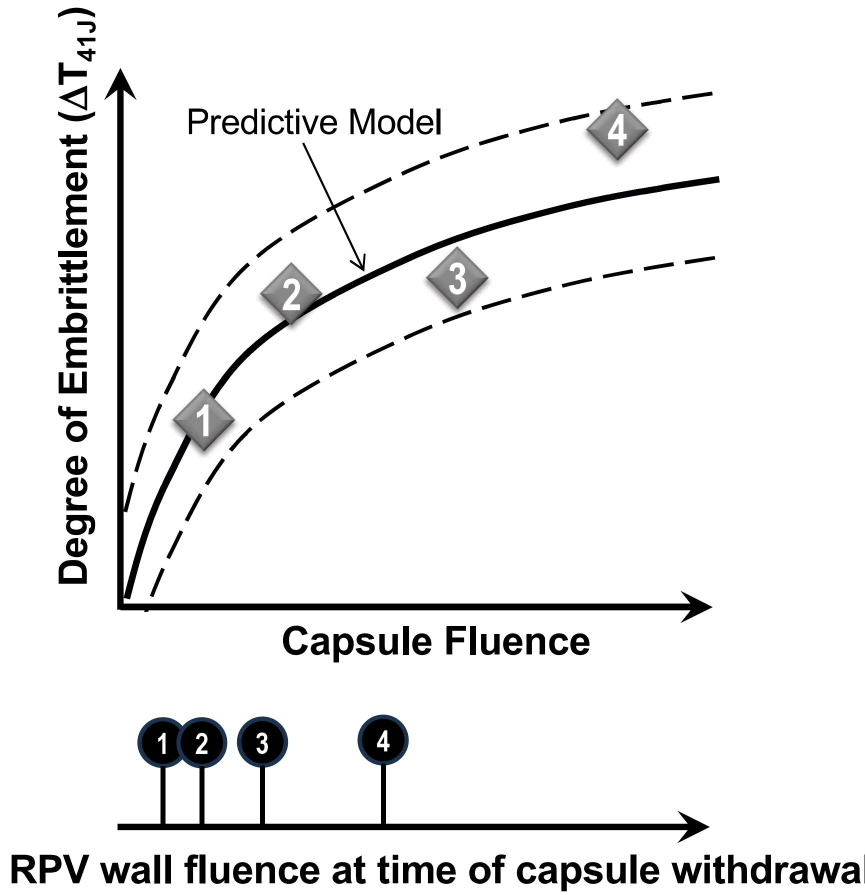
- Specimens in capsules experience more embrittlement than the RPV
- Specimens represent the RPV condition years into the future
- Long intervals between capsule withdrawals are acceptable

Surveillance Program & Embrittlement Forecasting

4

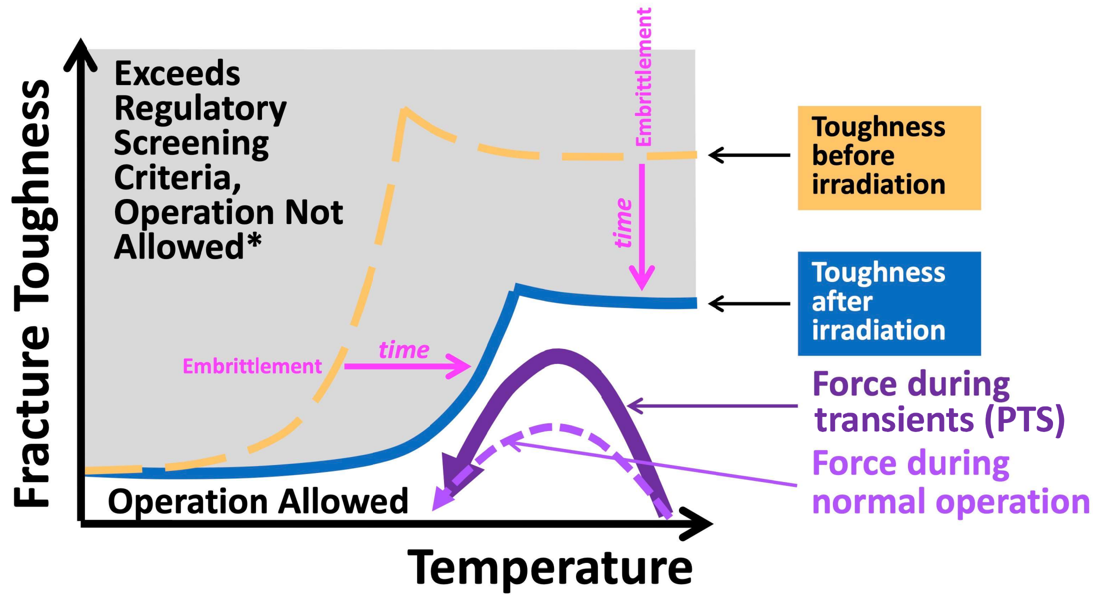
Capsule data & predictive model forecast future RPV embrittlement

- Both surveillance data and T_{41J} model are used to forecast future embrittlement
- After 1st capsule Unit 1 is not extrapolating to higher fluences than in the RPV



Assessment of RPV Integrity

- Screening criteria ensure material resistance to fracture exceeds structural driving force
  - Pressure-Temperature operating limits
    - Per ASME Code (shown)
  - Pressurized thermal shock
    - RT_{PTS} < 132 °C for axial welds
  - Upper shelf energy
    - USE > 68 Joules



* - not allowed without additional justification, plant modification, or testing

<u>Screening criteria</u>

- Are not a failure condition
- Are conservatively established at a low failure probability
- Indicate need for additional actions to demonstrate adequate margins
- NRC requires action plan 3 years in advance

2. NRC Requirements & Guidance

Part 1 Report
Section 2.4 and 2.5

Embrittlement monitoring & prediction

- 10CFR 50 App. H
- RG 1.99 Rev 2
- NUREG-1801

RPV Integrity

- 10CFR 50 App. G
- 10 CFR 50.61
- 10 CFR 50.61a

Embrittlement Monitoring

- NRC 10CFR50 Appendix H sets requirements for surveillance during the initial 40-year license. Incorporates ASTM standard by reference
  - Unit 1 licensed to ASTM E185-70, which requires 3 capsules
  - Same requirements for 50.61 and 50.61a
- NUREG-1801: NRC recommendations for license renewal (beyond 40 years)
  - Compliance with E185-82 recommended
  - More capsules needed during license renewal if
    - Highest fluence is < 60-year fluence, **<u>or</u>**
    - The ΔT_{41J} predicted at 60-years increases the number of capsules per ASTM E185-82 table

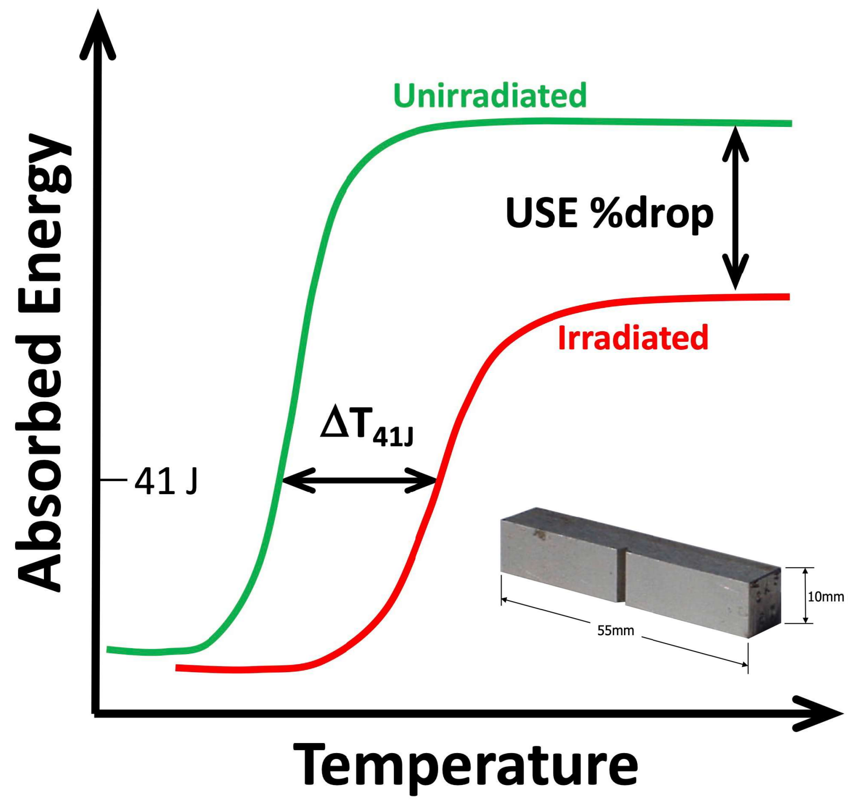
<u>ASTM E185-82</u>

Predicted ΔT_{41J} on Vessel ID at EOL Fluence [°C]	# of Capsules Required
< 56 °C	3
56 - 111 °C	4
> 111 °C	5

- Last capsule withdrawn between 1x & 2x EOL fluence
- Monitoring of limiting plate and limiting weld required

Embrittlement Prediction

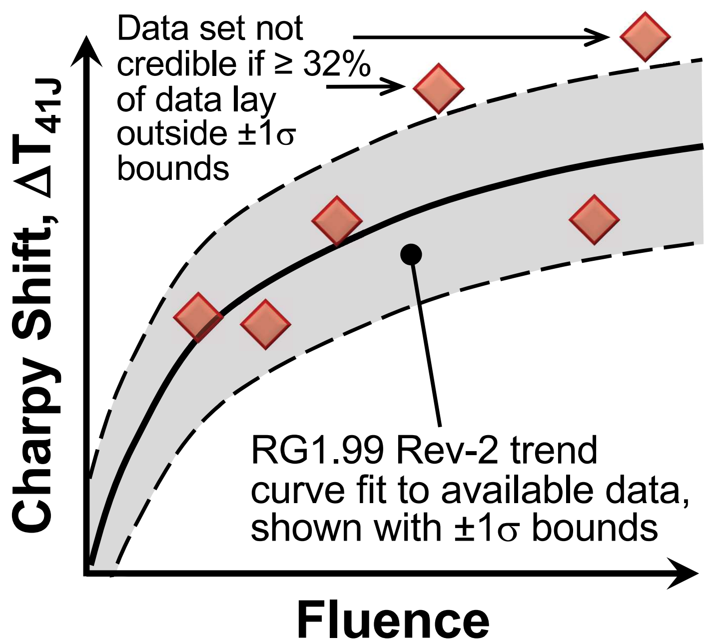
- Reg Guide 1.99 Rev 2 provides NRC's embrittlement prediction models
- Formulas estimate ΔT_{41J} and %drop of USE
- The ΔT_{41J} formula (reference temperature shift)
  - Is required by the PTS rule (10CFR 50.61)
  - Is used by custom for P-T limits (ASME Code)
  - Alternate PTS rule (10CFR50.61a) requires a different model
- Guidance and requirements for what data to use are provided ("credibility") … next slides



Embrittlement Prediction (Credibility)

- Guidance and requirements for what data to use are provided ("credibility")

What is credibility?	Credible data follows RG predicted trends
	Not credible data deviates from these trends (see diagram)*



Total # of ΔT_{41J} Measurements	Max # of data allowed outside shaded bounds in a credible data set
2-3	0
4-6	1
7-9	2
10-12	3

* This is one of RG1.99's five credibility criteria. Unit 1's compliance with this criteria has been questioned.

Embrittlement Prediction (Credibility)

- Guidance and requirements for what data to use are provided ("credibility")

What is credibility?	Credible data follows RG predicted trends
	Not credible data deviates from these trends (see diagram)
How is credibility assessed?	On the whole dataset (all data is either credible or not credible)
	Different rules apply to ΔT_{41J} and USE %drop data
Why does credibility matter?	Credible data are used to adjust ΔT_{41J} and USE %drop predictions
	Not-credible data are modeled more conservatively using RG formulas
What data are used?	For PTS, use of plant-specific data and similar data from "sister plants" is required. P-T limits follow this practice by convention, not requirement.
	For USE assessment, 10CFR50 Appendix G makes no statements about "sister plant" data. Sister plant data is not typically used.

RPV Integrity Screening Criteria (not limits)

Event	NRC Rule	Mechanical Property	Screening Criteria
Pressure-Temperature Limits for Heat Up and Cool Down	10 CFR 50 App G	USE	USE > 68J
		ΔT_{41J}	NRC Incorporates ASME Code: Section XI, Appendix G
Pressurized Thermal Shock	10 CFR 50.61		• Axial Weld RT_{PTS} < 132 ^oC • Plate or Circ Weld RT_{PTS} < 149 ^oC
	10 CFR 50.61a		

Screening criteria
- Are not a failure condition
- Are conservatively established at a low failure probability

TABLE 1—PTS SCREENING CRITERIA

Product form and RT_{MAX-X} Values	RT_{MAX-X} limits [°F] for different vessel wall thicknesses[6] (T_{WALL})		
	$T_{WALL} \le 9.5$ in.	9.5 in. < $T_{WALL} \le 10.5$ in.	10.5 in. < $T_{WALL} \le 11.5$ in.
Axial Weld RT_{MAX-AW}	269	230	222
Plate RT_{MAX-PL}	356	305	293
Forging without underclad cracks RT_{MAX-FO}[7]	356	305	293
Axial Weld and Plate RT_{MAX-AW} + RT_{MAX-PL}	538	476	445
Circumferential Weld RT_{MAX-CW}[8]	312	277	269
Forging with underclad cracks RT_{MAX-FO}[9]	246	241	239

3. Diablo Canyon Unit 1 Evaluation

Part 2 Report
Sections 2, 3, and 4

Surveillance Program
- Program status
- Capsule B deferrals

Evaluation based on NRC requirements

Supplemental Evaluation

Diablo Canyon Unit 1 Surveillance Program

- Program licensed to ASTM E185-70
  - 3 capsules required over 40-years
  - Requirement fulfilled since 2002
- Capsule B deferrals were appropriate because only 3 capsules required in first 40-years
  - NRC 2006 letter (ML061660220) does not make Cap-B testing a condition of recovery of low power testing time into Unit 1's licensed life (September 2021 → November 2024)
- During license renewal (40 → 60 years)
  - NUREG-1801 recommends testing 2 more capsules based on ASTM E185-82 guidance & Unit 1 estimated embrittlement
  - Testing of Cap-B planned for the next refueling outage in 2025 (DCL-23-118, ML23311A154), recommended before 2028
- Palisades "sister plant" data incorporated in 2011, provides 60+ year fluence data

Capsule ID	Lead Factor	Removal Year	Status
S	3.46	1986	Tested
Y	3.44	1992	
V	2.26	2002	
T	3.44	1992	Removed, Not Tested
Z	3.44	1992	
U	1.28	TBD	Standby
X	1.28	TBD	
W	1.28	TBD	
A	1.31	TBD	
B	3.46	2025	Planned Removal & Testing
C	3.46	2004	Removed, Not Tested
D	3.46	2004	

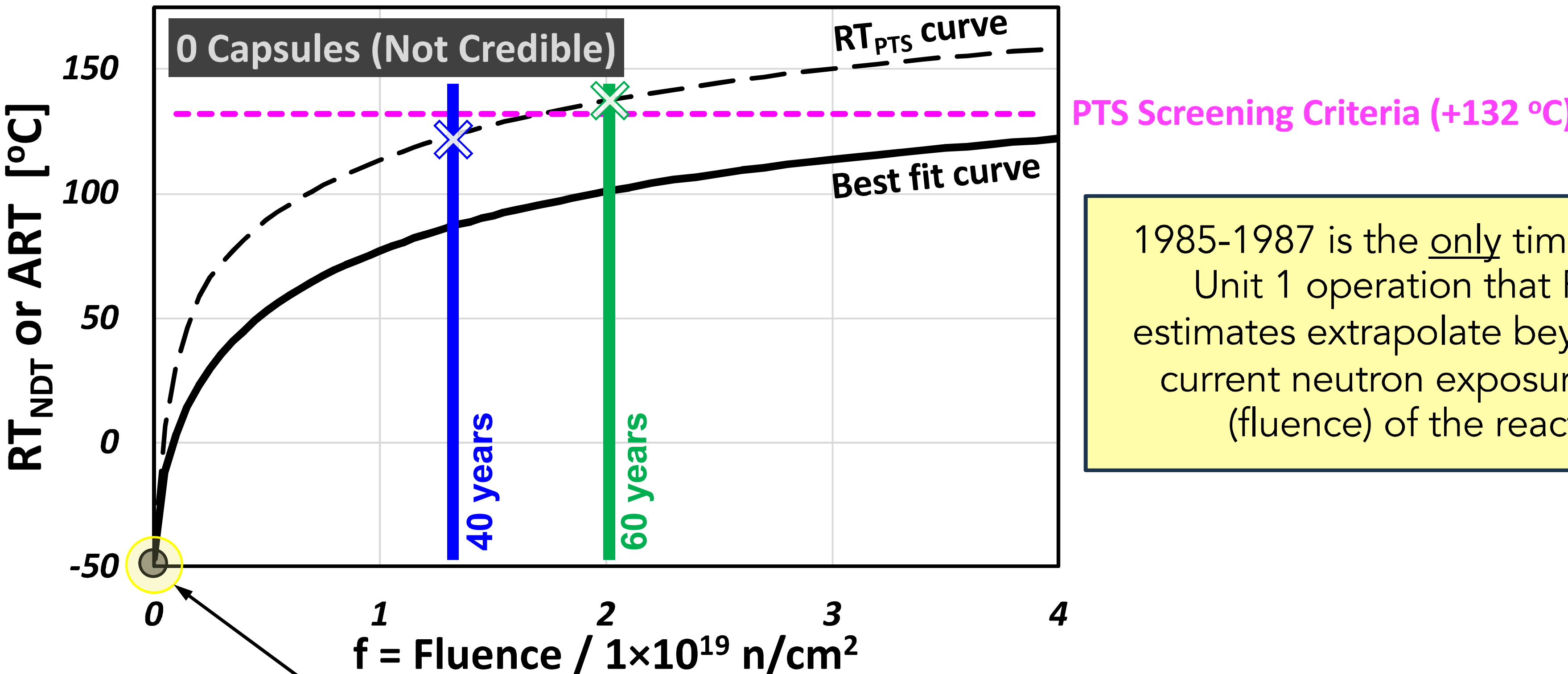
Contains limiting weld

- 3 capsules required for first 40 years, Capsule B deferrals therefore appropriate.
- Capsule B testing recommended by 2028. Capsule B data may alter PTS and USE assessments.

NRC Required PTS Analysis

0 Capsules (1985-1987)

Surveillance data
Estimated RT_{PTS} at 40 years
Estimated RT_{PTS} at 60 years



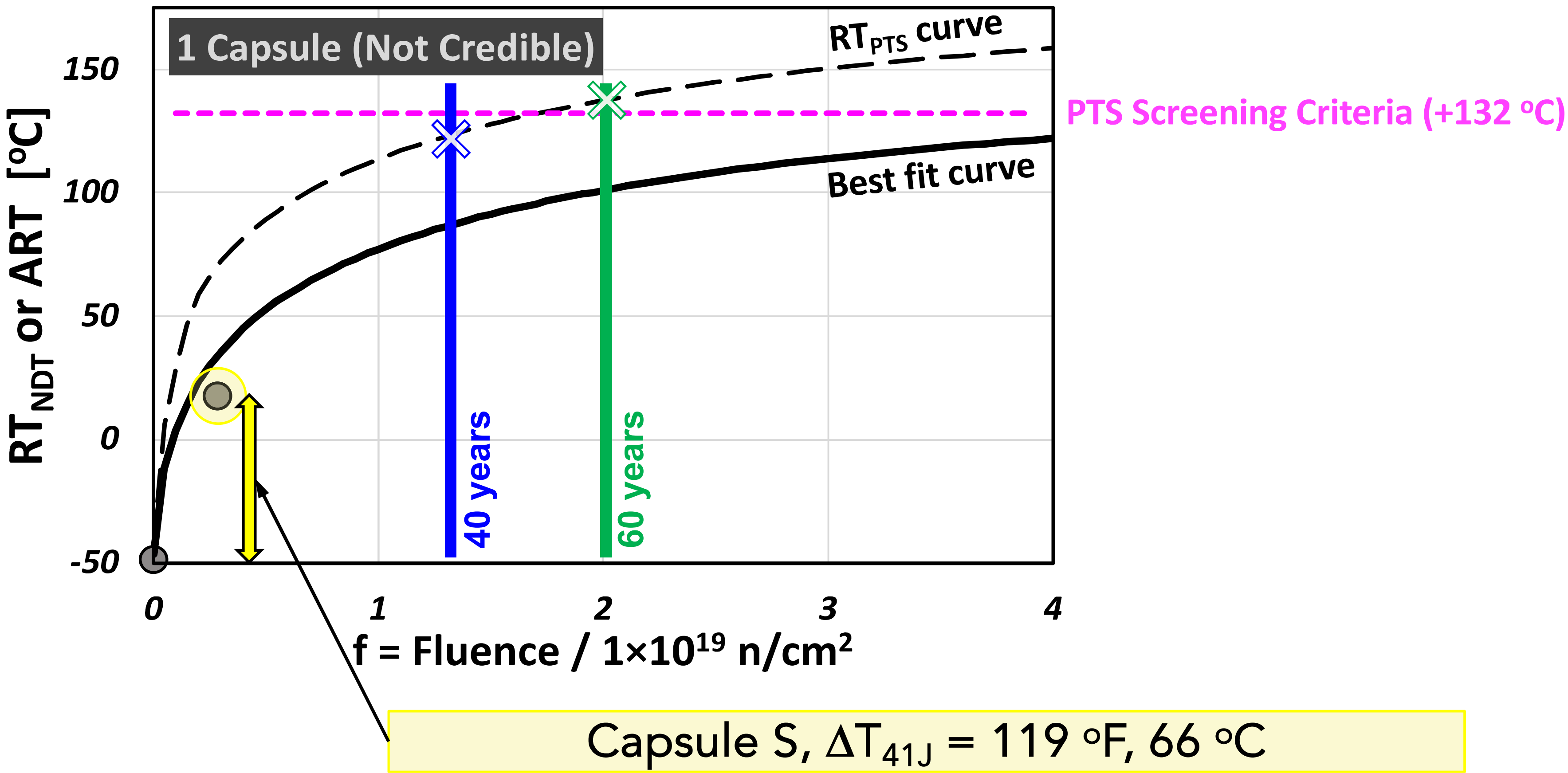
1985-1987 is the only time during Unit 1 operation that RT_{PTS} estimates extrapolate beyond the current neutron exposure level (fluence) of the reactor

Generic Unirradiated RT_{NDT} = -56 °F or -49 °C

NRC Required PTS Analysis

1 Capsule (1987-1993)

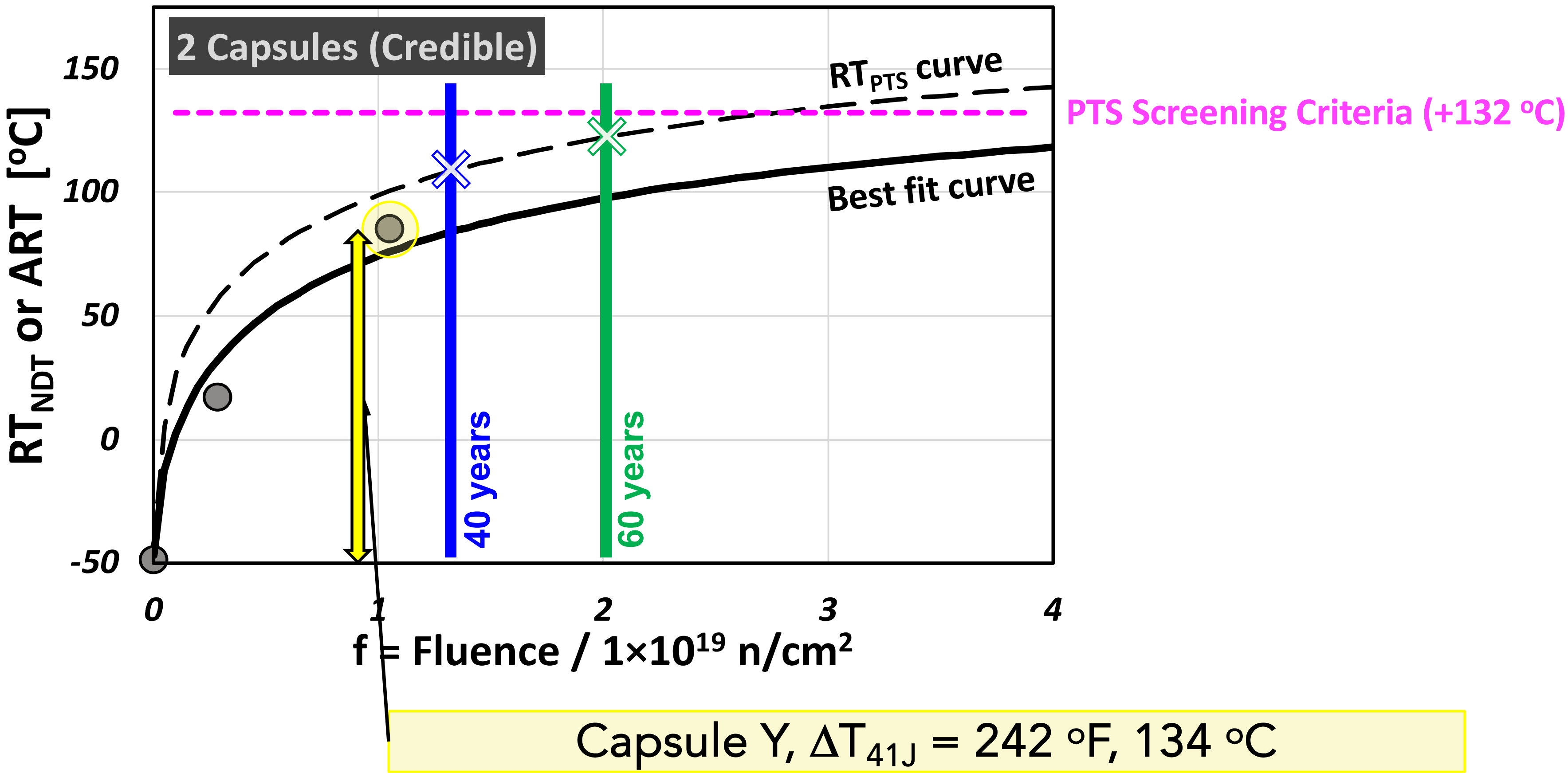
Surveillance data
Estimated RT_{PTS} at 40 years
Estimated RT_{PTS} at 60 years



NRC Required PTS Analysis

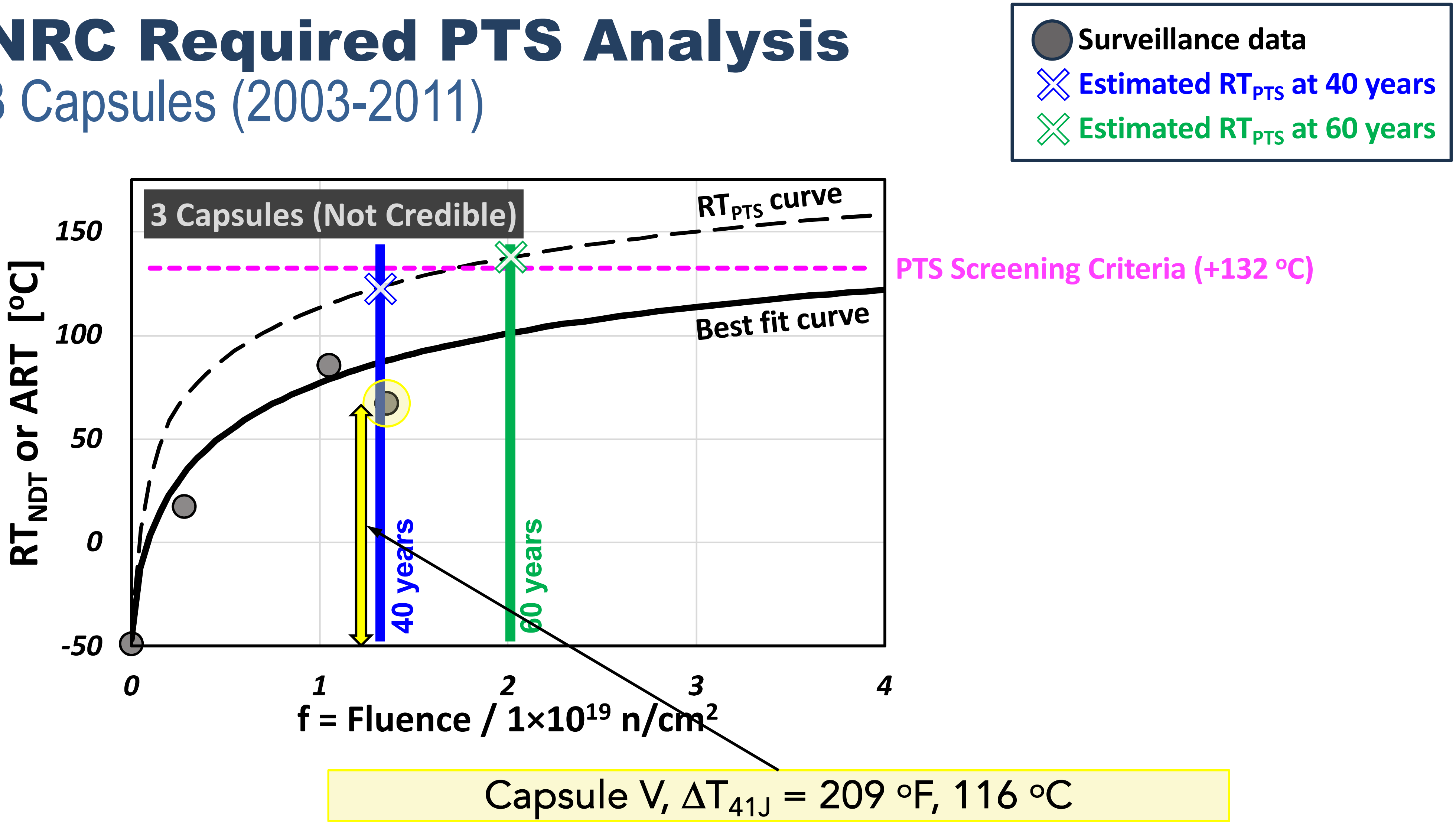
2 Capsules (1993-2003)

Surveillance data
Estimated RT_{PTS} at 40 years
Estimated RT_{PTS} at 60 years



NRC Required PTS Analysis

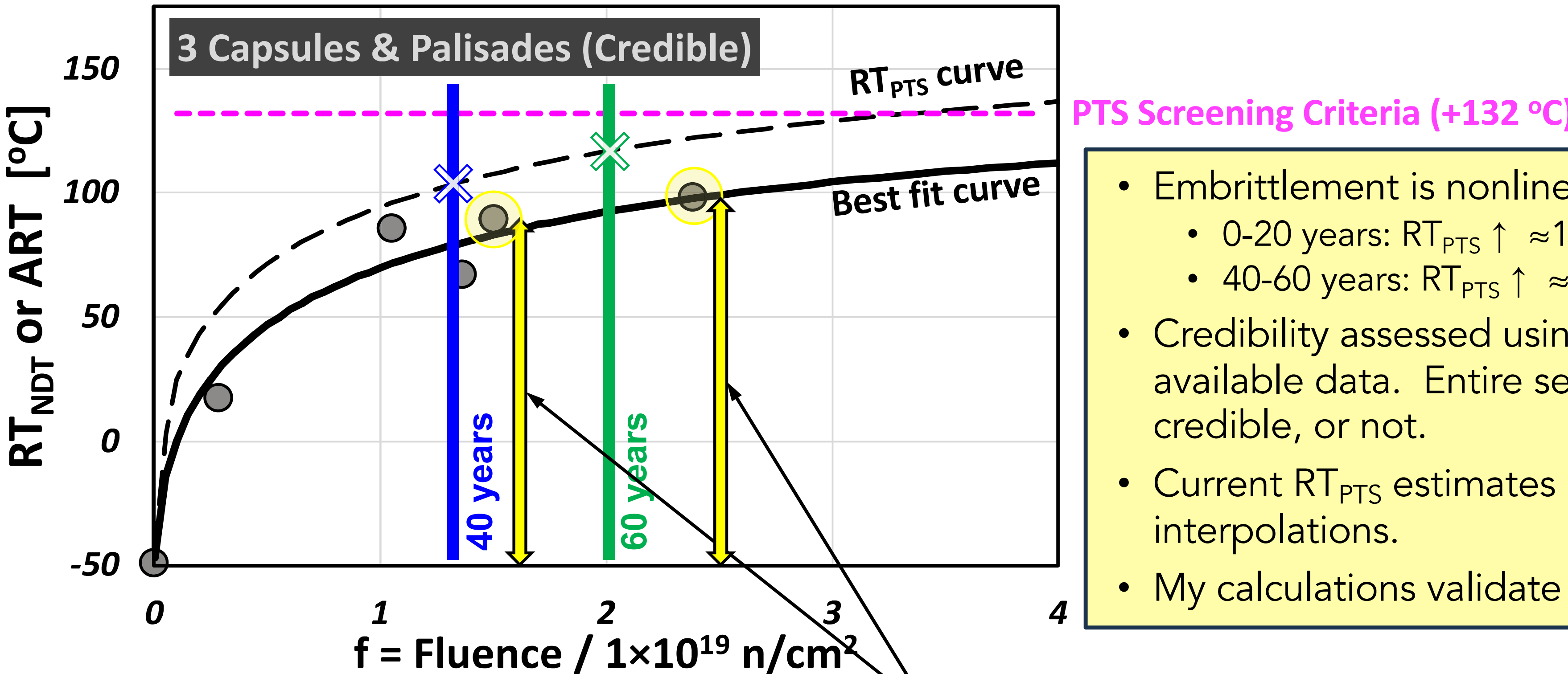
3 Capsules (2003-2011)



NRC Required PTS Analysis

3 Capsules & Palisades (2011-today)

- Surveillance data
- Estimated RT_{PTS} at 40 years
- Estimated RT_{PTS} at 60 years



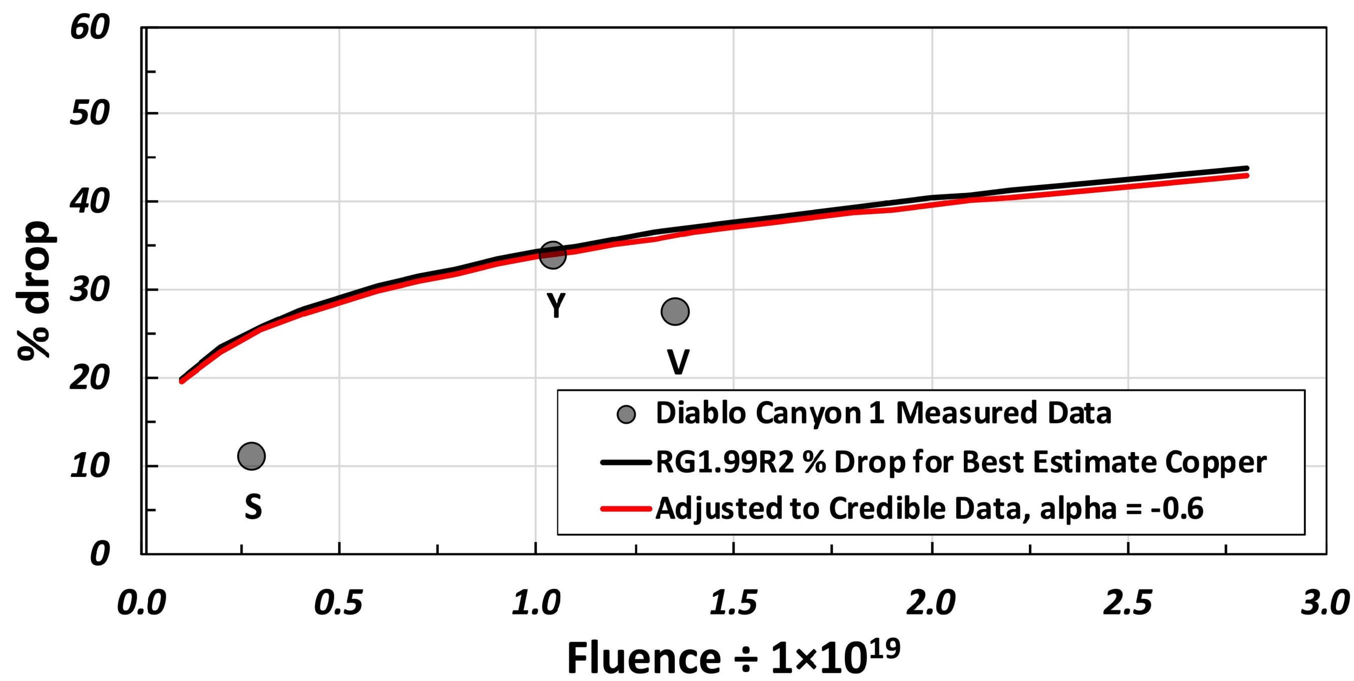
- Embrittlement is nonlinear
  - 0-20 years: RT_{PTS} ↑ ≈100 °C
  - 40-60 years: RT_{PTS} ↑ ≈15 °C
- Credibility assessed using all available data. Entire set is credible, or not.
- Current RT_{PTS} estimates are interpolations.
- My calculations validate PG&E's.

Palisades, ΔT_{41J} = 249 & 265 °F, 138 & 147 °C

NRC Required Analysis for USE

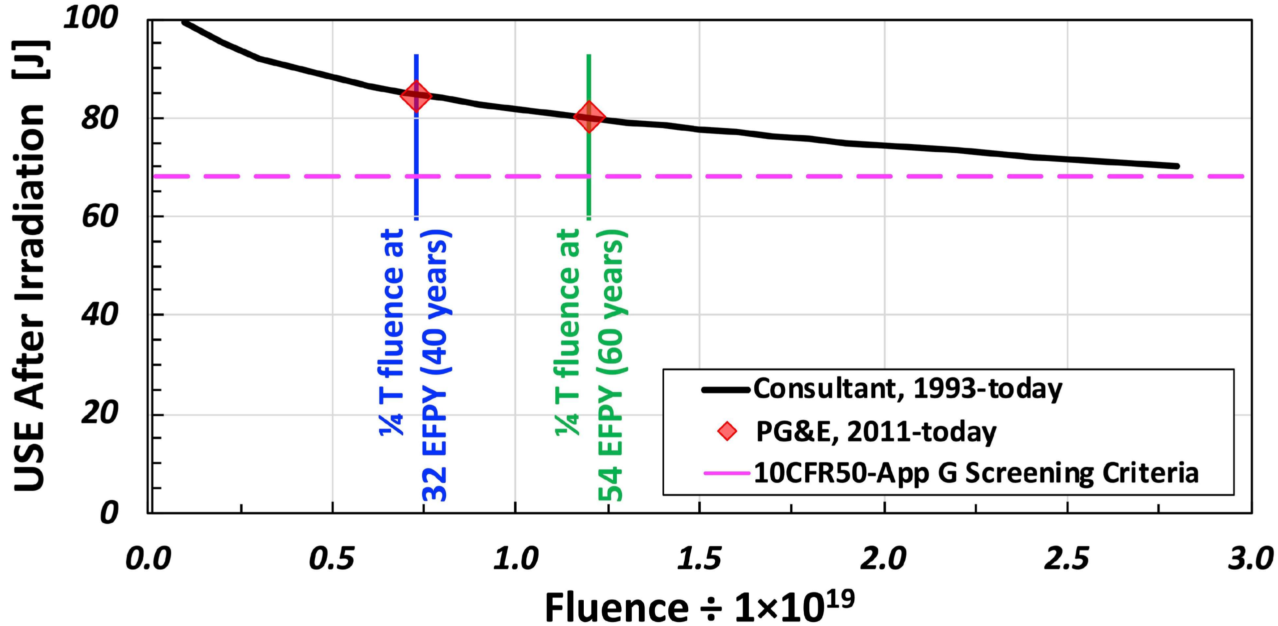
- NRC requires that the RG1.99 model (black curve) be adjusted to bound all credible data (red curve)
  - For Unit 1 the adjustment is minor
- NRC is silent on the use of similar sister plant data for USE assessments
  - USE data from sister plants not typically used

Years	Capsules	Credible?	Curve
1985-1993	none, S	No	Black
1993-2003	S, Y	Yes	Red
2003-now	S, Y, V	Yes	Red



NRC Required Analysis for USE

- This analysis suggests that Unit 1 will not fall below NRC's 68J screening criteria until long after 60 years
- My calculations validate those of PG&E from 2011 until today
  - PG&E made a small error in 2003 on credibility assessment for USE. This was identified in an NRC "request for additional information" and corrected
  - Error was in conservative direction. Negligible impact on prediction.



Supplemental Analysis
Objective

Use more data and more modern analytical techniques* to help inform judgements concerning

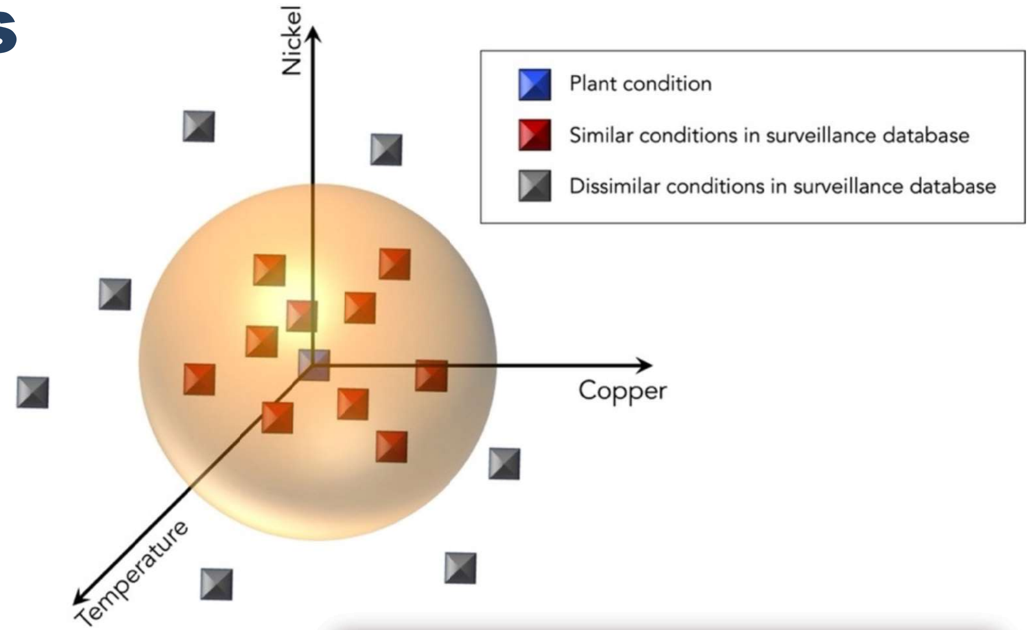
- Confidence in existing techniques
- Need, or not, for additional testing and analysis

* Techniques not yet endorsed by ASME or NRC

Supplemental Analysis

More Data

- Used a machine-learning inspired technique to identify data with similar embrittlement characteristics to the Unit 1 weld
  - Intent is the same as for “sister plants,” but based on more objective / physically based similarity metrics
  - Similarity based on copper, nickel, and irradiation temperature
  - Used international ASTM database

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STRUCTURALMAT

Application of a Machine Learning Approach Based on Nearest Neighbors to Extract Embrittlement Trends from RPV Surveillance Data

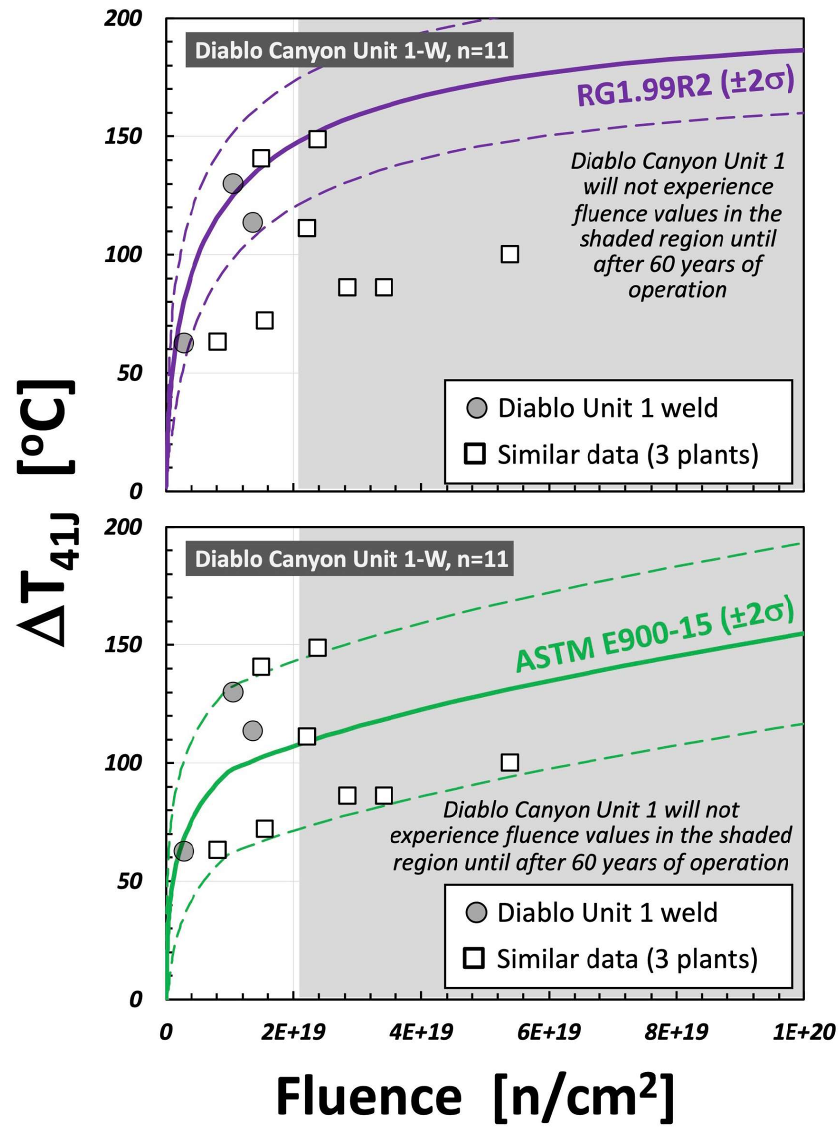
Mark Kirk, Yoshinori Hoshimoto, Akiyoshi Nomoto

https://www.sciencedirect.com/science/article/abs/pii/S0022311522003725?via%3Dihub

Supplemental Analysis

More Data

- Data similar to Diablo Canyon 1 weld:
  - Palisades
  - Two PWRs in Germany
- Shows NRC curve provides a conservative representation of data similar to the Diablo Canyon Unit 1 weld



Supplemental Analysis

Modern Analytical Techniques

- For PTS and P-T limits (ΔT_{41J} data)
  - Used **draft** ASME Code Case N-914
  - Permits use of fracture toughness data (available for Diablo weld in EPRI MRP-127)
  - Permits use of a modern trend curve (ASTM E900-15)
  - **Now in review: not yet approved by either ASME or NRC**
- For USE
  - Used additional data with standard RG1.99R2 analysis process
  - Advanced techniques using fracture toughness data are available [RG1.161, ASME SC-XI App-K] but were not used in this study

DRAFT CODE CASE N-914

Section XI Record No. 19-1113

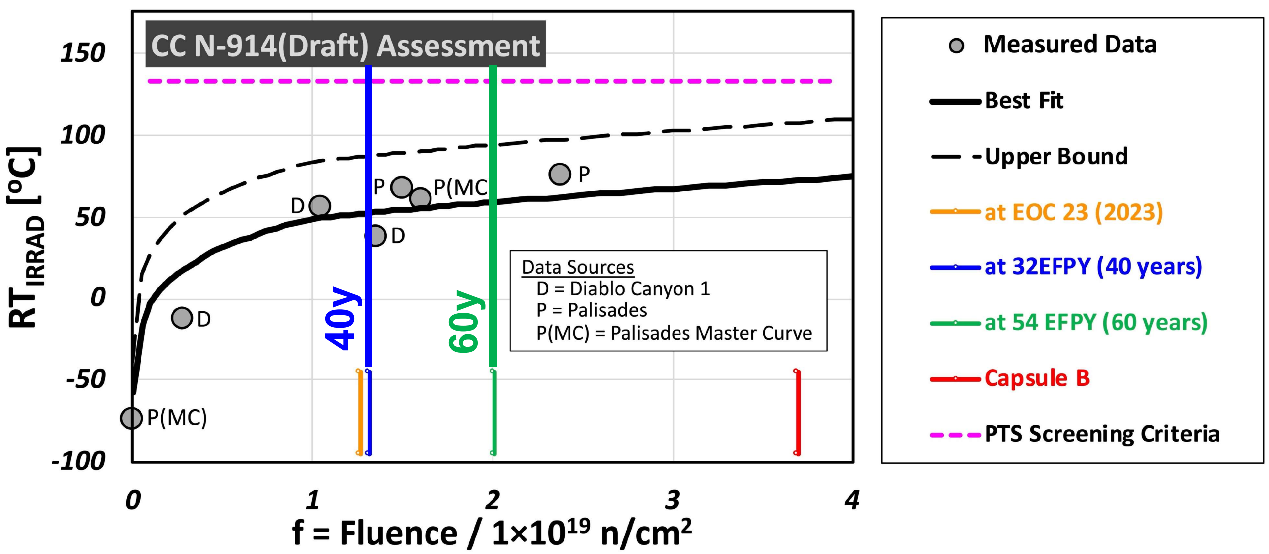
Accounting for the Effect of Embrittlement on Fracture Toughness Properties Used in Evaluations of Pressure Boundary Materials in Class 1 Ferritic Steel Components, Section XI, Division 1

https://www.epri.com/research/products/000000003002020911

Supplemental Analysis

PTS

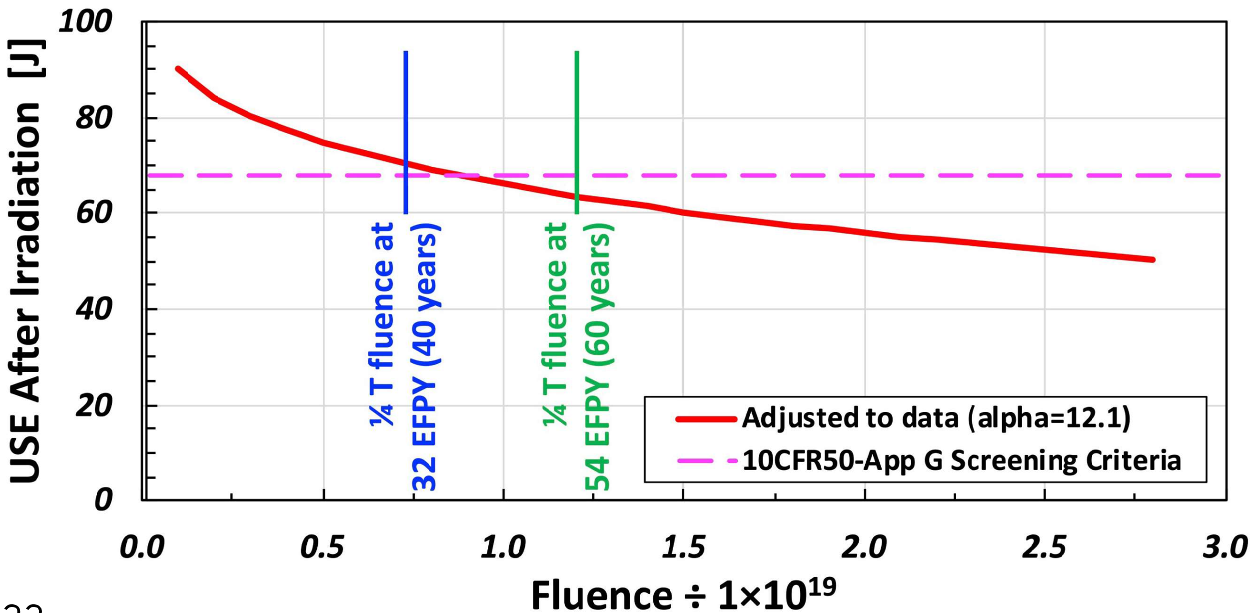
- Analysis shows larger distance from the PTS screening criteria than standard NRC analysis (larger margin)
- Fracture toughness data demonstrates the level of conservatisms in the current NRC approach



- Greater margin exists than suggested by NRC approach
- PTS screening criteria not exceeded until well after 60 years of operation

Supplemental Analysis
USE

- Consideration of similar USE data & the RG1.99R2 method suggests Unit 1 will fall below the NRC 68J screening criteria before 60 years (≈ 2029-2030)
- This could be addressed by performing an "equivalent margins analysis" (EMA) following RG1.161
- EMAs performed on other plants show, without exception, the acceptability of lower USE values



- Unit 1 may fall below USE screening criteria by ≈2030
- Further analysis using RG1.161 very likely to demonstrate acceptability through 60 years
- Screening criteria are not failure conditions

4. Answers to Public Concerns

Part 1 Report
Section 3

- ✓ Credibility
- Sister Plant Data
- RPV beltline inspections
- Alternate testing methods (nano indentation)
- Alternative Charpy analysis method
- Aspects of the RPV analysis methodology
- Deficient materials

Sister Plant Data

- **<u>Concern</u>**
  - *"the accumulation of damage depends upon the temperature history of the component, i.e., on the power level history"*
  - *"the complex nature of radiation embrittlement ... is idiosyncratic to individual reactors and may change unexpectedly over time"*

- **Review on sister plant data**
  - Provides similar data to the plant of interest
  - Required for PTS
  - No guidance for USE
- **Plant temperature history**
  - Temperature affects embrittlement
  - Most time plant is at 100% power (≈ 285 ^{o}C)
  - Plant is at higher temperature at lower power
  - 100% power temperature is used (conservative)
- **Reactor specific variables other than temperature**
  - Magnitude of effect on ΔT_{41J} small
  - Existing trend curves predict embrittlement in many types of light water reactors well
  - **Idiosyncratic** plant-specific effects not seen

RPV Beltline Inspections

Concerns

- The small number of indications found by UT inspections of Unit 1 are not plausible. Many more flaws found in Belgian RPVs and in predictions of models (FAVOR & GRIZZLY)
- The UT inspection interval permitted is too long, especially because Unit 1 is significantly embrittled
- Material damage by hydrogen embrittlement not considered

- **Small number of indications not plausible**
  - Reason for flaws in Belgian reactors not plausible for Unit 1
  - FAVOR and GRIZZLY models ultra-conservative
  - Indication density in Unit 1 similar to industry experience
- **UT inspection interval too long**
  - Embrittlement does not cause cracking
  - UT cannot detect embrittlement
  - Unit 1 inspection interval is appropriate and allowed by ASME & NRC
- **Hydrogen embrittlement not considered**
  - No evidence of hydrogen cracking in a light water RPVs
  - The stainless-steel liner protects the RPV steel
  - Liner cracking prevented by water chemistry control (low oxygen)

Alternative Testing Methods

Concern

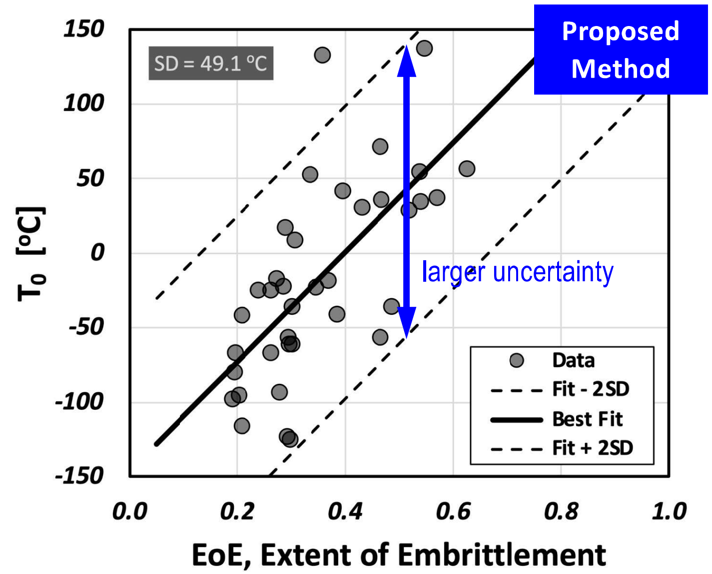
- Proposal to use Nano-Indentation data to further investigate the embrittlement status of the Diablo Canyon Unit 1 weld samples
- Technique developed by Prof. Hosemann at UC Berkley

- Prof. Hosemann's work is well documented in the literature
- Hardness can be related to fracture toughness using multiple correlations
  - Hardness → strength
  - Strength → fracture toughness

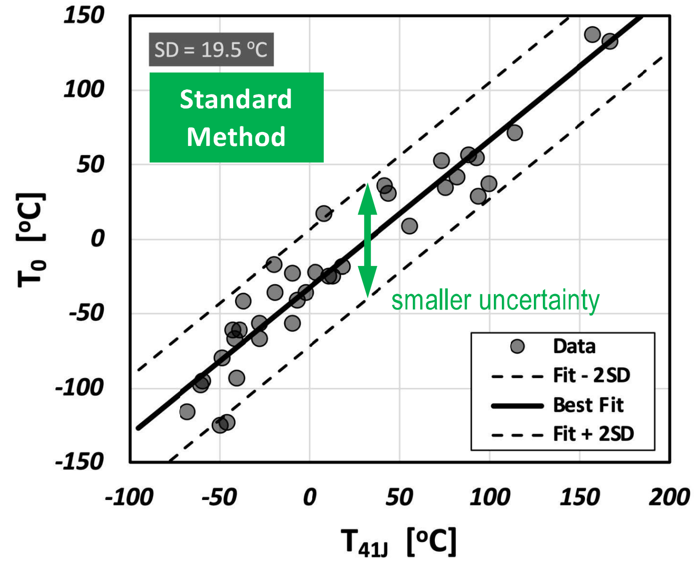
  Each correlation introduces uncertainty to the final predicted value.
- No precedent exists for using fracture toughness inferred from hardness in regulatory decisionmaking
- Lack of precedent and compounding uncertainties complicates interpretation of any data collected
- If supplemental data are needed, direct fracture toughness (T_0) testing using the "mini-CT" specimen & ASTM E1921 for which there is regulatory precedent would produce clearer outcomes

Alternative Charpy Analysis Method

- **<u>Concern</u>**
  - Existing methodology not sufficiently accurate
  - Prof. Macdonald proposed a new Charpy metric (Extent of Embrittlement: EoE)



- I assessed the accuracy of Charpy metrics (T_{41J} and EoE) by evaluating how well they predict the <u>fracture toughness</u> transition temperature (T_0)
- Used data from a wide variety of RPV steels before and after irradiation (see https://doi.org/10.1115/1.3109987)
- EoE predicts fracture toughness transition temperature less accurately than T_{41J}



RPV Assessment Methodology

Concerns

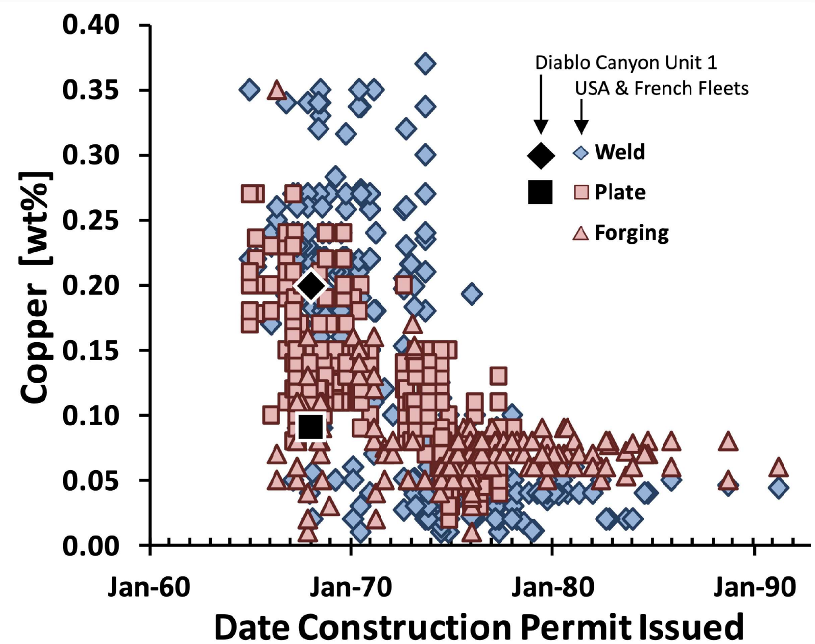
- *"Why has no account been taken for the stainless-steel **liner** in determining the susceptibility of the RPV to brittle fracture & hence a LOCA?"*
- *"Why has no attention been given to low temperature thermal **annealing** of radiation damage"*
- *"there were still concerns expressed by PG&E … regarding the **nozzle** shell welds* [who admitted] *… that the nozzle shell welds and related components may not meet fracture toughness limits through the entire 20-year extension"*

- **Cladding (liner)**
  - Defects accounted for conservatively by the PTS screening criteria
  - High toughness stainless steel cladding is ignored by structural integrity calculations (conservative)
- **Low temperature annealing**
  - Effect quantified directly by surveillance data
  - Trend curves reflect temperature effects
- **Accounting for steels in nozzle course**
  - Apparent misinterpretation of NRC guidance on nozzles
  - PG&E demonstrated nozzle materials do not restrict plant operability more than the beltline weld

Deficient Materials

- **Concern**
  - "*There are known metallurgical flaws in the Unit 1 reactor vessel, excessive copper and nickel impurities in welds and plate metals*"
  - *"It is well documented that there were engineering errors made in the metallurgical specifications of Unit 1 plate and weld alloys*"

- Unit 1 steel chemistry typical before 1973
- Specifications based on contemporary knowledge
- Embrittlement of higher copper materials accounted for
  - Plants with more copper than Unit 1 operate safely
  - 25 units with more copper Unit 1 remain in service today



Wrap Up

Wrap Up

- **PG&E's current reactor vessel integrity calculations were confirmed and validated**
  - Unit 1 meets the NRC requirements for reactor vessel integrity though 60 years of service
  - Credibility of data was was correctly assessed
  - Sister plant data was correctly used
  - Deferrals of Capsule B testing were appropriate
  - Vessel inspection schedule is appropriate
- **Capsule B is scheduled for withdrawal during the 2025 refueling outage**
  - Removal is recommended by 2028 to be consistent with NRC guidance
  - These new data may change the outcome of current RPV integrity assessments
- **Supplemental analyses performed using more data and modern analytical techniques**
  - <u>Pressurized thermal shock</u>: Shows NRC screening criteria very unlikely to be exceeded in 60 years
  - <u>Upper Shelf Energy</u>
    - Shows USE may fall below NRC screening criteria by 2029 or 2030
    - A more accurate analysis following NRC guidance is highly likely to show that acceptable margins will remain past these dates

Thank you for your time!